ornl

OAK RIDGE
NATIONAL
LABORATORY

MARTIN MARIETTA

NUREG/CR-3885
Volume 1
ORNL/TM-9267/V1

High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Accident Evaluation Quarterly Progress Report, January 1–March 31, 1984

S. J. Ball
J. C. Cleveland
R. M. Harrington
I. Siman-Tov
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

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

Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161

Available from
GPO Sales Program
Division of Technical Information and Document Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

NUREG/CR-3885
Volume 1
ORNL/TM-9267/V1
Dist. Category R8

HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION QUARTERLY PROGRESS REPORT, JANUARY 1—MARCH 31, 1984

S. J. Ball, Manager
J. C. Cleveland
R. M. Harrington
I. Siman-Tov
J. H. Wilson

Manuscript Completed — August 7, 1984
Date Published — August 1984

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 Agreement 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

PRIOR HTGR SAFETY REPORTS

Quarterly Progress Reports

Ending date	Designation
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

Topical Reports

S. J. Ball, *ORECA-I: A Digital Computer Code for Simulating the Dynamics of HTGR Cores for Emergency Cooling Analyses*, ORNL/TM-5159 (April 1976).

T. W. Kerlin, *HTGR Steam Generator Modeling*, ORNL/NUREG/TM-16 (July 1976).

R. A. Hedrick and J. C. Cleveland, *BLAST: A Digital Computer Program for the Dynamic Simulation of the High Temperature Gas Cooled Reactor Reheater-Steam Generator Module*, ORNL/NUREG/TM-38 (August 1976).

J. C. Cleveland, *CORTAP: A Coupled Neutron Kinetics-Heat Transfer Digital Computer Program for the Dynamic Simulation of the High Temperature Gas Cooled Reactor Core*, ORNL/NUREG/TM-39 (January 1977).

J. C. Cleveland et al., *ORTAP: A Nuclear Steam Supply System Simulation for the Dynamic Analysis of High Temperature Gas Cooled Reactor Transients*, ORNL/NUREG/TM-78 (September 1977).

S. J. Ball et al., *Evaluation of the General Atomic Codes TAP and RECA for HTGR Accident Analyses*, ORNL/NUREG/TM-178 (May 1978).

J. C. Conklin, *ORTURB: A Digital Computer Code to Determine the Dynamic Response of the Fort St. Vrain Reactor Steam Turbines*, ORNL/NUREG/TM-399 (March 1981).

S. J. Ball et al., *Summary of ORNL Work on NRC-Sponsored HTGR Safety Research, July 1974–September 1980*, ORNL/TM-8073 (March 1982).

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 January 1–March 31, 1984. 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, JANUARY 1–MARCH 31, 1984

S. J. Ball, Manager

J. C. Cleveland   I. Siman-Tov
R. M. Harrington   J. H. Wilson

ABSTRACT

Modeling and code development work for predicting source terms for the Fort St. Vrain and 2240-MW(t) reactors continued and investigations and modeling work for small modular High-Temperature Gas-Cooled Reactor designs were begun. Fission-product transport experiments to determine coefficients for diffusion through graphite have included studies with Ag, Rh, and Pd. The review of an FSV technical specification (tech spec) on limiting maximum core temperature involved code development and FSV data analysis, leading to new proposed limiting conditions and validation tests.

---

1. HTGR SYSTEMS AND SAFETY ANALYSIS

S. J. Ball

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 code development and verification efforts for the Fort St. Vrain (FSV) and lead plant HTGRs and initiation of modeling work on modular HTGRs. Fission-product (FP) release and transport experiments and technical assistance on an FSV tech spec review were continued.

1.1 Development of the ORECA Code for Simulating FSV Severe Accident Transients

R. M. Harrington

Minor code modifications were made as required for the scoping studies for the 2240-MW(t) lead plant and FSV HTGRs (reported in Sect. 1.5).

1.2 Fission-Product Release from HTGRs

J. H. Wilson   R. L. Towns

The objective of this task is to generate experimental data required for the analysis of FP release in HTGR severe accidents. Initial efforts involve the determination of FP vapor pressures and diffusion rates through graphite. The experimental procedure involves measuring the rate of loss at high temperatures from a powdered graphite and simulated FP mixture which has been placed in a 6.4-mm-diam graphite tube. As the products diffuse through the tube wall, they are transported through a cold collection tube by argon carrier gas.

As discussed in the last quarterly report, the initial experimental runs had been made using mixtures of several FPs. Loss rates were determined by neutron activation analysis, which involved a relatively long turnaround time. To generate data more quickly, and at a lower cost, experiments were begun in which only one FP at a time was studied, thus allowing the rate of loss to be determined simply by weight loss of the sample tube. After building a data base for single elements or compounds, it is planned to again study mixtures of FPs in order to more closely match the chemical species actually present.

To determine both the vapor pressure and the coefficient of diffusion in graphite (assuming the primary transport mode is gas phase diffusion) for a particular species, the initial intent was to measure the rate of loss of a species from the sample tube as a function of the flow rate of the argon carrier gas. The desired parameters could then be evaluated by minimizing differences between the experimental data and the values as predicted by the pertinent theoretical equation. However, calculations indicated that the dependence of the rate of loss on flow rate would not be strong enough to allow the accurate determination of the two unknown parameters. The modified technique involved measuring the rate of loss as a function of temperature at a constant, high value of argon flow rate. The diffusion coefficient and the vapor pressure of the species of interest could then be determined from a comparison with results from experiments with a reference species, such as silver, whose vapor pressure behavior with temperature is well characterized. The equation utilized in this comparison is

$$\text{rate of loss} = K\, T^{0.5}\, P/(\Omega\sigma^2),$$

where $T^{0.5}$ includes the temperature dependence of the diffusion coefficient, P is the vapor pressure, σ is a collision diameter, and Ω is a dimensionless function of temperature and of the intermolecular potential field between the diffusing species and argon. The "constant" K includes total pressure, a factor for converting from concentration to partial pressure, a factor involving the physical dimensions of the sample tube and the porosity of the graphite tube wall, and a factor involving molecular weights and a constant from the theoretical equation for diffusion coefficient. Having determined $K/\Omega\sigma^2$ from the experimental data for the reference species (since its vapor pressure is known), $K/\Omega\sigma^2$ for the species of interest is calculated by making necessary adjustments to account for the effects of differences in molecular weight, molecular diameter, and interaction potential on K, Ω, and σ^2. Vapor pressure as a function of temperature is then determined by fitting the rate of loss equation to the experimental data. That is, assuming vapor pressure varies exponentially with the reciprocal of absolute temperature, the heat of vaporization (ΔH_v) is obtained when above the melting point and the heat of sublimation (ΔH_s) when below the melting point.

Rate of loss measurements made with silver produced a ΔH_v of 61,600 cal/gmol, which agrees very well with the literature value of 62,600 cal/gmol.[1] After backing out unrelated factors, such as the physical dimensions of the sample tube, the effective (i.e., the tube wall porosity has not been factored out) vapor-phase diffusion coefficient of silver through the graphite tube wall is, for example, 0.042 cm^2/s at a temperature of 1600 K and 1 atm pressure. Assuming a graphite porosity of 0.1, this value is very close to the theoretical prediction. Thus, the experimental data for silver appear to substantiate the assumption of gas-phase diffusion.

Measurements with Cu, Rh, and Pd, all at higher temperatures than with Ag, produced values for ΔH_v significantly lower than those reported in the literature.[1] These differences appeared not to result from possible experimental problems, as the data were reproducible and the semilog plots of rate of loss vs the reciprocal of absolute temperature were linear, as shown in Fig. 1 (the term $T^{0.5}$ does not produce significant curvature in the semilog plot). Nevertheless, to check for the effects such as mass transfer limitations at the higher operating temperatures (which could lower the apparent ΔH_v), experiments with palladium were conducted using a sample tube with a thicker wall (0.79 cm vs 0.16 cm). Palladium runs were also made at one temperature level with the thinner wall sample tube in which the palladium concentration (in the mixture with powdered graphite) was increased from 10 to 40 wt %. As expected, the results with the new sample tube showed lower rates of loss of the palladium at comparable temperatures. However, the ΔH_v was the same as had been obtained previously. Also, the rate of loss from the more concentrated sample was the same as that at the 10 wt % level. Consequently, mass transfer limitations do not appear to be present.

Attempts to resolve the discrepancy between the experimental and the literature ΔH_v values will be made before continuing the diffusion coefficient and vapor pressure measurements. A possible explanation for the discrepancy is interaction with the graphite. However, copper (which was run as a reference material) would be expected to behave like silver and exhibit little or no interaction.

1.3 Review of FSV Reactor Technical Specification on Limiting Maximum Core Temperatures

S. J. Ball

Investigations of the proposed amendments to FSV Tech Spec LCO 4.1.9 continued. The intent of this tech spec is to ensure that core temperatures are properly limited during startup and shutdown (typically between 0–15% power and flow). The object of this task is to provide technical

ORNL–DWG 84–5912 ETD

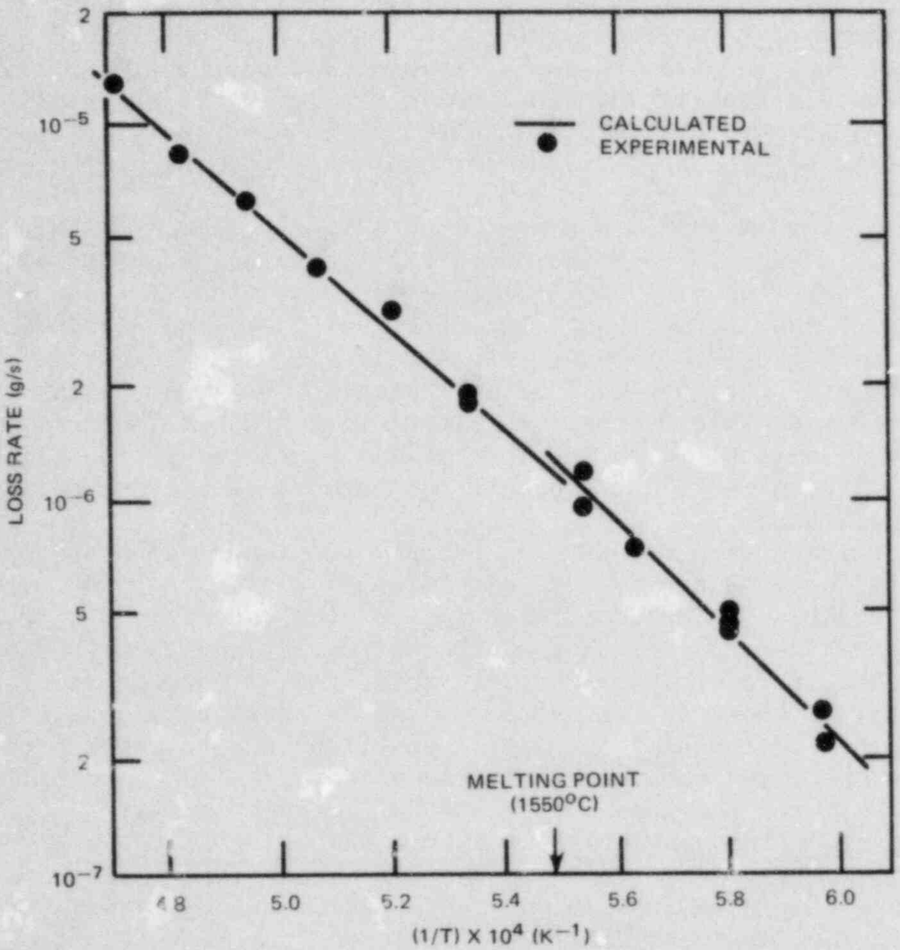

Fig. 1. Experimental results for palladium diffusion.

support to the Nuclear Regulatory Commission (NRC) in their review of the changes to LCO 4.1.9 proposed by Public Service Company of Colorado (PSC). This work is supported in part by NRC Region IV in Arlington, Texas.

A review was made of the GAT reanalysis of the technical bases of the tech spec which was performed for PSC. Our major concern was that the tech spec limitations and the supporting analysis are based entirely on the idea of avoiding laminar flow instability without showing due consideration for other combinations of operating conditions that could lead to fuel overheating.

Further refinements were made to the code verification version of ORECA-FSV[2] to facilitate study of the tendencies for flow stagnation and fuel overheating. A watchdog routine was added to ORECA for detecting violations of LCO 4.1.7, which is also designed to prevent excessive fuel temperatures. Simulations of both bounding and "typical" shutdown and startup scenarios were run to get a better idea of operating limits that will ensure safe maximum fuel temperatures. Special tests that could be used to validate proposed limiting conditions were outlined.

Plant data logger records and backup information for the November 1983 startup and January 1984 shutdown were received from PSC, along with the "HISTORY" program that PSC uses to process the data. Portions of the startup and shutdown runs were set up on the ORECA code, and the calculations generally corresponded well with the data. The ORECA routines for detecting tech spec violations were rewritten for the HISTORY code.

1.4 Development of the BLAST Steam Generator Dynamics Code

J. C. Cleveland

Information was received from Kernforschungsanlage (KFA) presenting their comparison of BLAST code[3] predictions with measured FSV steam generator steady state and transient data from the November 9, 1981, loop shutdown transient. The results compared well with measured data. Specific questions regarding the comparison were sent to KFA. Additional information regarding the cause of this transient was supplied to KFA. This information was based on the FSV Monthly Operations Report for November 1981 and on ORNL's review of the event.

The most recent ORNL version of BLAST was also sent to KFA.

1.5 Scoping Studies for Fission-Product Redistribution for the 2240-MW(t) Lead Plant and FSV HTGRs

R. M. Harrington S. J. Ball

One assumption made in all of our unrestricted core heatup accident (UCHA) analyses to date has been that the afterheat terms for the refueling region nodes are independent of fuel failure. This has generally been considered a conservative assumption in terms of the predictions of peak temperatures and total fuel failure fraction, since allowing the failed fuel FP heat sources to go elsewhere should reduce the hottest fuel heatup rates.

For the 2240-MW(t) lead plant, the ORECA code[2] predictions of core temperatures and fuel failure histories for three bounding UCHA calculations are shown in Figs. 2 and 3. Case A is the reference calculation in which the afterheat terms are independent of fuel failure (i.e., the FP heating in a node is dependent on the initial region peaking factor and time only).

The least conservative bounding case would be one in which FPs (as heat sources) leave the core entirely as soon as the fuel fails. This is shown as Case B, where the grouping of FPs into volatile and mobile (which leave) and stationary (which stay) was taken from a GAT categorization for their CORCON code.[4] Note that the predicted peak temperature is reduced by ~640°C (1150°F), and the total fuel failure fraction (after 120 h) is reduced from 84 to 69%. Based on Brookhaven National Laboratory (BNL) observations[5] that fission products in GAT's "stationary" group exhibit considerable mobility at high tempertures, it is possible these reductions could be even larger.

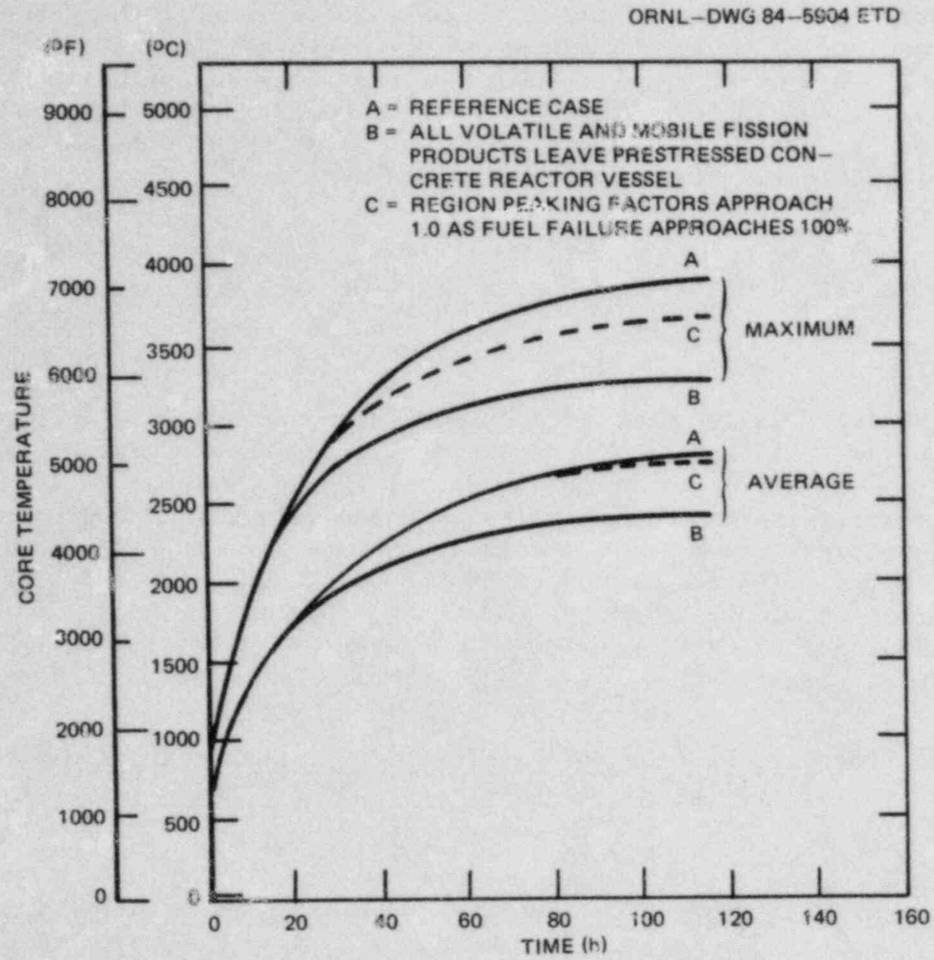

Fig. 2. Predicted core temperatures during hypothetical 2240-MW(t) UCHAs (liner cooling system operational).

A third bounding case (C) was run in which all the FPs were assumed to stay in the core region but were redistributed from the hotter to cooler regions. This effect was approximated by adjusting the radial and axial power peaking factors to approach unity as the total core fuel failure fraction approaches unity. As expected, the peak fuel temperature is lower (by 220°C or 400°F) than that of the reference case. The effect on fuel failure fraction, however, is to increase the failure rate somewhat over the reference case (88% vs 84% at 120 h) because the relocated FPs cause some of the cooler fuel to heat up more than it would otherwise.

For the FSV plant, the ORECA-FSV code predictions of core temperatures and fuel failure histories for three bounding UCHA calculations are shown in Figs. 4 and 5. The hypothetical UCHA postulated for these examples involves the loss of forced circulation of the helium primary coolant with continued operation of the liner cooling system. Case A is the reference calculation in which the afterheat terms are independent of fuel failure (i.e., the FPs do not relocate).

ORNL-DWG 84-5905 ETD

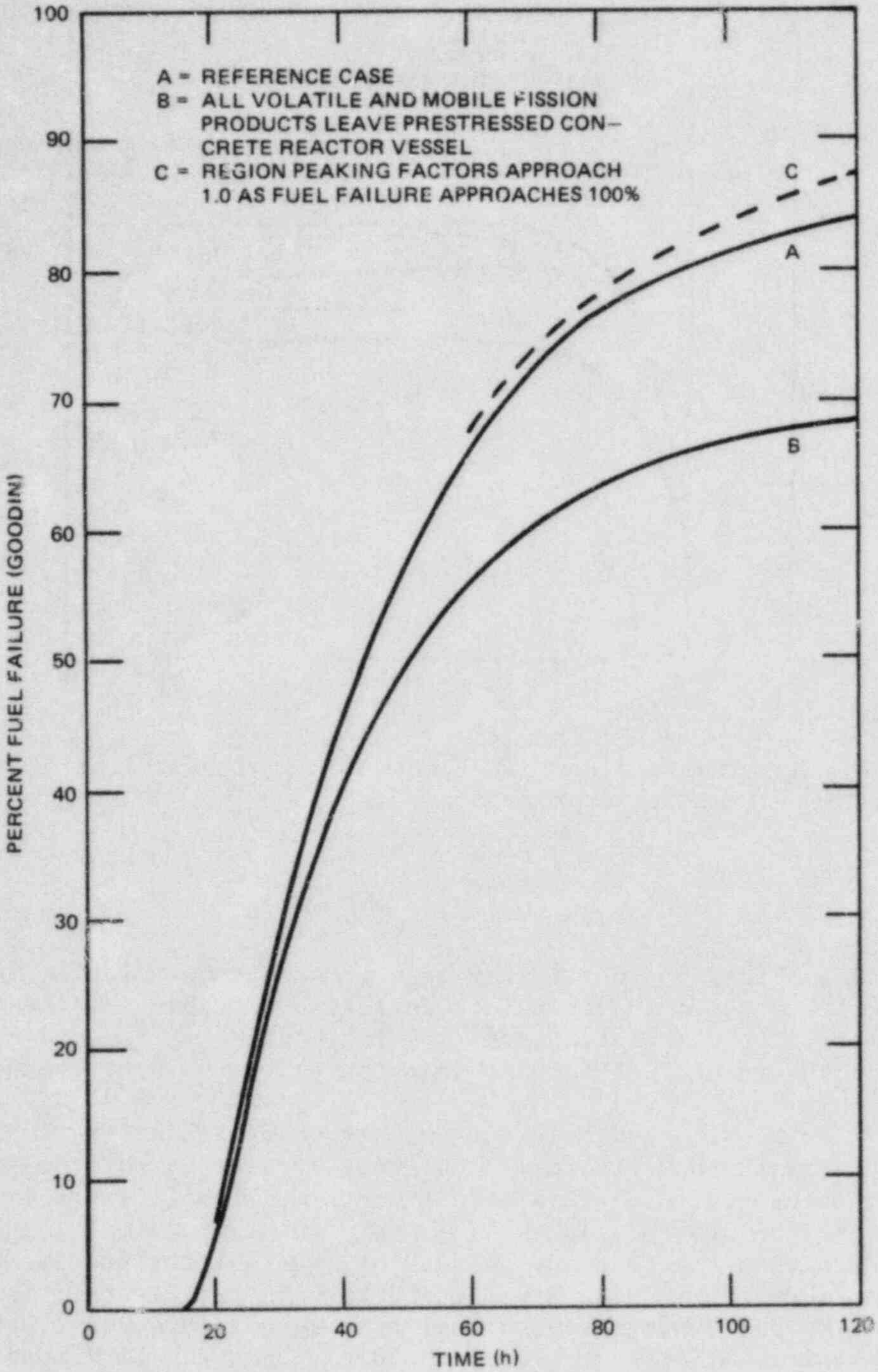

Fig. 3. Predicted fuel failure during hypothetical 2240-MW(t) UCHAs (liner cooling system operational).

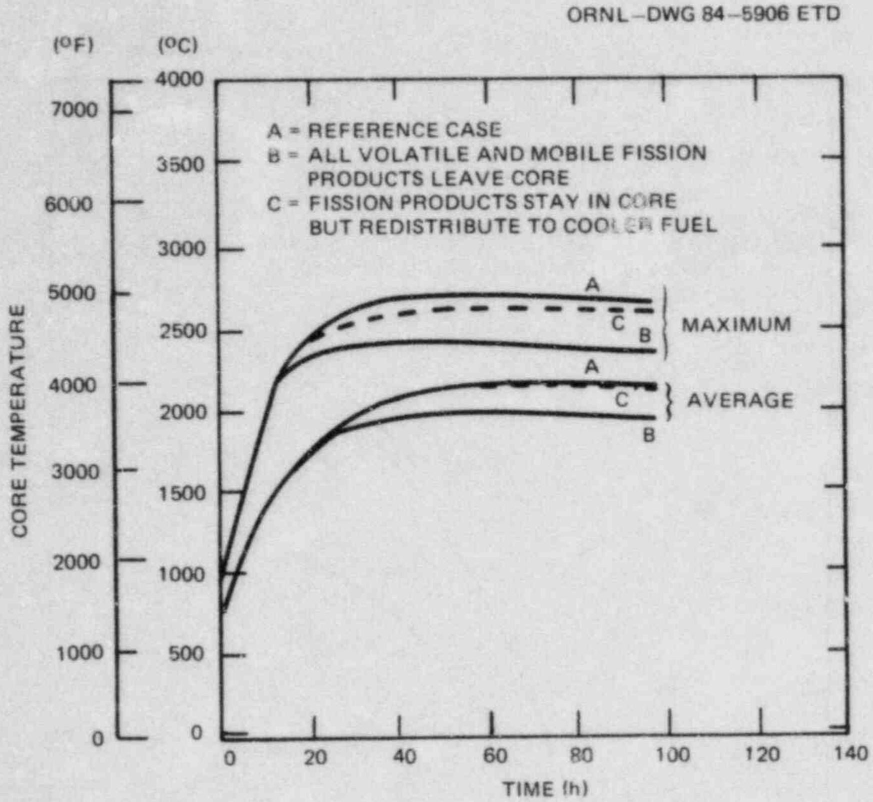

Fig. 4. Hypothetical Fort St. Vrain UCHAs without liner cooling system failure — fuel temperatures.

Case B in Figs. 4 and 5 is the case in which the volatile and mobile FPs leave the core after fuel failure. For this least conservative bounding case, the calculated peak fuel temperature is reduced by 297°C (535°F) and the total fuel failure fraction (after 96 h) is reduced from 70 to 50%.

Case C in Figs. 4 and 5 is the case in which all of the FPs were assumed to stay in the core region but were redistributed from the hotter to cooler regions. This effect was simulated in the same manner as it was for the 2240-MW plant. The peak fuel temperature is 97°C (175°F) lower than the reference case, but the fuel failure is about 4% higher than the reference case. The FSV results are generally similar to the 2240-MW case, but the peak fuel temperatures are lower because the FSV core is smaller and more of the decay heat can be removed to the liner cooling system.

ORNL-DWG 84-5907 ETD

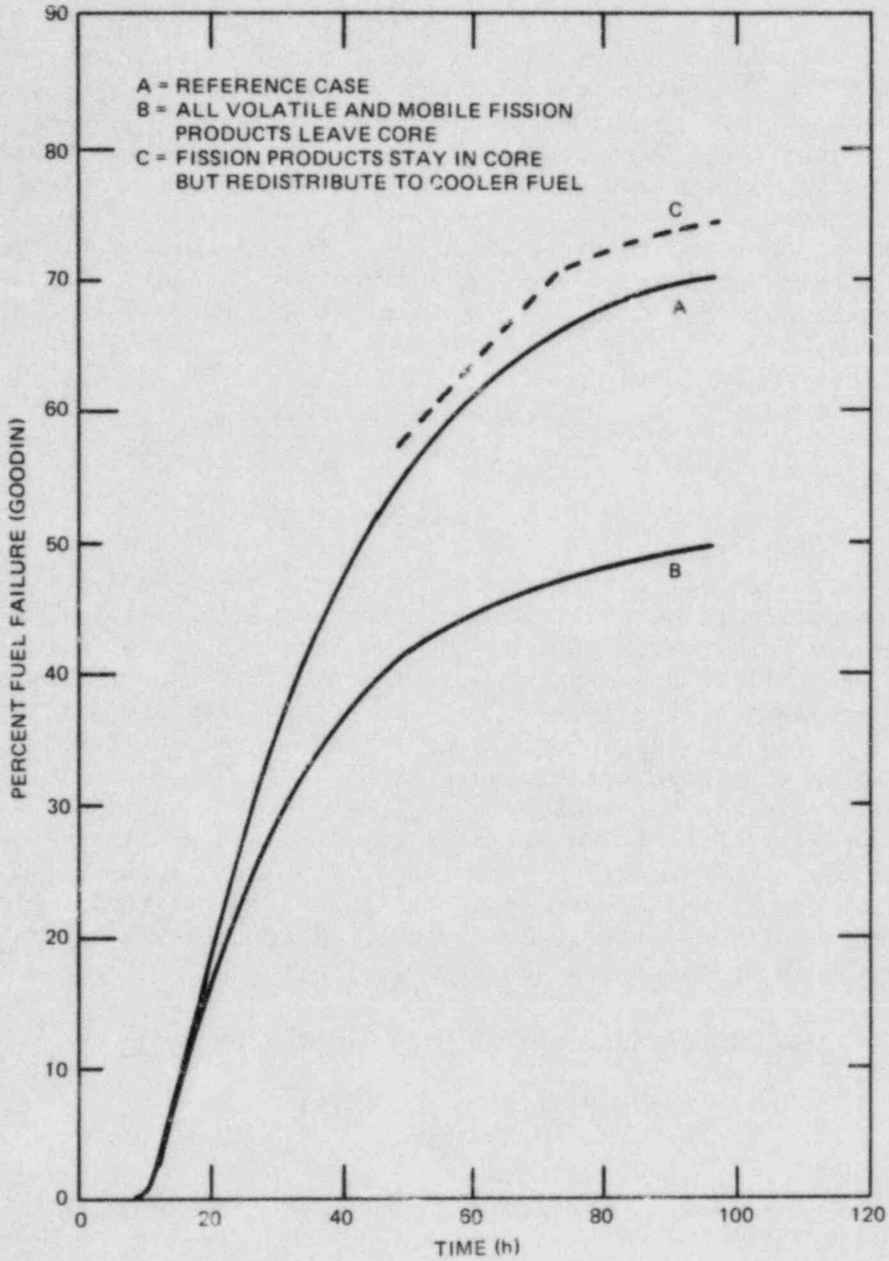

Fig. 5. Hypothetical Fort St. Vrain UCHAs without liner cooling system failure — fuel failure.

1.6 Model and Code Development for Fission-Product Redistribution During Severe Accidents

I. Siman-Tov

Work continued on code development. A model is being prepared for the ORIGEN-S code.[6] This model will result in the initial FP inventories as a function of time during either a 6-year or a 4-year cycle of normal full-power operation. Fission-product decay and production yields after accident initiation will be obtained as well as their equivalent heat generation rates.

In the process of preparing data for this model, contacts were made to try to obtain valid cross-section and nuclide concentration sets needed for the ORIGEN-S model. At this point all the required data have not been located.

1.7 Cooperative Programs with the Federal Republic of Germany (FRG)

J. C. Cleveland

Rheinisch Westfalischer Technischer Überwachungs Verein e.V. (RWTÜV) was contacted concerning their schedule for preparing a safety assessment report for the THTR, which will be used by the West German Ministry of the Interior (BMI) as a basis for granting the THTR rise-to-power license. ORNL has been asked to review RWTÜV's safety analysis as part of a recently approved NRC-FRG information exchange agreement. RWTÜV indicated that they plan to complete the report in May, and BMI plans to grant the rise-to-power license by about October 1984. K. D. Paul (RWTÜV) indicated that discussion with ORNL some time in the June-August time frame concerning their THTR safety review would be quite helpful. THTR has successfully completed the zero-power tests and is currently shut down for completion of the water/steam circuit. Rise-to-power tests are planned to begin by about the end of this year.

1.8 Investigations of Modular HTGR Dynamics

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

Initial investigations were made on data availability and dynamic modeling techniques for modular HTGR designs with pebble bed fuel. Both FRG and GAT reactor concepts are being considered. Scoping calculations were set up for both at-power and long-term cooldown transients using the CSMP language. A single-channel point-neutron-kinetics model derived previously for the Arbeitsgemeinschaft Versuchs Reaktor (AVR) was updated, and sensitivity studies were run to check agreement with AVR rod job and flow ramp tests. The agreement was very good. A simplified three-dimensional (3-D) core plus primary circuit model was also developed and is currently being debugged. Problems of low flow, natural circulation, and core radial flow modeling are being investigated.

2. TRIP MADE UNDER PROGRAM SPONSORSHIP — MEETING WITH NRC ON LONG-RANGE PLANS FOR HTGR SAFETY RESEARCH, BETHESDA, MARYLAND

S. J. Ball

S. J. Ball attended a meeting on February 28 and 29, 1984, with NRC, DOE, and other contractor laboratory participants (BNL, Los Alamos National Laboratory, and Idaho National Laboratory). NRC was represented by their Office of Research, Region 4, and licensing divisions.

The purpose of the meeting was to first learn about DOE plans for HTGR development and then determine what the appropriate licensing needs would be. Given that as input, the objective was to determine a suitable research strategy. The DOE plan outline showed phasing out the 2240-MW(t) steam cycle/cogeneration plant design work and phasing in work on a half-size integrated plant and a small [$\sim$250-MW(t)] modular design. The intent is to choose between the three concepts by the end of FY 1985.

The Office of Nuclear Reactor Regulation drew up a preliminary list of research needs for HTGR licensing based on the revised DOE plan. All of the contractor labs were requested to revise their proposal submittals in view of the new guidelines.

REFERENCES

1. R. Hultgren, P. D. Desai, D. T. Hawkins, M. Gleiser, K. Kelley, and D. D. Wagnan, *Selected Values of the Thermodynamic Properties of Elements*, reproduced by the American Society for Metals, 1973.

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

3. R. A. Hedrick and J. C. Cleveland, *BLAST: A Digital Computer Program for the Dynamic Simulation of the High-Temperature Gas-Cooled Reactor Reheater Steam Generator Module*, ORNL/NUREG/TM-38, Union Carbide Corp. Nuclear Div., Oak Ridge Natl. Lab., August 1976.

4. K. E. Schwartztrauber and F. A. Silady, *CORCON: A Program for Analysis of HTGR Core Heatup Transients*, GA-A12868 (GA-LTR-13), July 1974.

5. C. A. Sastre, Brookhaven National Laboratory, personal communication, February 1984.

6. O. W. Hermann and R. M. Westfall, *ORIGEN-S, SCALE System Module to Calculate Fuel Depletion, Actinide Trasmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms*, ORNL/NUREG/CSD-2/V2, Union Carbide Corp. Nuclear Div., Oak Ridge Natl. Lab, October 1981.

NUREG/CR-3885
Volume 1
ORNL/TM-9267/V1
Dist. Category R8

Internal Distribution

1-5.	S. J. Ball	27.	I. Siman-Tov
6.	N. E. Clapp	28.	R. S. Stone
7-11.	J. C. Cleveland	29.	H. E. Trammell
12.	J. C. Conklin	30.	C. F. Weber
13-14.	D. S. Griffith	31.	R. P. Wichner
15-19.	R. M. Harrington	32-36.	J. H. Wilson
20.	P. R. Kasten	37.	I&C Publications Office
21.	A. D. Kelmers	38.	ORNL Patent Office
22.	T. S. Kress	39.	Central Research Library
23.	T. B. Lindemer	40.	Document Reference Section
24.	A. P. Malinauskas	41-42.	Laboratory Records Department
25.	D. L. Moses	43.	Laboratory Records, RC
26.	J. P. Sanders		

External Distribution

44-47. Director, Office of Nuclear Regulatory Research, Nuclear Regulatory Commission, Washington, DC 20555

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

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

50-51. Technical Information Center, DOE, Oak Ridge, TN 37831

52-251. Given distribution as shown in category R8 (10-NTIS)

NRC FORM 335
(2-84)
NRCM 1102
3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

SEE INSTRUCTIONS ON THE REVERSE

1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any)
NUREG/CR-3885
ORNL/TM-9267/V1

2. TITLE AND SUBTITLE
High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Accident Evaluation Quarterly Progress Report, January 1 - March 31, 1984

3. LEAVE BLANK

4. DATE REPORT COMPLETED

MONTH	YEAR
August	1984

5. AUTHOR(S)
S. J. Ball, J. C. Cleveland, R. M. Harrington, I. Siman-Tov, J. H. Wilson

6. DATE REPORT ISSUED

MONTH	YEAR
September	1984

7. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Oak Ridge National Laboratory
P. O. Box X
Oak Ridge, Tennessee 37831

8. PROJECT/TASK/WORK UNIT NUMBER

9. FIN OR GRANT NUMBER
B0122

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

11a. TYPE OF REPORT
Quarterly

b. PERIOD COVERED (Inclusive dates)
January 1 - March 31, 1984

12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

Modeling and code development work for predicting source terms for the Fort St. Vrain and 2240-MW(t) reactors continued and investigations and modeling work for small modular High-Temperature Gas-Cooled Reactor designs were begun. Fission-product transport experiments to determine coefficients for diffusion through graphite have included studies with Ag, Rh, and Pd. The review of an FSV technical specification (tech spec) on limiting maximum core temperature involved code development and FSV data analysis, leading to new proposed limiting conditions and validation tests.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS

b. IDENTIFIERS/OPEN-ENDED TERMS

15. AVAILABILITY STATEMENT
Unlimited

16. SECURITY CLASSIFICATION
(This page)
Unclassified
(This report)
Unclassified

17. NUMBER OF PAGES

18. PRICE

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