/16/78
5 to NTIS

NUREG/CR-0256
ORNL/NUREG/TM-233

MASTE

High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research Quarterly Progress Report, April 1–June 30, 1978

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

OAK RIDGE NATIONAL LABORATORY
OPERATED BY UNION CARBIDE CORPORATION · FOR THE DEPARTMENT OF ENERGY

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

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.

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

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor any of its employees, nor any of its 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.

NUREG/CR-0256
ORNL/NUREG/TM-233
Dist. Category R8

Contract No. W-7405-eng-26

HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF REACTOR SAFETY RESEARCH QUARTERLY PROGRESS REPORT, APRIL 1—JUNE 30, 1978

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

Manuscript Completed — September 21, 1978

Date Published - October 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
NRC FIN No. B-0122-7

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

NOTICE
This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Department of Energy, 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.

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

jy

CONTENTS

PRIOR HTGR SAFETY REPORTS v
FOREWORD vii
ABSTRACT 1
1. HTGR SYSTEMS AND SAFETY ANALYSIS 1
  1.1 Development of the FSV Nuclear Steam Supply System Simulation Code (ORTAP-FSV) 1
  1.2 Assistance with the NRC Review of the FSV 100% Power License Application 3
  1.3 Comparisons of ORECA Code Predictions with FSV Scram Test Data 5
  1.4 Investigations of the FSV Temperature and Power Oscillation Problem 10
  1.5 Preliminary Heated Plume Experiments 11
2. MEETINGS ATTENDED UNDER PROGRAM SPONSORSHIP 15
  2.1 NRC Meeting to Discuss ORNL Assistance on FSV 100% Power License Review, Bethesda, Md., Apr. 18, 1978 15
  2.2 NRC Meetings to Review FSV Licensing Questions, Bethesda, Md., Apr. 19–20, 1978 15
  2.3 NRC Gas-Cooled Reactor Safety Research Review Group Meeting, Silver Spring, Md., May 15, 1978 16
  2.4 NRC Meeting to Discuss Current FSV Oscillation Data and Analyses, Bethesda, Md., May 16, 1978 16
  2.5 NRC Meeting to Describe ORNL Licensing Calculations for FSV, Bethesda, Md., May 26, 1978 16
  2.6 Visit to General Atomic Co., San Diego, to Discuss FSV Oscillation Problems, June 22–23, 1978 17
REFERENCES 18

PRIOR HTGR SAFETY 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

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 April 1 to June 30, 1978. Previous quarterly reports and topical reports published to date are listed on p. v. 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, APRIL 1—JUNE 30, 1978

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

ABSTRACT

Assistance was provided to NRC in evaluating and confirming ECCS analyses for the Fort St. Vrain (FSV) reactor in support of a 100% power license application. Comparisons of ORNL accident analyses with those of the applicant were generally in good agreement. Other work included further development of the ORTAP-FSV and ORECA codes and comparisons of data from several FSV scram tests with ORECA code predictions.

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 the present quarter included analyses in support of the Fort St. Vrain (FSV) reactor request for a 100% power operating license, further development of the ORTAP-FSV code, comparisons of ORECA code predictions with data from several FSV reactor scram tests, and further investigations of the FSV temperature and power oscillation problem.

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

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

Further development work on the ORTAP-FSV code[1] was continued. The detailed model and code for the steam lines in the turbine plant was completed. The code uses a variation of the MATEXP[2] method, call MATEX2, as the integration routine. The first draft of a report[3] on MATEX2 was completed.

A new computer simulation model of the intermediate- and low-pressure turbine has been written and coupled with the present ORTAP feedwater heater model. Several totally different mathematical expressions from those previously used in ORTAP have been implemented in order to decrease computation time.

The intermediate-pressure turbine inlet flow and extraction flows are calculated by equations derived from combining the ideal gas law, mass continuity, and Bernoulli's equation. The pressure at each extraction point is calculated by an empirically determined equation of the form

$$\frac{P_j}{P_{j,0}} = \left(\frac{W_j}{W_{j,0}}\right)^n ,$$

where

P_j = pressure at extraction point j,
$P_{j,0}$ = pressure at extraction point j for initial conditions,
W_j = turbine mass flow downstream of extraction point j,
$W_{j,0}$ = turbine mass flow downstream of extraction point j at initial conditions,
n = empirically determined exponent.

The exponent n was determined from the turbine heat balance data furnished by Public Service of Colorado (PSCo) and is equal to 0.994 for all turbine extraction points.

A power runback from 100 to 25% was modeled, and the results at 25% compared favorably with those of a turbine heat balance supplied by PSCo.

An abstract of a paper for presentation at the HTGR Safety Seminar in Tokai, Japan, was written and subsequently accepted. The paper is entitled "Investigations of Postulated Accident Sequences for the Fort St. Vrain HTGR," by S. J. Ball, J. C. Cleveland, J. C. Conklin, M. Hatta, and J. P. Sanders. The abstract is as follows:

> The present systems analysis capability of the ORNL HTGR Safety analysis research program consists of a family of computer codes, including an overall plant NSSS simulation (ORTAP)

and detailed component codes for investigating core neutronic accidents (CORTAP), shutdown emergency-cooling accidents via a 3-dimensional core model (ORECA), and once-through steam generator transients (BLAST). The component codes can either be run independently or in the overall NSSS code.

Several postulated accident sequences have been, or are being analyzed, including: rod-pair-withdrawal accidents, design-basis depressurization accidents, loss of forced-convection cooling accidents, and slow depressurization accidents. Sensitivity studies are run in conjunction with each accident to determine the importance of both model and parameter uncertainties.

Code verification efforts to date have consisted of using existing Fort St. Vrain reactor dynamics data to compare with predictions. Comparisons made for a reactor scram from 28% power showed good agreement using ORECA. An optimization program was used to rationalize the differences between the predicted and measured refueling region outlet temperatures, and excellent agreement was attained by adjustment of parameters within their uncertainty ranges.

Copies of the BLAST,[4] ORECA,[5] and CORTAP[6] codes were sent to RWTUV, West Germany, at the request of the Director of the Office of Nuclear Regulatory Research. The codes are to be used in an independent safety assessment of the THTR pebble bed reactor.

1.2 Assistance with the NRC Review of the FSV 100% Power License Application

S. J. Ball J. C. Conklin
M. Hatta J. P. Sanders R. M. Wright

A letter report[7] was submitted to NRC in response to a request for evaluation and confirmation of ECCS analyses for FSV, specifically to five items[8] (see Sect. 2.1). Items 1 and 2 related to audit calculations (to confirm GA analyses) for postulated loss of forced convection (LOFC) accidents followed by firewater cooldown (FWCD) and for design-basis depressurization accidents (DBDAs). The analyses were based primarily on calculations using the ORECA code[5] and an input data package supplied by GA.[9]

For the LOFC/FWCD analyses, a model of each of the upper core plenum cover plate regions above the 37 refueling regions was added as an "optional" ORECA subroutine package. The model includes T^4 radiation heat transfer between individual cover plate regions and the upper surfaces of the refueling regions, a plume heat transfer model for regions experiencing reverse flow, convection heat transfer from the other cover plates to the mixed-mean upper plenum gas temperature, and conduction from the cover plates through the Kaowool insulation to the liner cooling system (which was assumed to remain in operation). Conservative features of the model are the omission of radiant heat transfer from the cover plates to the side reflector and side walls, omission of cover plate heat capacity, omission of any overall plenum convection flows, and neglect of the effect of the heat loss to the liner cooling system on the mixed-mean upper plenum gas temperature. [This may amount to ~17°C (30°F) cooling.] Limitations of the model which may or may not be conservative (and hence require further investigation) are the neglect of plume convection and radiant heat transfer to the control rod guide tubes, the derivation of an adequate model for the plume heat transfer coefficient (h plume), and consideration of an effective augmentation of h plume when a number of reverse-flow regions are clustered together.

Calculations of core conditions for the first 2 hr of an LOFC accident were done for both the worst-case equilibrium core and the worst-case initial core. In both cases, the maximum predicted fuel temperatures were below the 1600°C (2912°F) long-term FSAR safety limit temperature, below which no fuel failure is expected. The major concern in this period is the ability of the carbon steel upper-plenum thermal barrier cover plates to withstand the heat from the reverse-flow plumes. The calculations indicated that some of the cover plate regions would exceed the 816°C (1500°F) damage limit during the 2-hr LOFC for the equilibrium (but not the initial) core.

The main concern after resumption of primary coolant flow following an extended LOFC is possible damage to the steel liners at the steam generator inlets due to hot streaking from the hottest refueling regions. In all cases, for FWCD starting times up through 2 hr after an LOFC, the predicted maximum liner and ducting temperatures were below the 1093°C (2000°F) damage limit.

Long-term calculations for the initial core LOFC with FWCD introduced after 2 hr indicated that there would be a recurrence of reverse flows after ~5 to 6 hr; however, the core has cooled to such an extent that the hot plumes would not cause any damage to the upper plenum cover plates.

Sensitivity studies were done for the equilibrium core LOFC/FWCD accident in order to note the effects some model and parameter variations would have on the peak temperature predictions. Parameters varied included FWCD core flow, afterheat, initial power, coolant heat transfer and friction factors, and computation time interval. Results of the sensitivity studies were all "reasonable" (i.e. there were no surprises).

The DBDA analyses were done only for the worst-case equilibrium core and assumed a 5-min delay in the startup of the emergency cooling system followed by the GA-supplied ECCS flow history. As with the LOFC, the predicted peak fuel temperatures and steam generator inlet duct temperatures were below their respective damage limits. Furthermore, sensitivity studies again indicated no surprises. A plot of several parameters of interest in the reference case DBDA is shown as an example in Fig. 1.

Item 3 of the letter of request concerned a review of the RECA code,[10] which the applicant used for FSV licensing calculations. ORNL agreed to respond to any questions the NRC staff had about RECA3 or the ORNL review report of the RECA and TAP codes.[11]

Item 4 of the request was for ORNL on-call assistance to review FSV licensing information. In the discussion of alternate accident scenarios (item 5), several points were raised which should be considered, although it was felt that their resolution should not necessarily be a prerequisite to 100% power operation.

1.3 Comparisons of ORECA Code Predictions with FSV Scram Test Data

S. J. Ball

The previous quarterly report[12] described the techniques used to compare FSV scram test data with ORECA code predictions and showed results of an optimized fit for the 28% power scram of Aug. 6, 1977. Subsequently, ORNL-DWG 78-17219

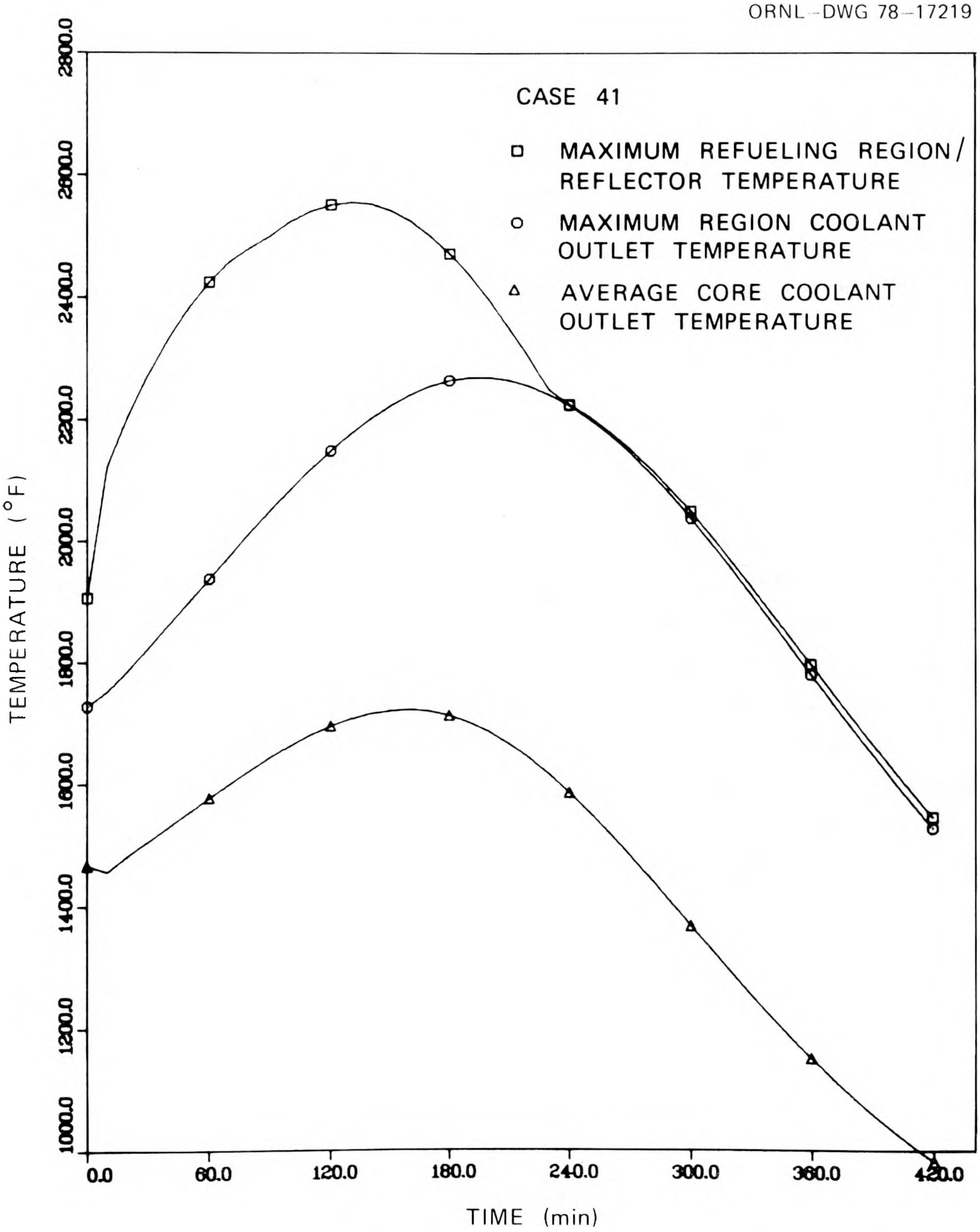

Fig. 1. Sample results of a postulated Fort St. Vrain reactor DBDA using the ORECA code.

data were obtained from GA on scrams from both 40 and 50% power, and analysis of the former case was completed within the quarter. Results of the 50% power scram analysis will be reported later.

The 40% power scram occurred on Oct. 25, 1977. Unlike the 28% power scram, which had two 2-min no-flow periods following the scram, the shutdown primary coolant flow was relatively steady. In both cases, the "reference case" (i.e. unadjusted parameters) ORECA calculations of refueling region outlet temperatures were in generally good agreement for the first 20 to 30 min of the transient, and then the predictions fell below the measurements. This behavior was typical of the majority of the refueling regions. After application of the RANOPT code, which adjusts selected parameters in an effort to force an agreement, an excellent fit again resulted for the outlet temperatures for all refueling regions. Once again, the parameter adjustments made were within reasonable uncertainty limits, although the optimized set of parameters differed from those obtained for the 28% power scram case. It could be concluded that more data would have to be analyzed and perhaps more model variations analyzed before a universal set of optimum coefficients were derived.

Figures 2 and 3 show examples of the optimized results for the 40% scram case, and Tables 1 and 2 indicate the parameter adjustments required.

Table 1. Parameter adjustments required for optimum data fits, FSV 28 and 40% scram tests

	~28% scram test of 8/6/77	~40% scram test of 10/25/77
Estimated initial thermal power, %	28	40
Optimized power, %	29.28	45.21
Measured initial primary flow, lb/sec	400	536
Optimized initial core flow, lb/sec	379.7	530
Optimized primary flow through core after loss of one loop, %	81	85.9
TGO thermocouple optimized T^4 fractio	0.11	0.092
Optimized temperature increase from measured circulator inlet to cavity outlet, °F	17	42
Optimum peaking factor adjustments	See Table 2 of Ref. 11	See Table 2

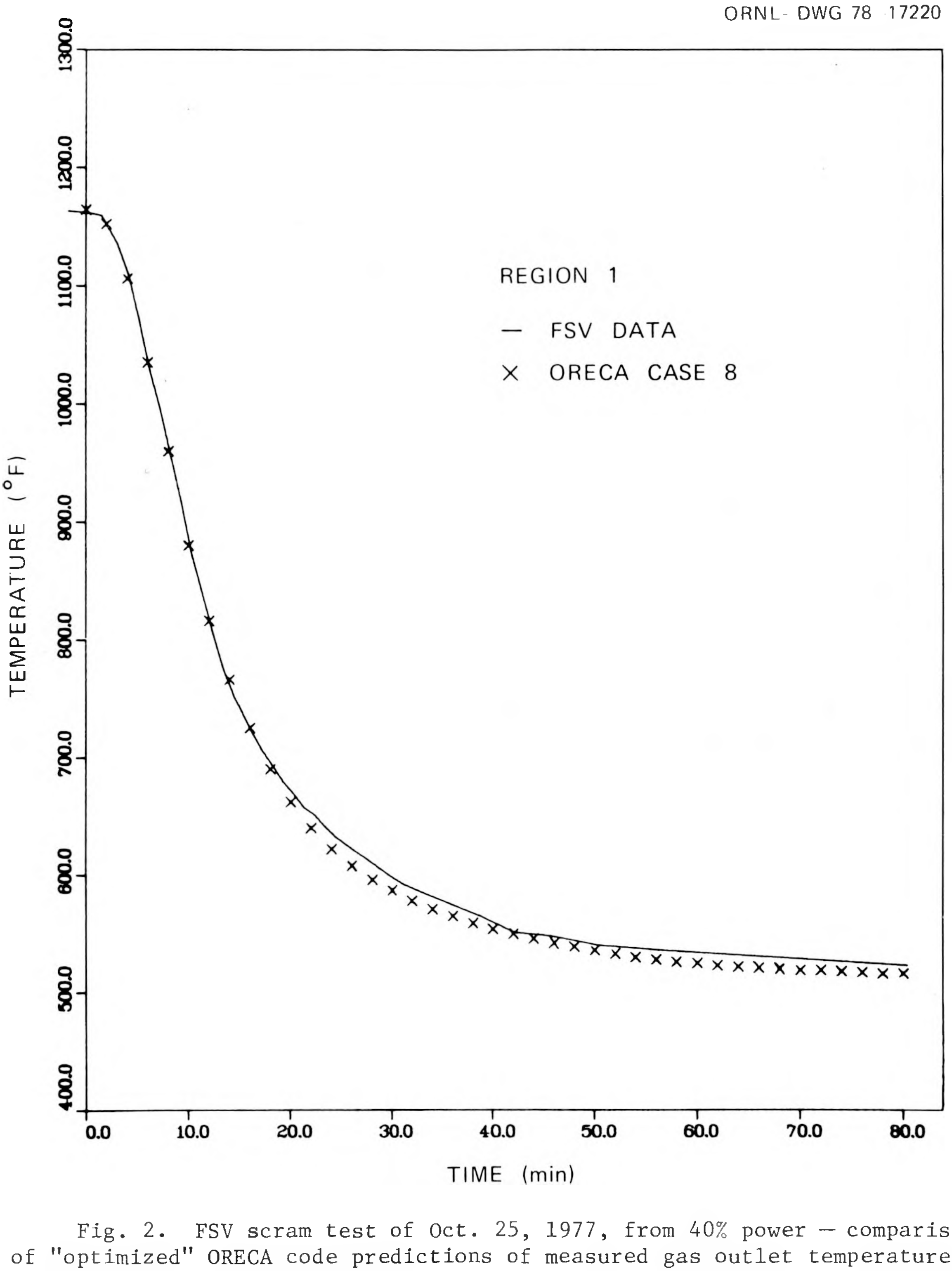

Fig. 2. FSV scram test of Oct. 25, 1977, from 40% power — comparison of "optimized" ORECA code predictions of measured gas outlet temperature from region 1 vs plant data.

ORNL-DWG 78-17221

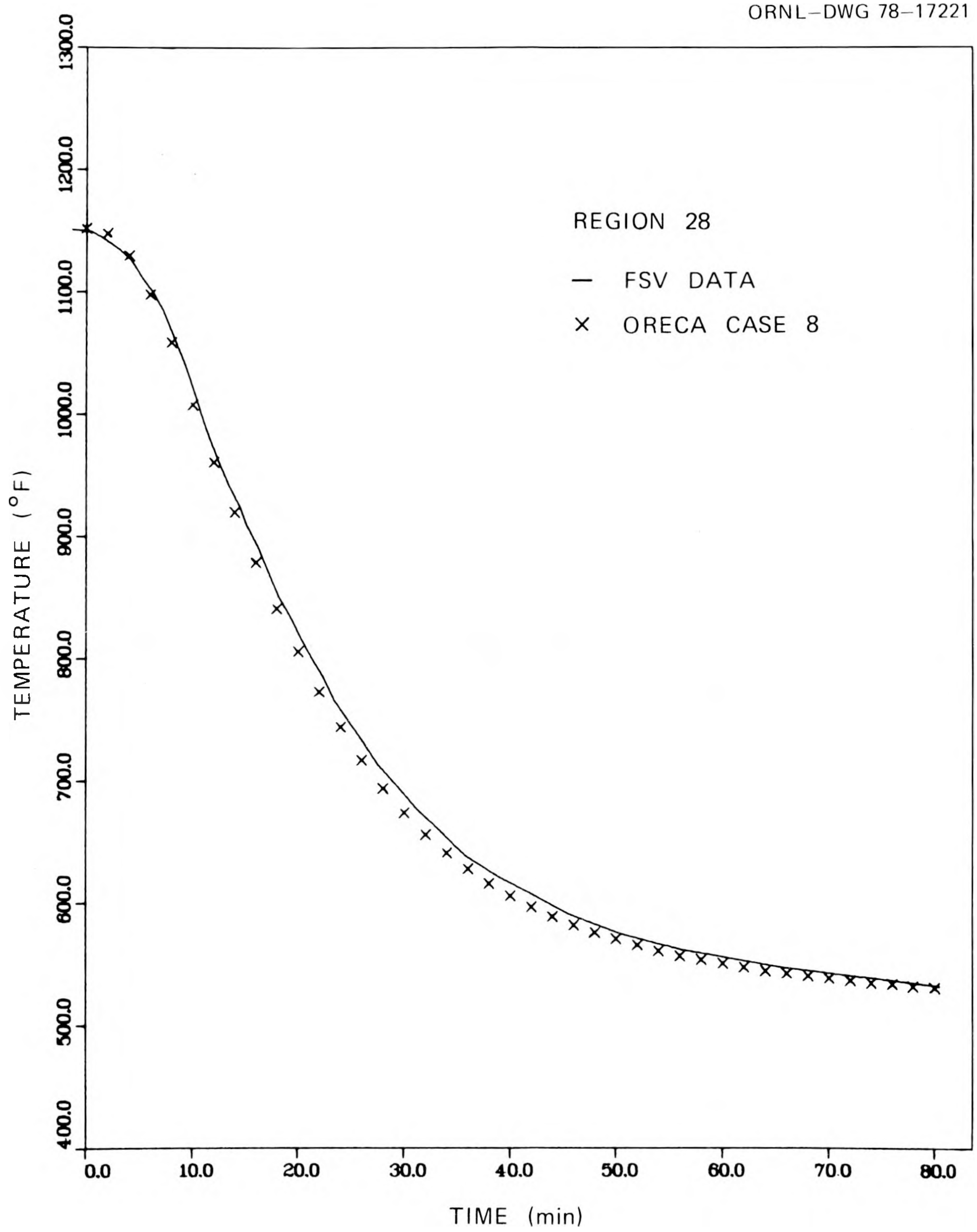

Fig. 3. FSV scram test of Oct. 25, 1977, from 40% power — comparison of "optimized" ORECA code predictions of measured gas outlet temperature from region 28 vs plant data.

Table 2. ORECA "optimized case" peaking factors — case 8 (ORECA FSV 40% scram fit)

Refueling region	Peaking factor	
	Original	New
1	1.090	Same
2	1.493	Same
3	1.325	Same
4	1.085	Same
5	1.088	1.108
6	1.464	1.484
7	1.460	Same
8	0.973	1.012
9	1.381	Same
10	0.913	Same
11	0.933	Same
12	1.245	1.206
13	1.203	Same
14	1.259	1.220
15	0.792	0.831
16	1.320	1.34
17	1.097	1.117
18	1.027	1.047
19	0.993	Same
20	0.343	0.363
21	0.688	Same
22	0.624	0.644
23	0.617	0.676
24	1.092	1.112
25	0.801	0.821
26	0.511	0.706
27	1.166	1.205
28	0.637	0.657
29	0.696	0.774
30	1.257	1.238
31	1.162	Same
32	0.372	Same
33	0.848	0.731
34	0.523	0.426
35	0.350	0.311
36	0.659	0.503
37	0.802	0.646

1.4 Investigations of the FSV Temperature and Power Oscillation Problem

S. J. Ball M. Hatta

Project personnel have attended several NRC and GA meetings on the FSV oscillation problem (see Sect. 2) and have acted in an advisory capacity to NRC on licensing-related questions. A considerable number of related reports and data packages have also been received and reviewed in detail. In general, it is believed that the GA-proposed explanation is sound; that is, the neutron and temperature fluctuation signals are due to refueling region and reflector block motion. The large neutron signal changes are due to variable-gap streaming, and the large temperature signal fluctuations are due to variable-gap bypass flows.

1.5 Preliminary Heated Plume Experiments

M. Hatta   S. J. Ball

A major uncertainty in the prediction of the consequences of sustained loss-of-forced-convection (LOFC) accidents in HTGRs is the effective heat transfer from the heated (upflow) plumes from the core refueling regions to the thermal barrier cover plates lining the top of the upper plenum (see Sect. 1.2). The reverse core coolant flow phenomenon occurs because of the buoyancy of hot gas in a refueling region and is typically significant only when the reactor is at or near full pressure (~700 psia). Reverse flows normally occur in the higher peaking factor regions. The problem is especially significant in the FSV upper plenum, which has carbon steel cover plates having a maximum temperature limit of 1500°F. Simulations of 2-hr LOFC accidents have indicated that this temperature limit might be exceeded, depending in large part on the assumptions of plume heat transfer.

To date, a search of the literature and consultations with experts in the field have indicated that there are no experimental data available that would be directly applicable to the HTGR LOFC case. Consequently, two approaches are being considered: (1) to conduct special reverse-flow tests on FSV and (2) to develop a low-temperature air model experiment which could simulate the high-temperature high-pressure helium. Plans for possible FSV tests are only in the preliminary planning stage.

A preliminary "scoping" experiment was set up and run to investigate the feasibility of a large-scale upper plenum air model plume experiment. A schematic diagram of the scoping experiment is shown in Fig. 4. Air flow to an industrial heat gun (hair dryer) is metered by a rotometer, and the plume temperature is controlled by varying the hair dryer power with a variac. Temperatures are monitored by thermocouples which are read out by a computerized data acquisition system (DAS). The DAS has a low-level scanner with an integrating digital voltmeter which is capable of high-resolution temperature measurements. The upflow plume impinges on a water-filled vessel, the bottom of which is a large, thin aluminum plate. The heat transfer coefficient from the plume to the plate is

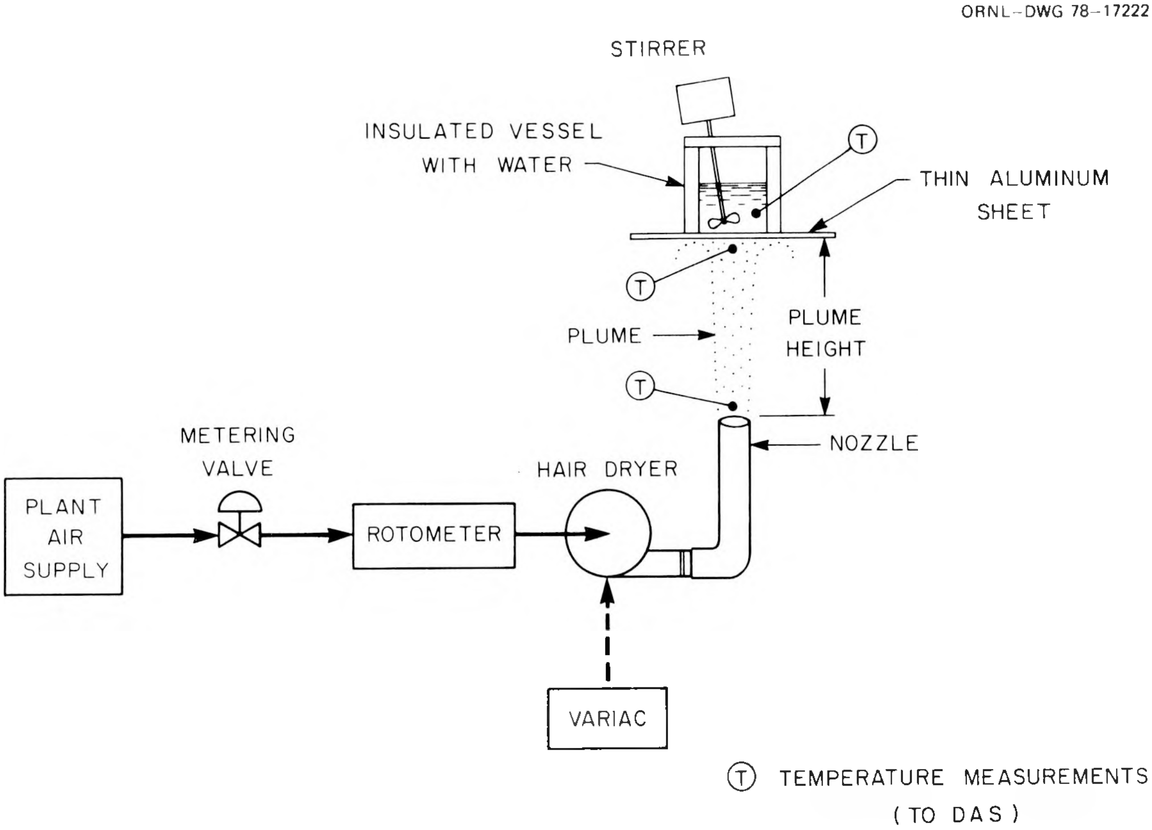
ORNL-DWG 78-17222
STIRRER
INSULATED VESSEL WITH WATER
T
THIN ALUMINUM SHEET
T
PLUME
PLUME HEIGHT
T
NOZZLE
METERING VALVE
HAIR DRYER
PLANT AIR SUPPLY
ROTOMETER
VARIAC
T TEMPERATURE MEASUREMENTS (TO DAS)

Fig. 4. Block diagram of heated plume scoping experiment.

inferred by measuring the heatup rate of the (known mass of) water in the vessel. The on-line computer program for monitoring and analyzing the course of the experiment was written in an augmented version of the FOCAL language.

In the initial runs made during the quarter, data showing the Nusselt number vs Reynolds number relationship were obtained for several conditions (Fig. 5). Eventually, data over a wide range of conditions will be taken in an attempt to derive the coefficients and functional relationships for an expression of the form:[13]

$$Nu = \frac{A\ Re^n Gr^m}{\exp(f(H/D))} ,$$

ORNL DWG 78 17223

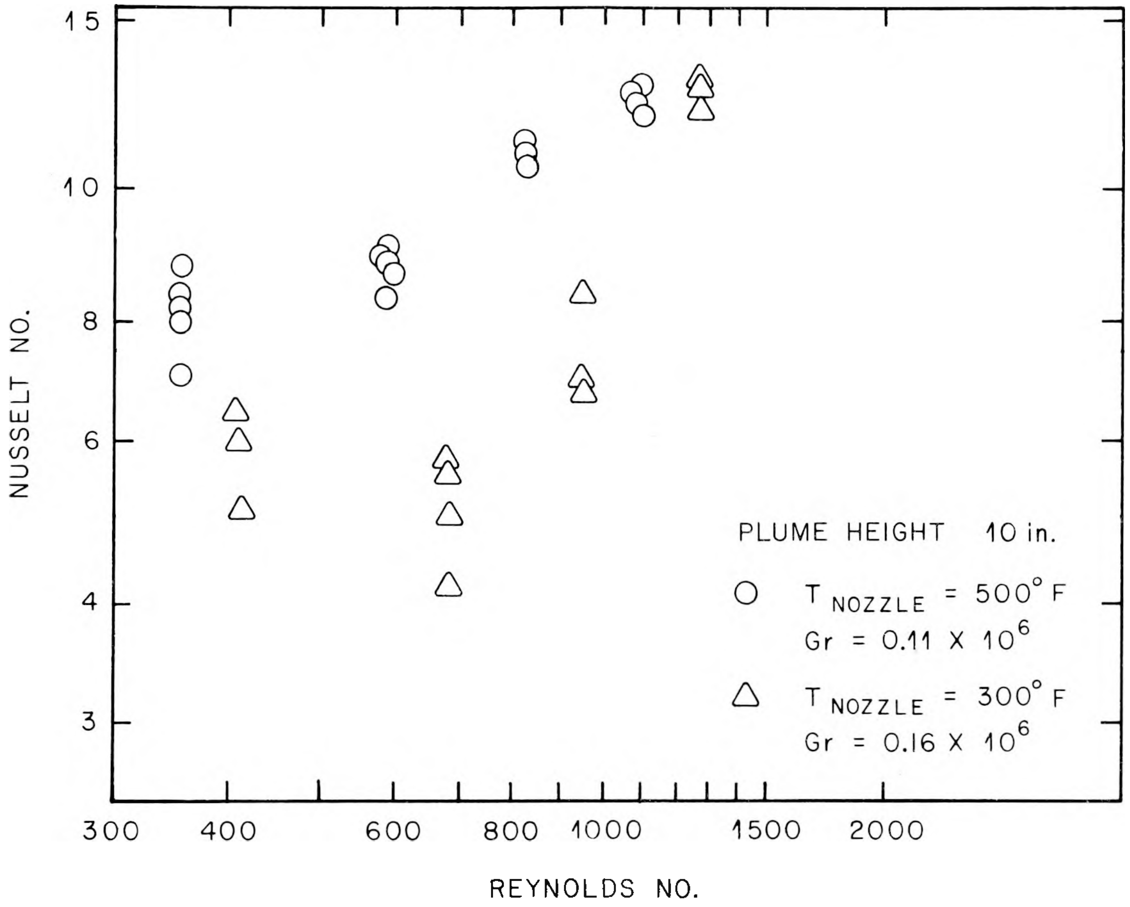

Fig. 5. Initial Nusselt number vs Reynolds number data from the heated plume scoping experiment.

where

Nu = Nusselt number,

Re = Reynolds number,

Gr = Grashof number,

H/D = ratio of plume height to effective nozzle discharge diameter,

A, n, m = coefficients to be determined.

Assuming Reynolds and Grashof scaling is appropriate for modeling the HTGR plumes, it appears feasible to simulate the high-pressure helium with a full-scale air model (Table 3).

Table 3. Comparison of HTGR plume and air model parameters

	HTGR helium plume	Model air plume
Temperature, °F	2000	200
Pressure, psia	700	14.7
Density, lb_m/ft^3	0.11	0.06
Viscosity, lb_m/ft hr	0.14	0.053
Mass flow, lb_m/min	15	5.9[a]
Equivalent orifice diameter, in.	17	17
Velocity, fps	1.5	1.04
Reynolds No.	6000	6000
Grashof No.	7×10^8	4.3×10^8
$Gr^{\frac{1}{4}}$	163	144

[a] 78 scfm, ~3.5 kW heater.

2. MEETINGS ATTENDED UNDER PROGRAM SPONSORSHIP

2.1 NRC Meeting to Discuss ORNL Assistance on FSV 100% Power License Review, Bethesda, Md., Apr. 18, 1978

S. J. Ball

A meeting was held with Office of Nuclear Reactor Regulation (ONRR) and RSR representatives to define ORNL's role in assisting with the FSV 100% power licensing questions. It was agreed that ONRR's request to RSR would include five items for "immediate" action:

1) provide audit calculations of the firewater cooldown and DBDA accidents using ORECA;
2) provide detailed calculations and parametric studies of the firewater cooldown accident with estimates of critical component thermal histories (e.g., upper thermal barrier cover plates) using ORECA and perhaps other codes;
3) respond to NRC questions about the ORNL review[11] of RECA;
4) provide continuing on-call assistance in support of licensing questions;
5) inform NRC of our judgment of GA's claim that the three analyses they did in support of the 100% power license application (i.e., cooldown on one firewater-driven Pelton wheel, rapid depressurization, and permanent loss of forced circulation) provide bounding consequences for other accidents identified within the FSAR.

2.2 NRC Meetings to Review FSV Licensing Questions, Bethesda, Md., Apr. 19–20, 1978

S. J. Ball M. Hatta

Several issues pertaining to FSV power ascension above 70%, Amendment 18 of the Safety Evaluation Report, and the repair of the steam generator tube leak were discussed. A detailed description of the FSV oscillation problem was also presented by GA. GA also outlined their plans for installing additional diagnostic instrumentation and for future analysis and tests.

2.3 NRC Gas-Cooled Reactor Safety Research Review Group Meeting, Silver Spring, Md., May 15, 1978

S. J. Ball M. Hatta

BNL presented results of their analytical and experimental research on gas mixing in the containment vessel following a postulated DBDA. S. J. Ball presented results of ORNL work on HTGR upper plenum plume modeling for postulated loss of forced convection flow accidents. Possible air model plume tests and FSV experiments were also discussed.

2.4 NRC Meeting to Discuss Current FSV Oscillation Data and Analyses, Bethesda, Md., May 16, 1978

S. J. Ball M. Hatta

Data from oscillation events in April and early May were presented. At that time it was clearly established that the tendency to have sustained oscillations was related to the core flow resistance and that the power and temperature oscillations were caused by refueling region and reflector block motion.

2.5 NRC Meeting to Describe ORNL Licensing Calculations for FSV, Bethesda, Md., May 26, 1978

S. J. Ball

Analyses as described in some detail in Section 1.2 of this report were presented and discussed. Several items requiring followup analyses were also discussed.

2.6 Visit to General Atomic Co., San Diego, to Discuss FSV Oscillation Problems, Jun. 22–23, 1978

S. J. Ball

The purpose of the meeting was to review progress and plans for understanding and solving the FSV reactor oscillation problems. The GA data reduction and analysis procedures were reviewed in some detail; GA handed out a large data package containing both raw data and internal memoranda on pertinent analyses. Discussions were held with NRC regarding the possibility of ORNL involvement in the collection and analysis of FSV oscillation data.

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 and R. K. Adams, MATEXP, A General Purpose Digital Computer Program for Solving Ordinary Differential Equations by the Matrix Exponential Method, ORNL/TM-1933 (August 1967).
3. M. Hatta and S. J. Ball, MATEX2 - A Variation on the Multipurpose Integration Routine MATEXP for Solving Large Sets of Nonlinear Differential Equations, ORNL/TM-6540 (in preparation).
4. 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).
5. 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).
6. 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).
7. Letter from S. J. Ball to R. D. Schamberger, Evaluation and Confirmation of ECCS Analyses for the Fort St. Vrain Reactor: "Immediate Tasks," May 25, 1978.
8. NRC Letter from E. G. Case to S. Levine, "Assistance in Evaluating and Confirming ECCS Analyses for Fort St. Vrain," May 1, 1978.
9. Letter from G. C. Bramblett (GA) to S. J. Ball, May 1, 1978.
10. J. F. Peterson, RECA3: A Computer Code for Thermal Analysis of HTGR Emergency Cooling Transients, GA-A14520 (GA-LTR-22) (August 1977).
11. S. J. Ball, J. C. Cleveland, and J. P. Sanders, Evaluation of the General Atomic Codes TAP and RECA for HTGR Accident Analyses, ORNL/NUREG/TM-178 (May 1978).
12. S. J. Ball et al., High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research, Quarterly Progress Report, Jan. 1—Mar. 31, 1978, ORNL/NUREG/TM-221 (July 1978).

13. K. P. Perry, "Heat Transfer by Convection from a Hot Gas Jet to a Plane Surface," *Proc. Inst. Mech. Eng.* 168(30), 775–84 (1954).

NUREG/CR-0256
ORNL/NUREG/TM-233
Dist. Category R8

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. R. P. Wichner
30. ORNL Patent Office
31-32. Central Research Library
33. Document Reference Section
34-36. Laboratory Records Department
37. Laboratory Records, ORNL R.C.

EXTERNAL DISTRIBUTION

38-41. Director, Office of Nuclear Regulatory Research, USNRC, Washington, DC 20555
42. Chief, Experimental Gas-Cooled Reactor Safety Research Branch, Division of Reactor Safety Research, Office of Nuclear Regulatory Research, USNRC, Washington, DC 20555
43. Director, Reactor Division, DOE, ORO
44. Director, Research and Technical Support Division, DOE, ORO
45-299. Given distribution as shown in category R8 (25 copies — NTIS)