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NUREG/CR-4402
Volume 1
ORNL/TM-9798/V1

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High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Accident Evaluation Quarterly Progress Report, January 1–March 31, 1985

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C. F. Weber
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Office of Nuclear Regulatory Research
Under Interagency Agreements DOE 40-551-75 and 40-552-75

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NUREG/CR-4402
Volume 1
ORNL/TM-9798/V1
Dist. Category R8

HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION QUARTERLY PROGRESS REPORT, JANUARY 1-MARCH 31, 1985

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NUREG/CR--4402-Vol.1
TI86 001938

Manuscript Completed -- September 26, 1985
Date Published -- October 1985

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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
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MARTIN MARIETTA ENERGY SYSTEMS, INC.
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U.S. DEPARTMENT OF ENERGY
under Contract No. DE-AC05-84OR21400

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PRIOR HTGR SAFETY REPORTS

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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, 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, JANUARY 1-MARCH 31, 1985

S. J. Ball, Manager

J. C. Cleveland C. F. Weber
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. A description and assessment report on Oak Ridge National Laboratory HTGR safety codes was issued.

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 and for fission-product (FP) redistribution during severe accidents in large HTGRs. FP release and transport experiments and technical assistance work on a Fort St. Vrain (FSV) technical specification (tech spec) review were continued. A report on Oak Ridge National Laboratory (ORNL) HTGR code assessments was issued.

1.1 Investigations of Modular HTGR Dynamics

R. M. Harrington S. J. Ball

Initial investigations were made of the Department of Energy (DOE) concept for the modular HTGR that uses the side-by-side (steam generator in a separate vessel beside the reactor vessel) design instead of the vertical in-line (VIL) design, where the steam generator is above the reactor in the same vessel.¹⁻³ The normal direction of primary coolant flow is downward in the side-by-side design: coolant enters at the top of the core, flows downward, and exits from the bottom of the core. Primary coolant flows upward through the core in the VIL design. In both designs, the fuel pebbles are introduced at the top of the core, removed from the bottom, and recycled back to the top; therefore, power peaking factors were assumed to be the same for both designs for the calculations reported here.

The side-by-side design has lower peak fuel temperatures because the coolant is introduced at the top where the freshest, and therefore the highest, power density pebbles reside. The full power steady state

axial temperature distributions are illustrated in Fig. 1 for the radially averaged coolant temperature and for fuel temperatures in the 0.86-m radius central region. The coolant temperatures at core inlet and outlet are, of course, the same as for the VIL because total core power and flow are the same for both designs. Even though the side-by-side design has a 96°C lower peak fuel temperature, its average central region fuel temperature is higher by 125°C.

A hypothetical depressurized loss-of-forced-circulation (LOFC) accident calculation was run for the side-by-side designs with initial temperature distributions as described previously. It was possible to do the depressurized case with minimal reprogramming because primary coolant natural circulation is close to zero and is, thus, not sensitive to the design of the coolant loop. This would not be the case for pressurized LOFC accidents in which natural primary circulation is an

Fig. 1. Coolant and fuel temperatures for the vertical in-line and the side-by-side modular HTGR designs.

important factor sensitive to loop design. The results of the depressurized LOFC show that the side-by-side design would have a 39°C lower peak central region fuel temperature than its VIL counterpart.

In other work, a review of the input data that describes the reactor core, reflector, and vessel configuration in the modular HTGR simulator was conducted. The input numbers were checked against the most recent numbers available for the in-line reactor vessel/steam generator configuration. With the exception of the reactor vessel and side reflector diameters, the input numbers were found to be correct. The code input was changed to reflect the more current data for reactor vessel and side reflector dimensions, and the worst-case depressurized heatup was repeated. The new results predict a 3°C higher hot-node fuel temperature.

1.2 Fission-Product Release from HTGRs

J. H. Wilson R. L. Towns

The objective of this task was 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. The experimental procedure consisted of measuring the rate of loss at high temperatures from a mixture of powdered graphite and simulated FP that had been placed in a 6.5-mm-diam graphite tube. The products diffused through the tube wall to its outer surface where they were picked up by argon carrier gas and transported through a cold collection tube.

Graphite tube diffusion experiments were performed with cerium and samarium over temperature ranges of 1400 to 1750°C and 1300 to 1600°C, respectively. Using silver as a reference material, vapor pressures and diffusion coefficients were calculated for these substances. This completed the diffusion experiments in which Ag, Ce, La, Pd, Pr, Rh, Ru, Sm, and Y were studied.

A draft of a report covering the work for the past 1.5 years was written. Initial plans were made to begin a study of the reaction of FPs with reactor vessel materials of construction.

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

C. F. Weber

Considerable progress has been made in reducing the running time for the FP transport routine by implementation of an implicit (Crank-Nicolson) numerical scheme for the diffusion calculation. Although this procedure requires solution of a matrix equation at each time step, it allows for much larger time steps, thus creating a significant overall savings of execution time. The calculations are still not realistic because fictitious parameters are still being used. However, the basic coding is in place and working, with minor refinements still being made.

Other work was focused on upgrading the parameters used in the FP transport calculation. A run of the ORIGEN-S code by other Computing

and Telecommunications Division personnel is available for the initial estimates of FP inventories. Because of suspicious input data (most notably in the cross sections), the values obtained should be considered very rough. However, without further development in ORIGEN cross sections, no improvements are available. Efforts have also begun to obtain better coefficients for diffusion and adsorption.

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

S. J. Ball

The ORNL review of the FSV tech spec limiting conditions for operation (LCO) 4.1.9, which is addressed at limiting maximum core temperatures during low power (0 to 15%) operation, is sponsored primarily by a subcontract with the Nuclear Regulatory Commission (NRC) Region IV. It is part of a large tech spec upgrade program, where the goal is to clarify and simplify the specs wherever feasible.

Further analyses were run of postulated startup and shutdown scenarios to observe the effects of column tilt factors on intraregion flow redistribution. More realistic operating limits (than are found in LCOs) were obtained from GA Technologies' core physics analyses for all expected loading cycles. The worst cases for region peaking factor (RPF) and column tilt factors were 1.80 and 1.36 (which occur in different cycles). A variety of flow/power ratios and orifice configurations have been tried, and no significant problems with flow redistributions, stagnation, or fuel overheating were detected to date with "reasonable" operating conditions.

Computer runs made with ORECA code,⁴ modified to include modeling of intraregion flow and heat transfer, led to several conclusions:

- 1) The current LCO 4.1.9, which bases its restrictions on avoidance of flow stagnation regimes, is not a sufficient basis for limiting operating parameters because stable intraregion flow regimes that can effectively choke off most of the flow to parts of the refueling region can result in core overheating.

- 2) On the other hand, for a reasonably wide operating path around normal conditions (with worst case RPFs and tilts), the extent of the intraregion distribution is small enough so that by limiting the power to flow ratio and measured outlet temperatures, the operators can be assured that no core overheating or flow stagnation will occur.

- 3) The effects of the relatively minor flow redistributions (noted in the simulations) on measured outlet temperatures are probably such that observations from relatively simple tests could indirectly verify the analysis.

- 4) It appears that a simpler, less restrictive, and verifiable tech spec is achievable. Preliminary investigations of appropriate verification tests were begun.

1.5 Assessments and Descriptions of
ORNL HTGR Safety Codes

S. J. Ball R. M. Harrington
J. C. Cleveland D. L. Moses

A letter report to the NRC entitled "ORNL HTGR Code Assessment Report" was completed and forwarded to the NRC Division of Accident Evaluation. This report consists of a description and assessment of 1) the codes ORTAP, ORECA, BLAST, CORTAP, and ORTURB developed at ORNL for transient thermal-hydraulic analysis of the HTGR plant or components; 2) recently developed ORNL simulations of the small modular pebble bed HTGRs; and 3) ORNL computer codes and data libraries that are applicable to the analysis of HTGR core physics. These writeups describe modeling techniques, code capabilities, and sample comparisons of code predictions with measured results.

2. TRIPS MADE UNDER PROGRAM SPONSORSHIP

S. J. Ball

S. J. Ball visited NRC-Bethesda, Maryland on January 30, 1985 to attend a DOE/Gas-Cooled Reactor Associates presentation on "Top Level Regulatory Criteria for the Standard HTGR." A second meeting was held with NRC and Brookhaven National Laboratory to discuss NRC needs for the HTGR Safety Handbook.

ORNL was asked to (and did) provide comments and suggestions on the proposed licensing plans.

REFERENCES

1. S. J. Ball et al., *HTGR Safety Studies for the Division of Accident Evaluation Quarterly Progress Report, April 1-June 30, 1984*, NUREG/CR-3885/V2 (ORNL/TM-9267/V2), Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., November 1984.
2. S. J. Ball et al., *HTGR Safety Studies for the Division of Accident Evaluation Quarterly Progress Report, July 1-September 30, 1984*, NUREG/CR-3885/V3 (ORNL/TM-9267/V3), Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., April 1985.
3. S. J. Ball et al., *HTGR Safety Studies for the Division of Accident Evaluation Quarterly Progress Report, October 1-December 31, 1984*, NUREG/CR-3885/V4 (ORNL/TM-9267/V4), Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., July 1985.
4. 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.

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