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High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research Quarterly Progress Report, October 1–December 31, 1978

S. J. Ball, Manager
J. C. Cleveland
J. C. Conklin
D. G. Lister

Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
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HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF REACTOR SAFETY RESEARCH QUARTERLY PROGRESS REPORT, OCTOBER 1—DECEMBER 31, 1978

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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

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).

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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).

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, 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, OCTOBER 1—DECEMBER 31, 1978

S. J. Ball, Manager
J. C. Cleveland   J. C. Conklin
D. G. Lister*

ABSTRACT

Further development of the ORECA code was done based on analyses of Fort St. Vrain (FSV) reactor test data. Work also continued on upgrading the ORTAP, BLAST, and FLODIS codes. Preliminary reverse-flow plume experiments for investigating the FSV upper plenum cover plate heating phenomenon were continued. Further assistance was given to NRC on FSV questions relating to their 100% power license application and the oscillation problem.

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 further analyses in support of the Fort St. Vrain (FSV) reactor request for a 100% power operating license, further development of the FSV simulator codes, comparisons of ORECA code predictions with data from a 30% power FSV reactor scram test, and further investigations of the FSV oscillation problem.

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

J. C. Cleveland   S. J. Ball
J. C. Conklin

Further work was done on the development and documentation of the ORTAP-FSV code[1] for simulating the overall dynamic behavior of the FSV

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*Student research assistant from Denison University, Great Lakes Colleges Association.

reactor. The problems of adapting the BLAST code[2] steam generator reheater dynamics portion to the West German THTR reactor design were investigated. This was done in preparation for a January visit (by J. C. Cleveland) to RWTUV in Essen, Federal Republic of Germany, to assist them in setting up and running the BLAST code. The routine used in both BLAST and ORTAP to perform an initial steady-state search for the steam generator helium and water side nodal pressures, enthalpies, and temperatures and for tube nodal temperatures was modified to achieve convergence with less accurate initial guesses at nodal conditions.

A paper entitled "Investigations of Postulated Accident Sequences for the Fort St. Vrain HTGR" was written and delivered at the Second U.S.-Japan HTGR Safety Technology Seminar held at Tokyo and Fuji, Japan, November 22–26, 1978.[3]

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

S. J. Ball

Several topics were addressed as part of the agreement[4] for ORNL to assist NRC with FSV reactor licensing questions.

The first topic involved identification of the important uncertainties in FSV accident analysis code predictions.[5,6] The major items were core bypass flow fraction, reverse-flow plume heat transfer, afterheat assumptions, delay time for start-up of the firewater cooldown (FWCD) system following a loss-of-forced-convection (LOFC) accident, lower (core outlet) plenum hot streak factors, total primary flow available for emergency cooling, region outlet thermocouple dynamic model (needed for code verification), and effects of gas mixture in the primary coolant following a design-basis depressurization accident (DBDA).

In response to a request by NRC/DSS, analyses of postulated FSV DBDAs and LOFC accidents were made using the ORECA code[7] to determine if initial conditions other than those postulated in the FSAR[8] would lead to more severe temperatures. The analyses indicated that none of the newly postulated cases resulted in temperatures greater than those for the FSAR cases.

Other items related to the FSV licensing review are described elsewhere: the oscillation problem in Sect. 1.4 and development of longer-range experiment plans in Sect. 1.5.

1.3 Comparisons of ORECA Code Predictions with FSV Scram Test Data

S. J. Ball

In previous quarterly reports,[6,9,10] descriptions were given of comparisons of ORECA code predictions with FSV scram test data at 28% (8/6/77), 40% (10/25/77), and 50% (5/8/78) power. Subsequently, a more refined version of General Atomic's (GA's) dynamic model for the region outlet thermocouple was received, along with data from a fourth test at 30% power (7/23/77). This last test had periods of no flow sufficiently long so as to allow many of the region outlet thermocouple readings to increase significantly. Initial ORECA simulations did not show this increase, however. Thus, the ORECA model of the core lower reflector and core support block, which consisted of a single node per region, was modified in order to properly account for the periods of no convection cooling by including an extra node for the lower part of the core support block in each region. Because of the much-reduced heat transfer area in the part of the core support block surrounding the thermocouple, the support block cools down much more slowly than the reflector after a scram, so that T^4 radiation heat transfer to the thermocouple becomes very important in the no-flow periods.

In addition to the support block model change, the general form of the thermocouple dynamic model proposed by GA has been adopted for use in the ORECA code. Based on a detailed simulation of the thermocouple (TC) assembly at nominal full-power conditions, GA derived an equivalent first-order response time constant τ_{TC} of 2.5 min for Eq. (1):

$$\frac{MC_p}{A_{TC}} \frac{d(T_{TC})}{dt} = h_{eff} (T_{eff} - T_{TC}) , \qquad (1)$$

where

M = effective mass of TC assembly, lb,

C_p = effective specific heat of TC assembly, Btu/lb-°F,

A_{TC} = effective heat transfer area of TC assembly, ft^2,

h_{eff} = effective heat transfer coefficient between surroundings and TC assembly, Btu/min-ft^2-°F,

T_{TC} = mean temperature of TC assembly, °F,

t = time, min,

T_{eff} = effective temperature of surroundings, °F,

τ_{TC} = $MC_p/h_{eff}A_{TC}$, min.

The GA model (Fig. 1a) assumes two parallel conductance paths to the thermocouple assembly mass: a convection path (h_c) from the region outlet gas temperature T_{GO} and a radiative path (h_r) from the support block temperature T_{SB}. Therefore,

$$h_{eff} = h_c + h_r \tag{2}$$

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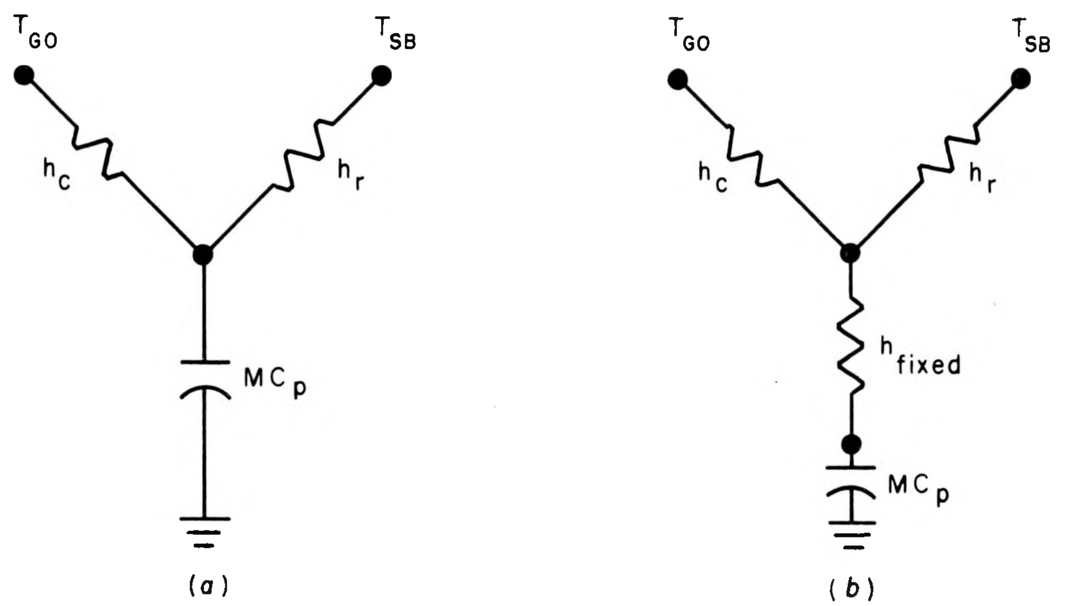

Fig. 1. FSV region outlet thermocouple models: (a) GA model, (b) ORNL model.

and

$$T_{eff} = \frac{h_c T_{GO} + h_r T_{SB}}{h_c + h_r} . \tag{3}$$

Using the GA expressions for h_c and h_r and the values given for A_{TC} and τ_{TC} (at full-power conditions), the resulting value of MC_p/A_{TC} is 11.14 Btu/ft^2-°F compared with an ORNL-derived value (based on the design drawings) of only 1.83 Btu/ft^2-°F. Instead of modifying the model heat capacity to give the proper reference value of τ_{TC}, as GA did, the ORNL model (Fig. 1b) assumes a fixed series conductance h_{fixed} with the value required to give τ_{TC} at nominal full-power conditions. The rationale for inserting h_{fixed} in the model is that it is a better way of representing the spacer conductance between the thick outer graphite sleeve and the thermocouple. While it is probably safe to assume that neither the GA nor the ORNL simplified model is "correct," note that the effective time constant τ_{TC} for the low-flow, low-temperature, off-design conditions is significantly longer for the GA model. In the ORNL parameter optimization studies for matching ORECA predictions to FSV data, this difference results in an optimized value of core bypass flow fraction that is significantly larger for the ORNL model. Because the assumption of larger core bypass flow fractions typically leads to higher maximum predicted temperatures in postulated accident studies, a more detailed off-design analysis of the thermocouple response is recommended.

Optimized model comparisons of the 30% power test (7/23/77) were made as before but using the modified support block model and the ORNL version of the outlet thermocouple model. The resulting agreement of the predicted vs measured refueling region outlet temperature transients was very good. To achieve the optimization, modifications (within reasonable limits) were required of some of the region peaking factors (Table 1). It was also necessary to assume that, after the scram, 23.1% of the total primary flow bypassed the core. This is a significantly higher value of bypass fraction than is in common use by GA and ORNL.

An example of the importance of the effect the thermocouple model has on implied bypass flow fraction is shown in Figs. 2 and 3. Figure 2

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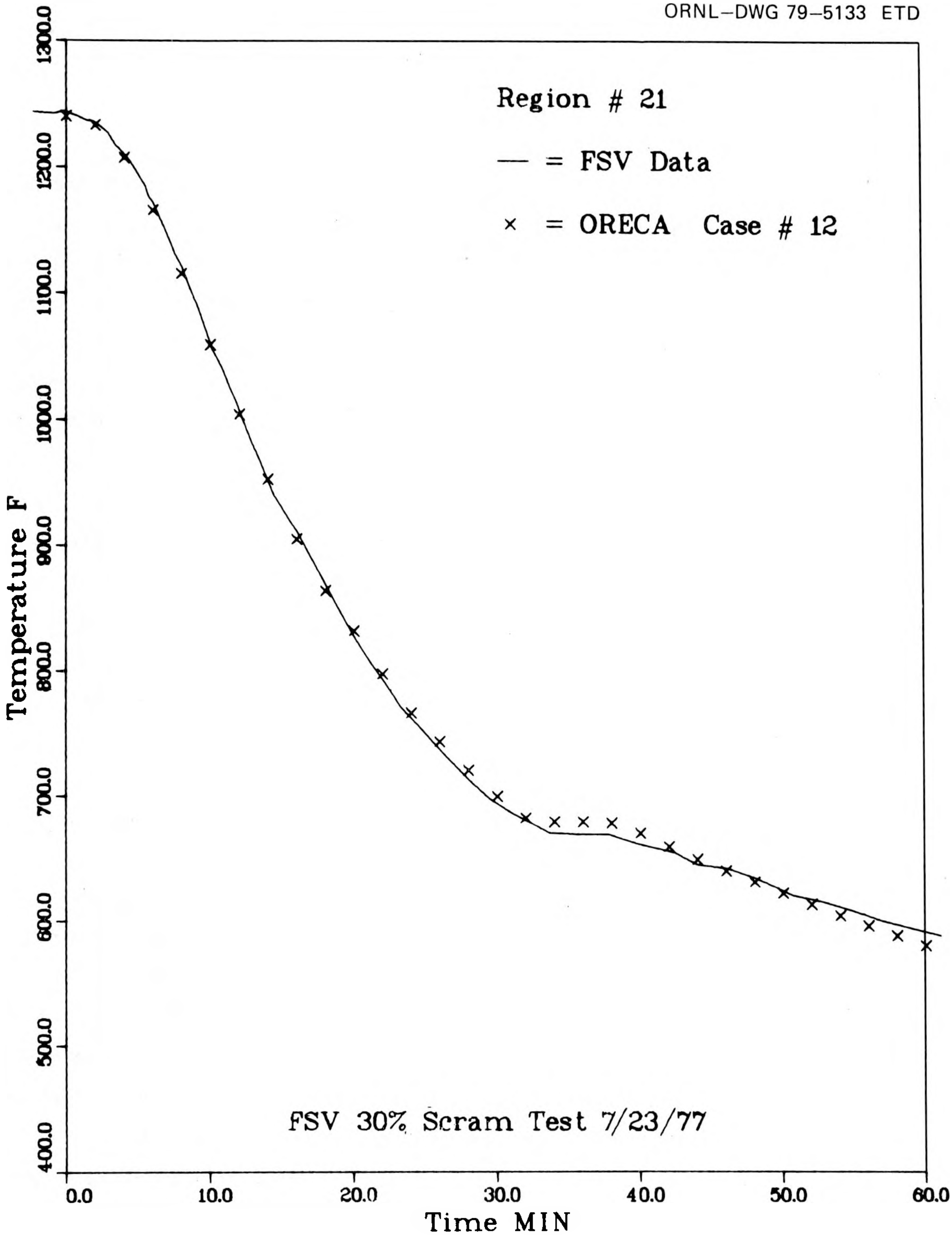

Fig. 2. FSV scram test of July 23, 1977, from 30% power — comparison of "reference" ORECA code predictions of measured gas outlet temperature from region 21 using the GA version of the thermocouple model vs plant data.

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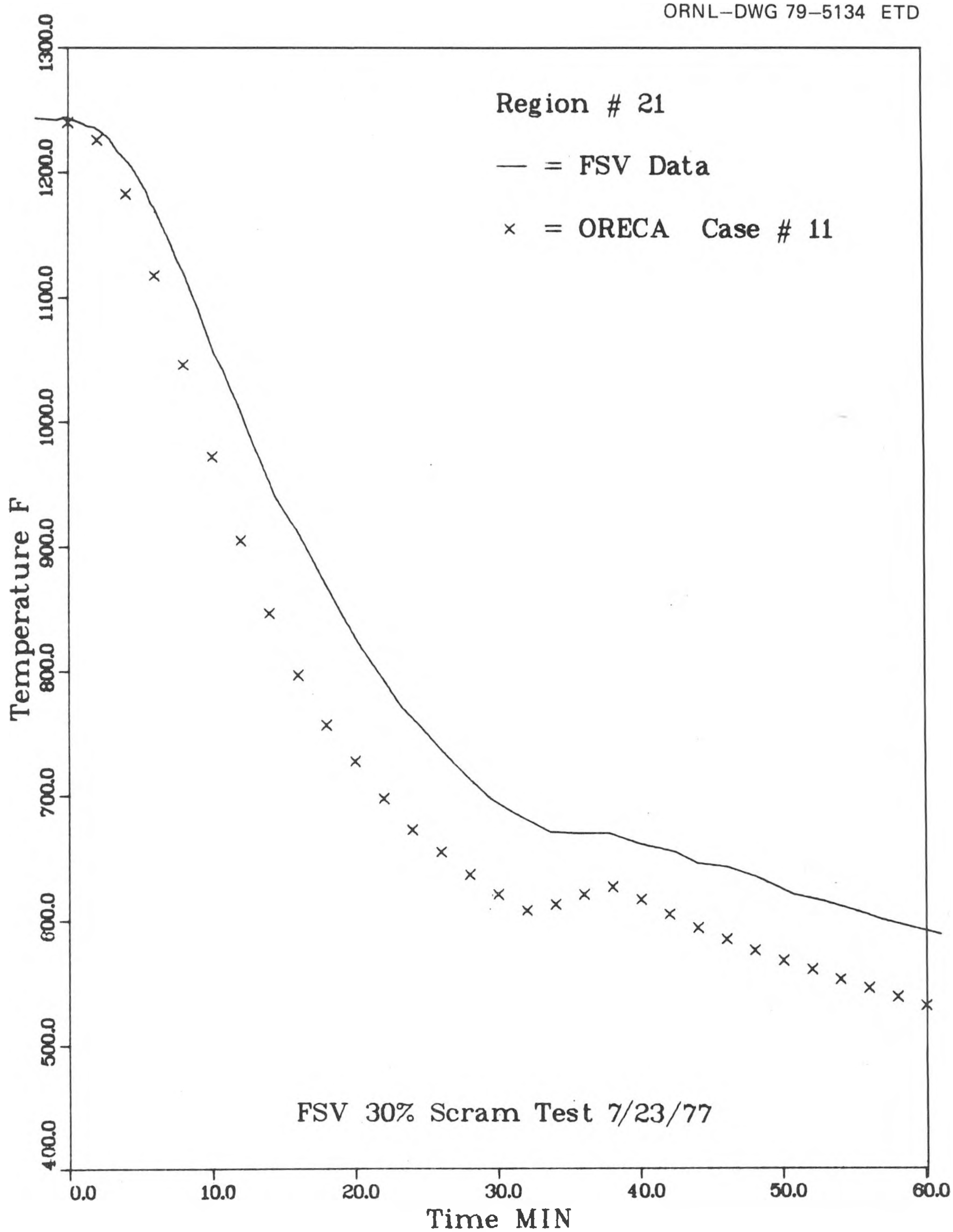

Fig. 3. FSV scram test of July 23, 1977, from 30% power — comparison of "reference" ORECA code predictions of measured gas outlet temperature from region 21 using the ORNL version of the thermocouple model vs plant data.

Table 1. ORECA optimized region peaking factors for the FSV 30% power scram test, 7/23/77

Refueling region	Peaking factor		Refueling region	Peaking factor	
	Original	New		Original	New
1	1.271	Same	21	0.741	0.719
2	1.731	Same	22	0.610	Same
3	1.485	Same	23	0.513	Same
4	1.214	1.170	24	0.646	0.624
5	1.215	Same	25	0.680	Same
6	1.651	Same	26	0.497	0.650
7	1.702	1.680	27	1.173	1.151
8	1.088	1.110	28	0.581	0.537
9	1.428	Same	29	0.540	Same
10	0.882	0.948	30	0.703	0.616
11	0.957	0.979	31	0.978	Same
12	1.298	1.254	32	0.347	0.281
13	1.194	1.150	33	0.877	0.746
14	1.144	1.013	34	0.581	0.494
15	0.882	Same	35	0.366	0.314
16	1.417	Same	36	0.447	0.368
17	1.177	1.199	37	0.773	0.467
18	1.116	1.138			
19	1.102	Same			
20	0.336	0.249			

is an ORECA "reference case" (i.e., no parameter optimization) calculation that uses the GA model and shows that little, if any, extra parameter adjustment would be required. On the other hand, the reference case calculation for the same region using the ORNL model (Fig. 3) shows that substantial adjustments are needed.

The conclusion is that further investigation and testing of the thermocouple response characteristics are needed to resolve the parameter verification uncertainties.

1.4 Investigations of the FSV Oscillation Problem

S. J. Ball

In analyzing magnetic tape records of the FSV oscillation runs (by the Noise Analysis Group in ORNL's Instrumentation and Controls Division), an attempt was made to infer induced reactivity transients from the recorded flux signals. To accomplish this, an on-line computer program was written and debugged. The program was developed by first calculating the at-power power-to-reactivity P/ρ transfer function (using CORTAP[11]) and then implementing the inverse of it with an optimized set of lead-lag digital filters.

Further work was done on the analysis of FSV oscillation data, particularly for the November 4, 1978, tests. A program representative (S. J. Ball) was appointed to a special NRC Technical Review Committee to help advise NRC on the problem.

1.5 Preliminary Heated Plume Experiments

D. G. Lister S. J. Ball

Further preliminary experiments on the FSV upper plenum reverse-flow plume heat transfer were run to determine the validity of using Reynolds (Re) and Grashof (Gr) similarity for air models.[10] For air models of the high-pressure helium plenum to be valid, the Nusselt number (Nu) must be a unique function of Re, Gr, and a geometric scaling parameter H/D, the ratio of plume height to discharge nozzle diameter.

While tests showed fairly conclusively that similarity relations would not hold in the transition flow regime, the results were not conclusive either way for the higher Re tests. As shown in Fig. 4, data for two runs, both having the same Grashof number and H/D but different geometric and temperature parameters, resulted in Nu vs Re curves that should have been the same (to confirm scaling) but were not. However, there is some question as to the significance of the differences, and further, more definitive tests were planned (see Sect. 1.6).

1.6 Proposals for Further Upper Plenum Reverse-Flow Plume Experiments and FSV Lower Plenum Hot Streak Tests

S. J. Ball

As part of the NRC request for longer-range FSV licensing support, problems requiring further investigation and experimentation were to be identified. Two areas involved the reverse-flow plume behavior during

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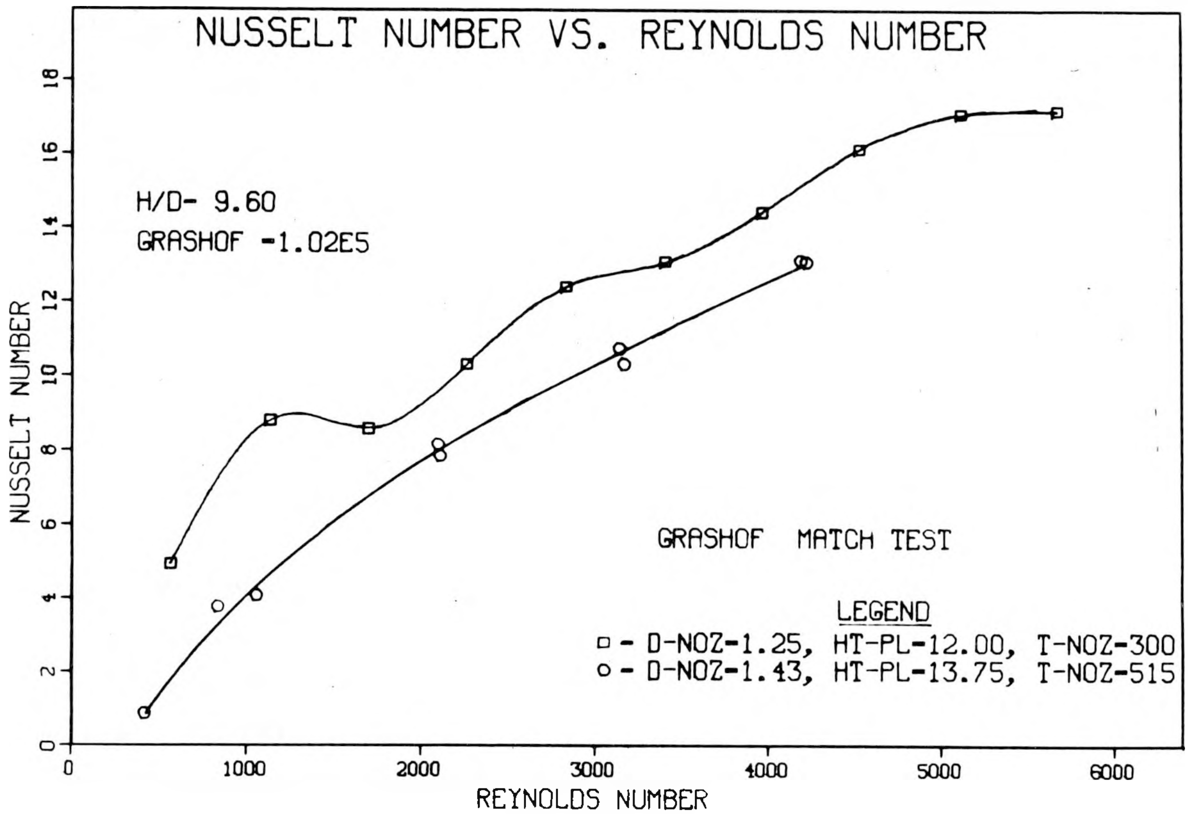

Fig. 4. Preliminary results of reverse-flow plume experiment showing Nu vs Re for two cases with equal Gr numbers.

postulated sustained LOFC accidents and the hot streaking behavior in the core outlet plenum during emergency core cooling system (ECCS) cooling periods. A preliminary proposal to proceed with these tests was submitted.[12]

The major tests involved in the reverse-flow plume investigation were a full-scale heated-air model of several refueling region areas and low-temperature tests at FSV, which would test the ability of the codes to predict the onset of reverse flow. With additional upper plenum thermocouples, some data may also be obtained on effective plume heat transfer.

The proposed hot streak tests at FSV would measure the changes in various steam generator module inlet gas temperatures that result from changes in a particular hot refueling region outlet temperature. Tests would be run during both normal operating periods and postshutdown conditions and would be used to verify the worst-case streaking factor derived by GA from scale-model air tests.

1.7 Development of the FLODIS Code for FSV ECCS Analysis

J. C. Conklin

The FLODIS code[13] has been rewritten to eliminate addressing errors and to improve the computation efficiency of the iterative solution methods. This has significantly decreased the computation time for the LOFC transient. Several FLODIS subroutines were upgraded and routines for implementing arbitrary flow, temperature, and power input functions were added. Debugging is continuing.

2. MEETINGS AND CONFERENCES ATTENDED UNDER PROGRAM SPONSORSHIP

2.1 Meeting to Discuss NRC Information Needs from the Japanese HTGR Program, Silver Spring Md., Nov. 2, 1978

S. J. Ball

NRC advised participants in the upcoming U.S.-Japan HTGR Safety Technology Seminar of information needs. Protocol and details of the trip arrangements were also discussed.

2.2 U.S.-Japan Seminar on HTGR Safety Technology in Fuji, Japan, and Related Site Visits in Japan, Nov. 20–Dec. 1, 1978

S. J. Ball W. G. Dodge*

The purpose of the trip was to participate in the Second U.S.-Japan HTGR Safety Technology Seminar and to observe and discuss related projects in government and industrial laboratories throughout Japan. Details of the trip are reported elsewhere.[14,15]

The theme of the Japanese HTGR program is the development of a 50-MW(t) experimental reactor with a helium coolant outlet temperature of 1000°C. The purpose of this experiment will be to demonstrate the feasibility of using VHTR process heat for nuclear steelmaking. The Japanese are not interested in using the HTGR/VHTR either for electrical or lower-temperature process heat production. The development program includes an impressive array of both operational and near-future tests on all phases of the process except the PCRV. They plan to begin VHTR construction in 1982 and operation in 1986.

The major experiments preceeding the experimental VHTR are (1) the Oarai Gas Loop-1 (OGL-1), presently operational in the Japan Materials Test reactor; (2) the Intermediate Heat Exchanger (IHX) 1.5-MW test loop at IHI-Yokohama, which attained its design temperature of 1000°C during

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*Research Engineer, DOE-Sponsored PCRV Research and Development Program.

the visit; and (3) the Helium Engineering Demonstration Loop (HENDEL), which will be operational at the Japan Atomic Energy Research Institute (JAERI) Tokai site in 1981. HENDEL is a large-scale helium test loop for high-temperature components such as pipes, valves, heat exchangers, fuel blocks, core support structures, and major elements of the direct-reduction steelmaking process.

At all of the laboratories visited, there appeared to be significant work in progress on materials testing, primarily because of the difficulties involved in 1000°C operation and in the He/H_2 high-temperature heat exchangers needed in the steam reformer and reducing gas heater sections of the steelmaking process. Much work was also being done on the seismic response of prismatic block cores.

HTGR dynamics code development work in Japan parallels our own. JAERI is developing a one-dimensional neutron kinetics plus 3-D thermal-hydraulic core kinetics code called SCOTCH, and IHI has developed a RECA-like core thermal model code called EMCOC, as well as an overall plant dynamics code PLANDY. Informal code information exchange agreements were set up with JAERI and IHI. Lengthy discussions of the FSV oscillation problem were also held.

The Seminar technical meeting at Fuji included four parallel sessions: accident delineation, material properties and design methods, helium technology, and seismic research. Preliminary but essentially complete proceedings of all sessions were distributed at the meeting.

Government laboratory sites visited were the JAERI laboratories at Tokai and Oarai and the National Institute for Metals at Tsukuba. Industrial sites included the MHI Technical Institutes at both Nagasaki and Takasago, the Engineering Research Association for Nuclear Steelmaking (ERANS) offices in Tokyo, and the IHI Research Institutes at Yokahama and Toyosu (Tokyo).

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