27
4-6-78
25 to NTIS

ORNL/NUREG/TM-195

High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research Quarterly Progress Report, October 1–December 31, 1977

S. J. Ball
J. C. Cleveland
J. C. Conklin
M. Hatta
J. P. Sanders

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

MASTER

OAK RIDGE NATIONAL LABORATORY

OPERATED BY UNION CARBIDE CORPORATION · FOR THE 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
Price: Printed Copy $4.00; Microfiche $3.00

ORNL/NUREG/TM-195
Dist. Category NRC-8

Contract No. W-7405-eng-26

HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF REACTOR SAFETY RESEARCH QUARTERLY PROGRESS REPORT, OCTOBER 1—DECEMBER 31, 1977

S. J. Ball, Manager
J. C. Cleveland M. Hatta
J. C. Conklin J. P. Sanders

Manuscript Completed — March 13, 1978
Date Published - March 1978

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

Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
DEPARTMENT OF ENERGY

CONTENTS

PRIOR HTGR SAFETY REPORTS

NRC-8 quarterly reports for period ending	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

NRC-8 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).

FOREWORD

HTGR safety studies at Oak Ridge National Laboratory (ORNL) are sponsored by 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 October 1 to December 31, 1977. Previous quarterly reports are listed on p. v, along with the topical reports published to date. Copies of the reports are available from the Technical Information Center, U.S. Department of Energy, Oak Ridge, Tenn. 37830.

HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF REACTOR SAFETY RESEARCH QUARTERLY PROGRESS REPORT, OCTOBER 1—DECEMBER 31, 1977

S. J. Ball, Manager
J. C. Cleveland M. Hatta
J. C. Conklin J. P. Sanders

ABSTRACT

Work continued on developing and refining the ORTAP-FSV code, a simulator of the dynamic behavior of the Fort St. Vrain reactor nuclear steam supply system.

Preliminary comparisons were made of data obtained from a Fort St. Vrain reactor scram and predictions using the ORECA code, and the agreement was good.

1. HTGR SYSTEMS AND SAFETY ANALYSIS

S. J. Ball

Work for the Division of Reactor Safety Research (RSR) under the HTGR Systems and Safety Analysis Program began in July 1974, and progress is reported quarterly. Work during this quarter consisted of further development of the ORTAP-FSV code, and calculations of a scram of the Fort St. Vrain (FSV) reactor from 28% power for comparison with test data.

1.1 Development of the FSV Nuclear Steam Supply System Simulation Code (ORTAP-FSV)

M. Hatta J. C. Conklin S. J. Ball
J. C. Cleveland R. M. Wright

Work on the FSV simulator code, ORTAP-FSV,[1] was continued. Updated versions of all the Fortran routines were collated and put on disk files for more ready access. Work on further documentation and annotation of the listings and sample transients is continuing. The purpose of this work is to make the code more readily exportable.

1.2 Development of the ORECA Code for Core Simulations of Emergency Cooling Transients

S. J. Ball

The ORECA code[2] was adapted for use on the ORNL PDP-10 time share computer. In addition to improving access and turnaround time, this feature will also allow use of ORECA at the FSV plant for on-site data analysis and experiment planning. An alternate feature was added to ORECA which allows for adjusting initial refueling region flows such that specified initial region outlet temperatures are attained (e.g., to match plant data).

In response to suggestions made by the NRC Division of Project Management, preliminary work was done on a plan to use FSV data to provide partial verification of the ORTAP and ORECA codes, at least for certain types of transients and limited ranges of plant operation. The ORECA code was used for preliminary scoping and sensitivity calculations to determine what types of test data could be used for code verification.

1.3 Preliminary Comparisons of ORECA Predictions with FSV Data from a 28% Power Scram Test

S. J. Ball

Preliminary calculations using the ORECA code were in generally good agreement with data supplied to us by General Atomic (GA) for the Aug. 6, 1977, FSV scram test from 28% power. Plant data supplied by GA included core power, inlet temperature, flow, and measured outlet temperatures from the 37 refueling regions, all as functions of time after the scram. GA also supplied estimates of initial region peaking factors and an equation for the region outlet thermocouple response time as a function of flow. The equation, which was derived from a model intended for use in the normal operating range, predicts a time constant of 1.9 min at full flow.

A particularly interesting feature of the test was that there were two 2-min periods of no flow (at 10 and 14 min after the scram). These caused significant delays in the cooldown transients.

Figure 1 shows that the ORECA prediction for the measured outlet temperature of refueling region No. 1 (the center region), which had a calculated peaking factor (P.F.) of 1.335, is in excellent agreement for the first 30 min and low thereafter. The response is slower and the overall agreement better for a lower P.F./flow region in the second ring (region No. 15, P.F. = 0.788), shown in Fig. 2.

A detailed study of the comparisons is under way, along with calculations of the sensitivity of the predicted responses to various parameter and model assumptions.

1.4 A Simplified Steam Property Routine for Use in the ORTAP-FSV Code

M. Hatta

A draft report on the SIMplified tabulation of steam PROperties (SIMPRO) was completed. SIMPRO is a routine developed at IHI (Japan) for use in their HTGR PLANt DYnamics (PLANDY) NSSS code. SIMPRO uses the JSME steam tables (1968) as a basis for a linear interpolation table lookup routine. The tabulation also contains heat transfer coefficients and pressure drop

ORNL DWG 78 4645

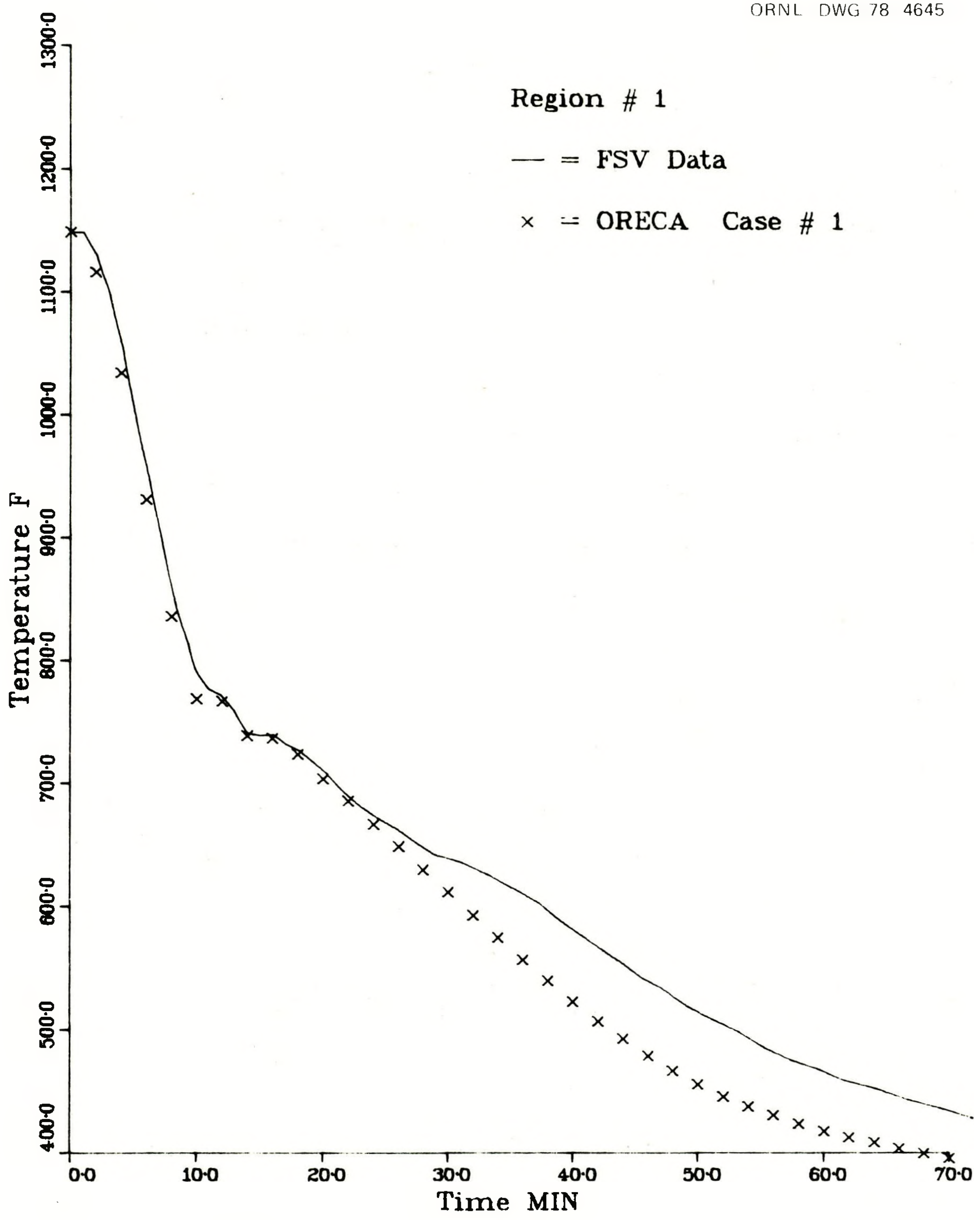

Fig. 1. FSV scram test of August 6, 1977 — comparison of reference case ORECA code predictions of measured gas outlet temperature from region 1 vs plant data.

ORNL-DWG 78 4646

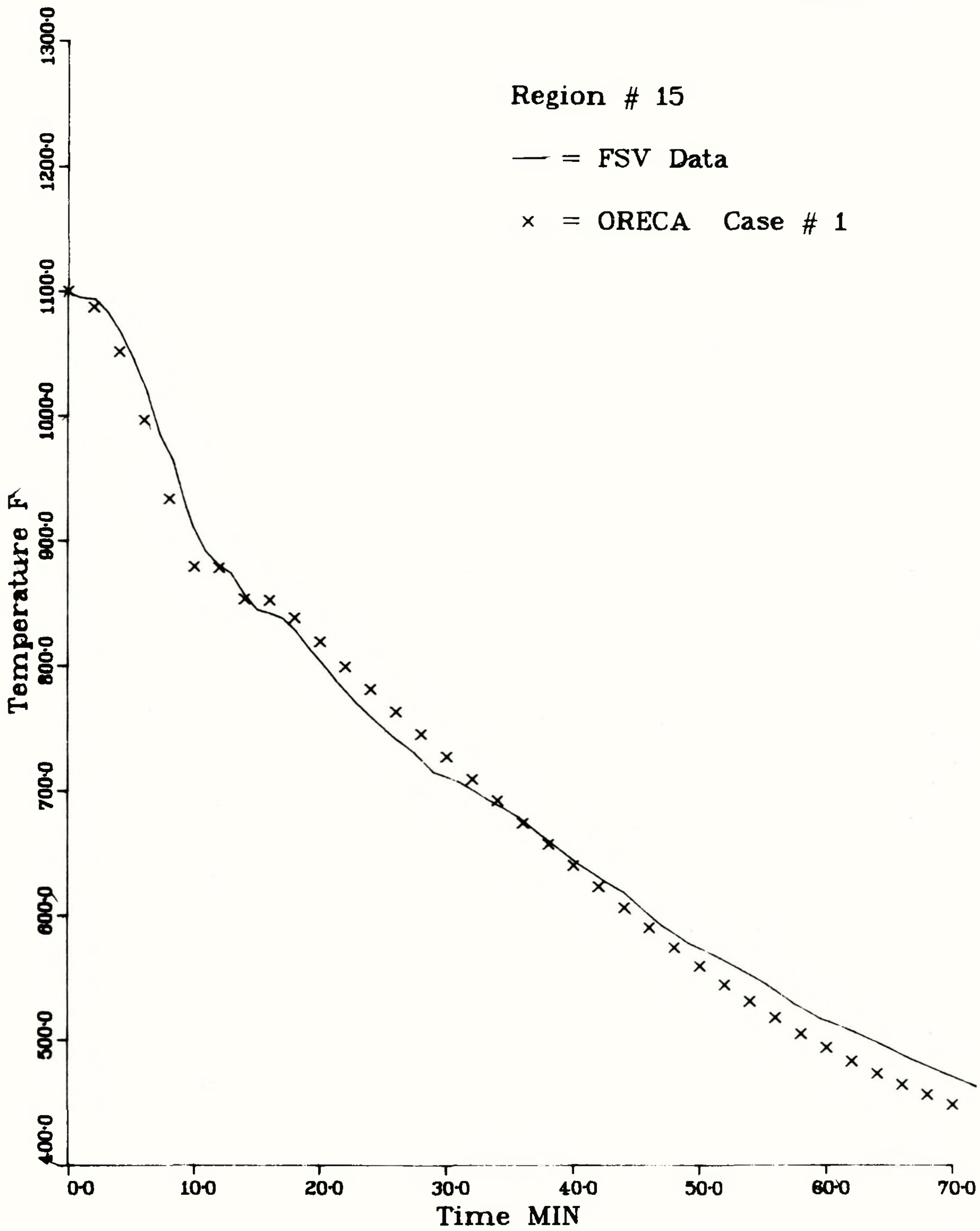

Fig. 2. FSV scram test of August 6, 1977 — comparison of reference case ORECA code predictions of measured gas outlet temperature from region 15 vs plant data.

coefficients. Heat transfer and pressure drop coefficients are broken up into three factors: a geometry factor, a flow factor, and a steam/water properties factor. Thermodynamic properties and state variables are determined by specifying any two state variables: pressure, enthalpy, entropy, temperature, and specific volume. The table also includes saturated steam and water and the two-phase range, so quality and two-phase flow and heat transfer coefficients can be determined. Thermodynamic properties included are viscosity, conductivity, and specific heat.

Plans are to try out SIMPRO in ORTAP to determine the effect on run time efficiency and accuracy.

1.5 Time Mesh Survey for the BLAST Code

M. Hatta

In order to improve the run time efficiency of ORTAP, a study was made to determine the limiting factors for increasing the time mesh (or computation interval). It was assumed that the BLAST code[3] (steam-generator and reheater subroutine) required the shortest time mesh and that the time step Δt had to be chosen to be less than the transport time of the fluid through a node. The conclusions of the study indicated that the BLAST time mesh could be increased from 0.1 sec (typical ORTAP Δt) to 0.4 sec with no serious degradation of stability or accuracy.

Presently the heat transfer and pressure drop coefficients for each water node are computed on the basis of the flow regime as determined by average nodal properties. For a large node, however, a substantial length of it may be in a flow regime other than that determined for that node, and oscillations can occur between regimes for successive time steps.

To prevent this, it is recommended that a weighted averaging scheme be used to determine coefficients that account for the fraction of the node that is in each regime.

It was also concluded that the limiting factor for stability was the steam holdup volume calculations.

1.6 A Simplified Steam Turbine Plant Model

M. Hatta

A draft report was written describing the IHI PLANDY code model of a simplified steam turbine plant. The components described are the high-, intermediate-, and low-pressure turbines, and the turbine-drive feedwater pump. This routine is to be used as an alternative for the detailed turbine-generator plant model now in ORTAP, at least for some types of transients that do not require such detail.

1.7 University of Tennessee Subcontract

T. W. Kerlin

The subcontract with the University of Tennessee for conducting research on gas-cooled reactor modeling and model evaluation expired in August 1977. Three Ph.D. theses will result from the subcontract and will be published shortly. These are I. Anastasiou, "A Comparative Study of 3 Once-Through Steam Generator Models for the FSV HTGR Plant," M. H. Lee, "A Detailed Nonlinear Dynamic Model of a Once-Through Steam Generator," and J. G. Thakkar, "Development of Identification Techniques and Nonlinear Dynamic Models for HTGR Plant Components."

2. COMPUTER CODE EVALUATIONS

S. J. Ball   J. C. Cleveland   J. P. Sanders

The draft of the report entitled Evaluation of the General Atomic Codes TAP and RECA for HTGR Accident Analyses, ORNL/NUREG/TM-178, was reviewed and revised for publication.[4–6]

REFERENCES

1. 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).
2. 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).
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 (August 1976).
4. A. Bardia and R. C. Potter, TAP, A Program for Analysis of HTGR Nuclear Steam System Performance Transients, GA-A13248 (GA-LTR-21) (Jan. 30, 1976).
5. H. W. Chi and G. J. Malek, Description of the Reactor Emergency Cooling Analysis Code, RECA, GA-10273 (Aug. 19, 1970).
6. J. F. Peterson, RECA3: A Computer Code for Thermal Analysis of HTGR Emergency Cooling Transients, GA-A14520 (GA-LTR-22) (August 1977).

ORNL/NUREG/TM-195
Dist. Category NRC-8

INTERNAL DISTRIBUTION

1-10. S. J. Ball
11. N. E. Clapp, Jr.
12. J. C. Cleveland
13. J. C. Conklin
14. M. Hatta
15. F. J. Homan
16-17. P. R. Kasten
18. T. W. Kerlin
19. Milton Levenson
20. A. L. Lotts
21. F. R. Mynatt
22-23. J. P. Sanders
24-25. Myrtleen Sheldon
26. R. S. Stone
27. H. E. Trammell
28. D. B. Trauger
29. ORNL Patent Office
30-31. Central Research Library
32. Document Reference Section
33-35. Laboratory Records Department
36. Laboratory Records, ORNL R.C.

EXTERNAL DISTRIBUTION

37-40. Director, Office of Nuclear Regulatory Research, USNRC, Washington, DC 20555
41. Chief, Experimental Gas-Cooled Reactor Safety Research Branch, Division of Reactor Safety Research, Office of Nuclear Regulatory Research, USNRC, Washington, DC 20555
42. Director, Reactor Division, DOE, ORO
43. Director, Research and Technical Support Division, DOE, ORO
44-271. Given distribution as shown in category NRC-8 (25 copies — NTIS)