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OAK RIDGE NATIONAL LABORATORY

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Summary of ORNL Work on NRC-Sponsored HTGR Safety Research, July 1974–September 1980

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LISTING OF QUARTERLY PROGRESS REPORTS THROUGH FY 1980

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June 30, 1975	ORNL/TM-5021, Vol. IV
September 30, 1975	ORNL/TM-5123
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June 30, 1980	ORNL/NUREG/TM-415
September 30, 1980	ORNL/NUREG/TM-429

LIST OF ACRONYMS

AEC	Atomic Energy Commission
AVR	Arbeitsgemeinschaft Versuch Reaktor, an HTGR in FRG
BNL	Brookhaven National Laboratory
BOP	balance of plant
CACS	core auxiliary cooling system
CEGB	Central Electricity Generating Board
CFTL	Core (Component) Flow Test Loop
CPU	central processing unit of a computer
CSMP	Continuous System Modeling Program (IBM)
DBDA	design basis depressurization accident
DOE	Department of Energy
DSS	Division of System Safety, NRC
DTR	Division of Technical Review, NRC
EGCR	experimental gas-cooled reactor
FRG	Federal Republic of Germany
FSAR	final safety analysis report
FSV	Fort St. Vrain
FWCD	firewater cooldown
FY	fiscal year
GA	General Atomic Company
GCFR	gas-cooled fast breeder reactor
Gr	Grashof number
H/D	height to diameter ratio
HTGR	high-temperature gas-cooled reactor
IBM	International Business Machines Company
ICRD	instrumented control rod drives
IHI	Ishikawajima-Harima Heavy Industries Company, Limited
JAERI	Japan Atomic Energy Research Institute
KFA	Institut fur Reaktorentwicklung at Kernforschungsanlage Julich
LANL	Los Alamos National Laboratory
LHTGR	large HTGR
LOFC	loss of forced convection cooling accident
LTR	licensing topical report
LWR	light-water reactor
MW(t)	megawatt thermal
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation Office in NRC
Nu	Nusselt number
ORGDP	Oak Ridge Gaseous Diffusion Plant
ORNL	Oak Ridge National Laboratory
PCRV	prestressed concrete reactor vessel
PSC	Public Service Company of Colorado
RCD	region constraint devices
R&D	research and development
Re	Reynolds number
rpm	revolutions per minute
RSR	Reactor Safety Research Office in NRC
RWTUV	Rheinisch-Westfalischer Technischer Uberwachungs-Verein e.V.

LIST OF ACRONYMS (continued)

SASA	severe accident sequence analysis
SI	Systeme Internationale (scientific units notation)
SNM	strategic nuclear materials
THTR	thorium high-temperature reactor, FRG
TGO	refueling region gas outlet temperature
UK	United Kingdom
3-D	three-dimensional

SUMMARY OF ORNL WORK ON NRC-SPONSORED HTGR SAFETY RESEARCH, JULY 1974–SEPTEMBER 1980

S. J. Ball, Manager

ABSTRACT

A summary is presented of the major accomplishments of the Oak Ridge National Laboratory (ORNL) research program on High-Temperature Gas-Cooled Reactor (HTGR) safety. This report is intended to help the Nuclear Regulatory Commission establish goals for future research by comparing the status of the work here (as well as at other laboratories) with the perceived safety needs of the large HTGR. The ORNL program includes extensive work on dynamics-related safety code development, use of codes for studying postulated accident sequences, and use of experimental data for code verification. Cooperative efforts with other programs are also described. Suggestions for near-term and long-term research are presented.

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1. INTRODUCTION AND SUMMARY

The Nuclear Regulatory Commission's (NRC's) Reactor Safety Research (RSR)-Sponsored high-temperature gas-cooled reactor (HTGR) Systems and Safety Analysis program was initiated at Oak Ridge National Laboratory (ORNL) in July 1974, when General Atomic Company (GA) had a number of orders to build large commercial HTGRs. Initially, the program was intended to develop an independent capability to perform and assess HTGR safety analyses that related to system dynamics and postulated accident transients. Such analyses were deemed necessary because at that time, only the vendor-developed codes were available for conducting in-depth studies of accident sequences. Other NRC-developed accident simulation codes for different reactor types were not directly applicable because of the many unique design and safety features of HTGRs.

A summary follows of the major accomplishments of the ORNL program [through fiscal year (FY) 1980]. More detailed descriptions of these items are given in the body of the report:

1. Computer codes were developed for dynamic simulation of the major HTGR system components. These codes can be run independently or combined as an overall system code. Codes were adapted to simulation of the Fort St. Vrain (FSV) reactor and, in their earlier stages, to the 2000- and 3000-MW(t) reactor designs. The codes developed include:

a. ORTAP – an overall system code for the HTGR
b. CORTAP – core simulation for at-power transients, including point neutron kinetics and single-average-channel thermal hydraulics

c. ORECA — core simulation for emergency core cooling transients, including three-dimensional (3-D) thermal hydraulics
d. BLAST — once-through steam generator and reheater simulation
e. ORTURB — detailed model of the turbine generator plant
f. ORCIRT — circulator turbine simulation
g. CACS — a core auxiliary cooling system (CACS) simulation was completed but not incorporated into the overall system code because of discontinuation of the Large HTGR (LHTGR)
h. FLODIS — an alternate 3-D core thermal hydraulics code developed originally under another program was refurbished for use as a backup for ORECA analyses.

2. Independent evaluations were conducted of GA dynamic simulation codes used for licensing calculations, and these assessments were used by the NRC as a basis for accepting the codes.

3. Detailed analyses of reactivity insertion accidents using ORTAP were conducted and published. These analyses provided confirmation of FSV final safety analysis report (FSAR) calculations.

4. Technical support was provided to the Office of Nuclear Reactor Regulation (NRR) on licensing questions relating to the FSV initial rise to power operation.

5. The program responded to an NRR request in April 1978 for assistance in five areas relating to FSV 100%-power licensing questions:

a. provided audit calculations of the firewater cooldown (FWCD) and design-basis depressurization accidents (DBDA) using ORECA;
b. provided detailed calculations and parametric studies of the FWCD accident with estimates of critical component thermal histories (e.g., upper thermal-barrier cover plates) using ORECA and other codes;
c. responded to NRC questions about the ORNL review of RECA;
d. provided continuing on-call assistance in support of licensing questions;
e. informed 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) did provide bounding consequences for other accidents identified within the FSAR.

6. Cooperative HTGR safety information exchange programs with Japan and The Federal Republic of Germany (FRG) were implemented that have resulted in considerable benefits to all parties.

7. Sensitivity studies were conducted in support of the accident reanalysis for FSV full-power licensing to identify important uncertainties affecting the predictions.

8. Code verification work was done in which FSV transient data were compared with simulator predictions. Comparisons of ORECA code predictions with data from four FSV scrams indicated generally good agreement but also indicated several problem areas that led to improvements in the code and a better understanding of core transient behavior. Similar analyses were done for CORTAP using FSV rod-jog experimental data. A verification analysis was also initiated for the BLAST code using FSV and Arbeitsgemeinschaft Versuch Reaktor (AVR) data.

9. Technical support and ad hoc analyses were provided to NRR in licensing-related reviews of the FSV oscillation problem.

10. Extended analyses and small scale-model experiments were used to resolve licensing questions relating to overheating of FSV upper-plenum cover plates by reverse or upward-flow plumes from the core during extended loss of forced-convection cooling (LOFC) accidents.

11. Technical support and analyses using the ORECA and FLODIS codes were provided to NRR in their evaluation of possible thermal stress problems in the FSV core-support regions during the cooldown following a design-basis earthquake LOFC accident.

12. A proposed experimental program was developed for FSV tests that would provide data required for critical areas of accident code verification.

A year-by-year summary of the program funding that shows both the funds received and the amount actually spent follows:

Fiscal year	Funds received ($000)	Actual cost ($000)
1975	120	104
1976	273	288
1977	230	166
1978	85	68
1979	160	110
1980	150	129
Totals	1018	865

The carry-over into FY 1981 was $153,000.

A chronological listing of all the program's quarterly progress reports through FY 1980 also is included in this report.

2. LONG-RANGE HTGR SAFETY RESEARCH PROGRAM PLANS

Two major documents elaborate on HTGR safety-related research needs and recommend approaches to resolve the problems. The two documents are ORNL's *Planning Guide for HTGR Safety and Safety-Related Research and Development*,[1] which was published in May 1974 as a cooperative effort with GA and the Atomic Energy Commission (AEC), and an RSR/NRC draft report *Program Plan for Confirmatory HTGR Safety Research*,[2] issued in February 1975.

The *Planning Guide* identifies specific tasks required for a comprehensive study and understanding of HTGR safety problems. It is divided into seven task areas. The first is "System and Safety Analysis," which includes primarily the types of accident sequence analyses studied in the present RSR program. The other six sections deal with specific areas of technology: fission products, primary coolant, vibration and seismic studies, confinement and containment, materials, and safety instrumentation. Detailed discussions of problem areas are given, with recommendations for programs required to solve the problems, and manpower and cost estimates are provided for each task.

The RSR program plan also presents detailed descriptions of generic problem areas, sets priorities, and recommends research programs for establishing independent assessment capabilities. Because this plan represents the basis of much of the work done in the ORNL program, an outline of the parts of the plan relevant to the ORNL program will be given, with a brief comparison of points in the plan and accomplishments by the program.

2.1 Objectives of the RSR Program Plan

The major objective of the program described by the plan is to provide independent capability for overall safety assessments of HTGRs, emphasizing the consequences to public safety of abnormal and accident conditions. A special aim is to assist NRR in its evaluation of safety problems. Needed research is divided into two categories. Topics in the category of Phenomenological Research covered by the plan are mostly addressed by research programs at other laboratories. The topic of safety instrumentation and control systems is addressed in the ORNL program in that it is related to overall systems analysis work. In the analytical research category, several topics in the plan are addressed by the ORNL program: (1) accident delineation, (2) phenomena modeling and systems analysis, and (3) proof tests. The ORNL program has assisted NRR on a number of occasions, noted in Sect. 4.5.

2.2 Noise Analysis and Dynamic Testing

This work is described as part of the safety instrumentation and control systems research topic (Chap. 6 in Ref. 2). The scope of this work includes (1) assessment of existing test methods, (2) determination of the feasibility of specific HTGR model verification and surveillance applications, and (3) evaluations of trial applications. Test plans for various model verification (but not surveillance) applications have been developed in the ORNL program, and those that have been implemented are described in Sect. 4.6. Plans have been developed but not implemented for frequency response tests (employing small perturbations in control rod position circulator speed and turbine inlet valve position) and various core model verification tests to determine bypass-flow fractions, reverse flow phenomena, and plenum mixing.

2.3 Accident Delineation

The objective of this task is to identify accident sequences that have potential impact on public safety and to provide descriptions or simulations of these sequences that can be used in probabilistic analyses, phenomena modeling, and systems analysis. In the ORNL program, identification, analyses, and delineation of many of the major HTGR design-basis accidents were accomplished. These efforts are discussed in Sects. 4.4, 4.5, and 4.8.

2.4 Phenomena Modeling and Systems Analysis

The object of this task is to develop independent and appropriately validated analytical models and simulations that can be used for predicting HTGR system response to postulated events, including accident sequences. This task constituted the major portion of the ORNL program and is described in Sects. 4.3 and 4.6. A related task was a review of the HTGR vendor's accident analysis codes noted in Sect. 4.2.

2.5 Proof Tests

The objective of the proof tests is to validate the component and system simulation codes. Work accomplished by the ORNL program in this area is described in Sect. 4.6.

3. RELATIONSHIP WITH OTHER PROGRAMS

3.1 NRC-Sponsored Programs

The ORNL RSR-sponsored program has benefited from collaboration and information exchanges with sister laboratory programs at Los Alamos National Laboratory (LANL) and Brookhaven National Laboratory (BNL). Several meetings were held with LANL project personnel to discuss common problems in the development of ORNL's ORTAP code and LANL's CHAP code. A recent cooperative effort with the LANL stress analysis group on the FSV core-support thermal stress problem (Sect. 4.9) resulted in a satisfactory resolution of the problem in February 1981.

Our main collaboration with BNL has been in the area of code review. A recent BNL review[3] of the ORECA code was very useful and resulted in many improvements to the code, including a much more accurate treatment of the core conduction equations. We are presently looking forward to their review of the BLAST code.

We have also participated in joint efforts with other ORNL-NRC programs, particularly concerning noise analysis and diagnostics in the FSV oscillation problem studies.

3.2 DOE-Sponsored Programs

Close ties have been maintained with the ORNL Department of Energy (DOE)-sponsored HTGR programs. On several occasions, RSR-developed codes were used to solve specific DOE-sponsored problems in such areas as steam generator stresses and core graphite oxidation. Liaison with the advanced HTGR projects also helps in anticipating future safety requirements.

4. SUMMARY OF ORNL PROGRAM MAJOR ACCOMPLISHMENTS

4.1 Familiarization of Project Personnel with HTGR Design, Operation, and Safety

When the RSR-sponsored HTGR safety program began at ORNL in July 1974, only a few of the project personnel had significant experience with gas-cooled reactor design and dynamics. Collectively, however, the principal participants had gas reactor system experience with commercial HTGRs, FSV, the experimental gas-cooled reactor (EGCR), the ORNL Pebble Bed Reactor Experiment, and in-pile gas-cooled loops. Other relevant experience included core physics, power and experimental reactor dynamics, steam generator dynamics modeling, and a variety of reactor safety analyses.

Several program personnel attended a one-week course on HTGRs in September 1974 given at the University of Tennessee as a part of Tennessee Industries Week. Early in the program, participants attended several Advisory Committee on Reactor Safeguards (ACRS) and licensing review meetings, meetings with the HTGR vendor (GA) and major utility [Public Service Company of Colorado (PSC)], and with sister laboratories (LANL and BNL) engaged in HTGR safety research — all of which could be considered part of a familiarization program.

A subcontract was initiated with T. W. Kerlin of the University of Tennessee to compare the modeling and analytical solution methods in common use for once-through steam generators. Professor Kerlin had considerable experience in dynamic modeling of nuclear power plants in general and of nuclear steam generators in particular.

Another part of the familiarization program was the acquisition and review of GA's major dynamic simulation and safety analysis codes, including TAP, RECA, LAP, BLOOST, POKE, CONTEMPT-G, OXIDE-3, and others. Reviews of the key licensing topical reports (LTRs), the FSV FSAR, and a general literature survey that included earlier U.S. research and development (R&D) and United Kingdom (U.K.) Dragon program reports were also part of the initial effort.

Since the initial familiarization process, continued close liaison has been maintained with GA, PSC, and the DOE-sponsored HTGR program at ORNL. Useful interaction and some joint programs are discussed in Sect. 4.7.

4.2 Review of the General Atomic Codes TAP and RECA

At the request of NRC, two of the GA dynamic simulation codes used for licensing analyses were evaluated[4]. The evaluations of the TAP (Transient Analysis Program)[5] and RECA (Reactor Emergency Cooling Analysis)[6] codes were based on (1) careful reviews of the reference documents and (to some extent) the coding and (2) comparisons of the results of benchmark calculations with those of other codes and with experimental data. Note that both TAP and RECA are very complex and sophisticated general-purpose codes. In such cases, the adequacy and accuracy of any given analysis depends strongly on user input, the severity of the postulated accident, and other factors that cannot be evaluated on the basis of the codes themselves. However, numerous opportunities have occurred to make direct comparisons between the GA and independently developed ORNL codes, and the agreement has been generally quite good.

In the CORTAP report,[7] comparisons were made of the steady-state core temperature conditions at 100% power predicted by TAP and CORTAP, and the agreement was very good. Comparisons of steam generator responses predicted by TAP and by BLAST,[8] which is the steam generator portion of ORTAP, showed very close agreement even though the models and solution techniques are different. Comparisons of a benchmark case for a DBDA transient are shown in Fig. 1.

In the course of performing several accident analyses, RECA and ORECA have been run for the same set of assumed operating conditions, and the results have been in good agreement. In an analysis of the FSV DBDA in which reduced Pelton-wheel drive speeds were assumed, the RECA and ORECA results were quite close (Fig. 2) and in neither case are operating limits reached. In subsequent calculations of postulated DBDA and LOFC accidents, and in predictions of measured FSV core-region outlet temperatures following scrams, the agreement between RECA, ORECA, and the data was all generally very good. More detailed commentaries on specific modeling assumptions, computation techniques, and the codes' susceptibility to misuse or input errors are given in a review report.[4]

A major long-range concern about the GA code reports was the lack of a sufficiently aggressive experimental verification program. Plans for some verification tests were outlined briefly; however, comparisons of theoretical computations with experimental data derived from "normal operating transients" are typically insufficient. Much care and planning is required to ensure against extraneous perturbations such as unexpected or unrecorded operator actions. In particular, verification of dynamic simulation codes requires that the test perturbations excite the dynamic features of the component of interest to ensure a proper measurement. We feel that PSC should pursue detailed verification plans and tests more aggressively. As the utility that owns the PSV plant, PSC would benefit from such tests by using confirmation data to (1) reduce uncertainty, (2) possibly reduce the imposed conservatism in the predictions, and (3) relax operating restrictions.

Because in a number of cases the comparisons of the GA code data with both experimental data and output from independently generated codes have been good, the credibility of the GA codes was very high.

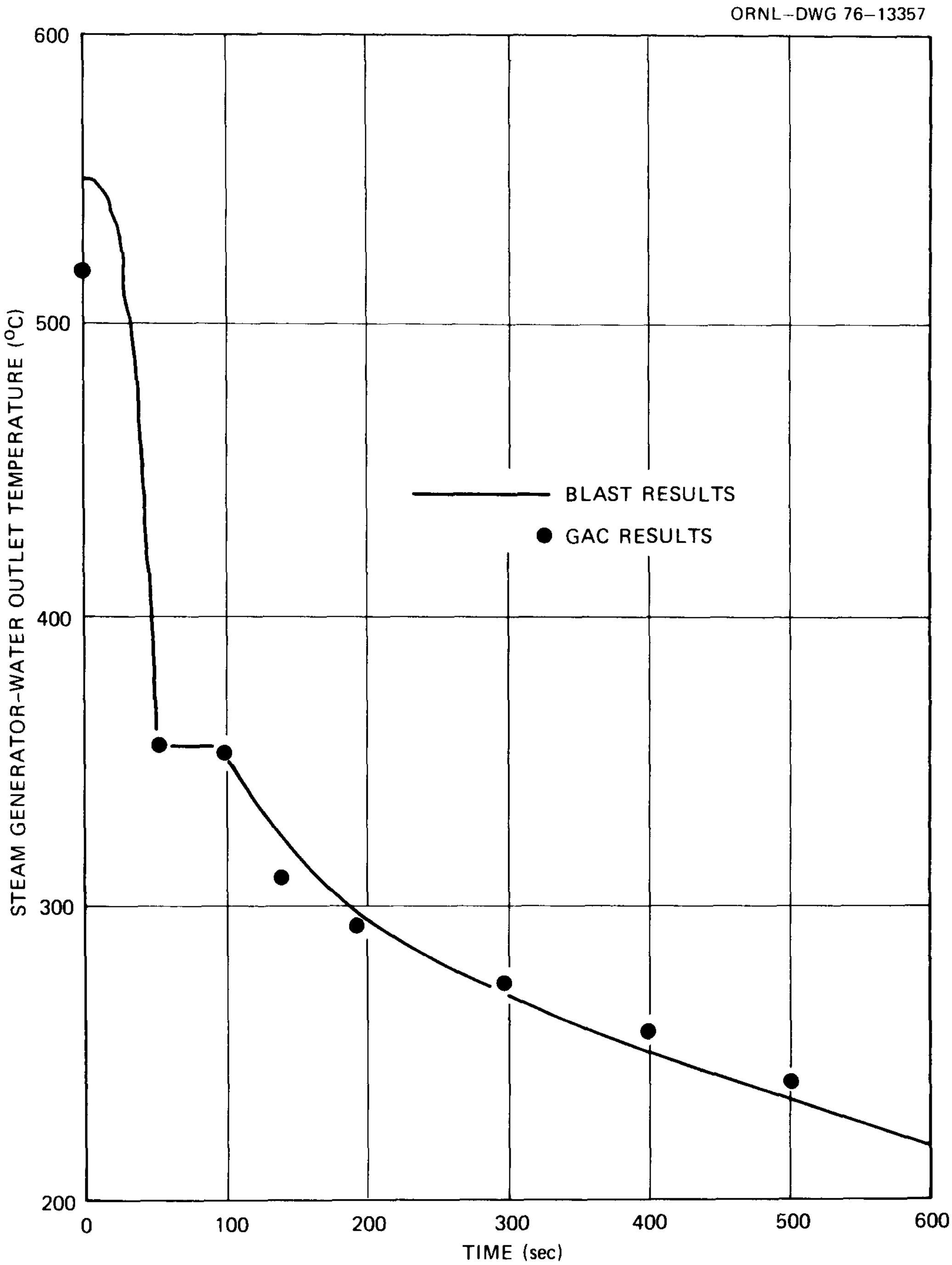

Fig. 1. Steam-generator water-outlet temperature following initiation of a DBDA [°F = 9/5(°C) + 32].

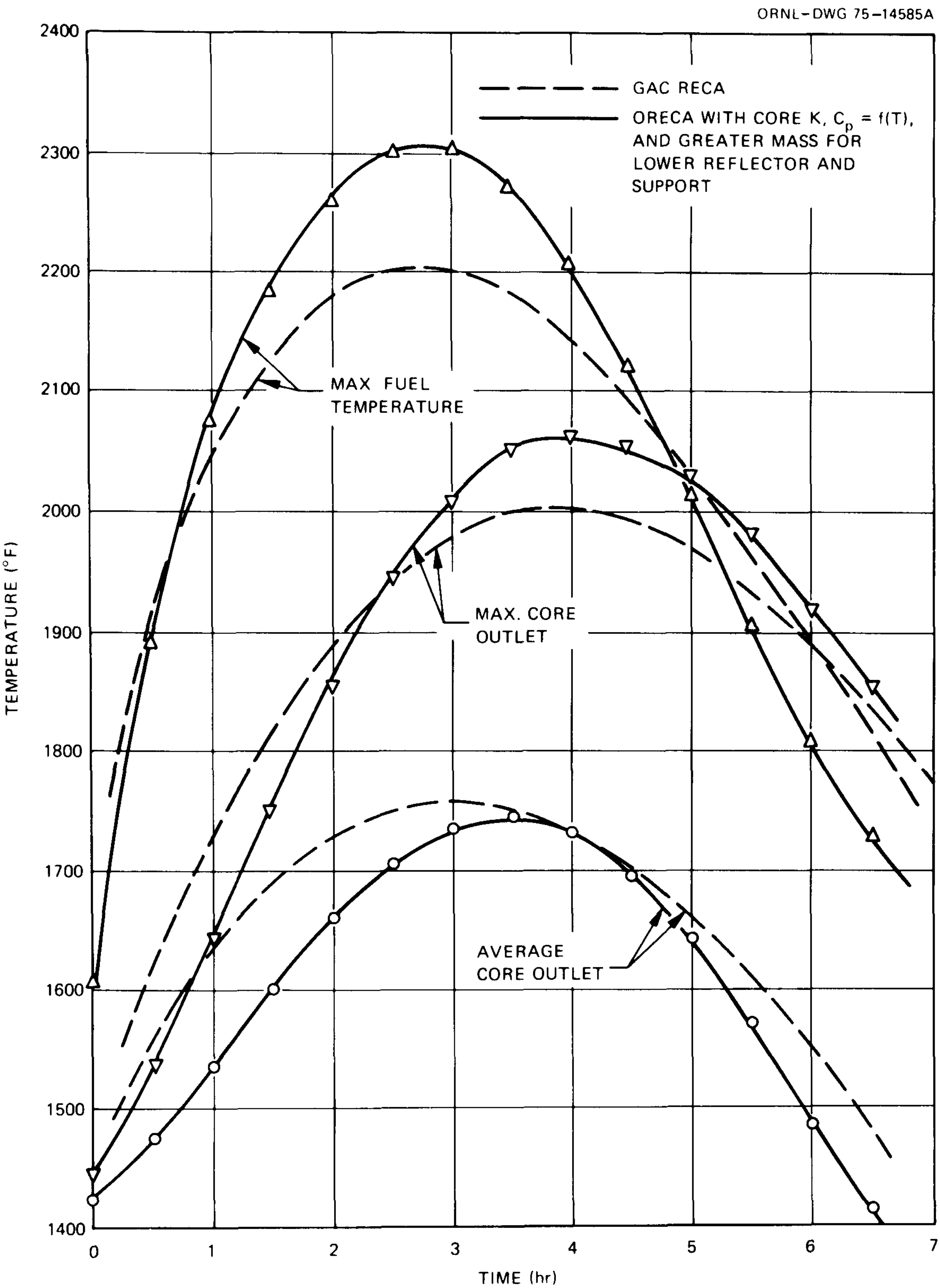

Fig. 2. Predicted system response of the FSV plant to a depressurization accident [°C = 5/9(°F) − 32].

4.3 Independent Development of HTGR Component and System Dynamics Codes

4.3.1 The ORTAP code[9,10]

The ORTAP code was developed as an independent method of predicting the dynamic response of the HTGR nuclear steam supply system to a wide range of conditions. The proprietary TAP[5] code is used at GA to predict HTGR transient response. The ORTAP code contains coupled component simulations of the core (CORTAP,[7] ORECA[11]), the reheater and steam generator (BLAST[8]), the helium circulator and circulator turbine, and the balance of plant (BOP). The major plant control systems are also modeled.

The core is normally simulated by a coupled heat transfer-neutron kinetics single-channel model (CORTAP). An alternate core model (ORECA) is used for transients involving post-trip power and flow conditions. The ORECA model includes 3-D temperature distribution calculations, accounts for the varying flow distribution among the individual refueling regions, and represents flow reversals.

The reheater and steam generator are simulated with the multinode, fixed-boundary, homogeneous-flow model (BLAST). Time-dependent conservation of energy, mass, and momentum equations for both the helium side and the water/steam side are solved by an implicit integration technique. Transients involving both start-up and flood-out of the steam generator can be simulated.

A detailed model of the turbine-generator plant is necessary to accurately predict primary system component response because of the close coupling between the primary system and the secondary system. The systems are coupled by heat transfer in the reheater and steam generator. Also, the exhaust steam from the high-pressure turbine drives the helium circulator turbine before entering the reheater. Additional coupling is introduced by the plant control loops. The turbine-generator model determines pressures, enthalpies, and flows at several points, including extraction and exhaust lines in the high-, intermediate-, and low-pressure turbines. The dynamic response of each feedwater heater and the deaerator is explicitly treated in the model. The circulator-turbine model includes calculations of the steam-side turbine pressure, flows, and enthalpies and the helium-circulator-side pressure rise, flow, and temperature rise. The circulator turbine speed and pressure ratio controls are included.

Following the turbine trips, steam to drive the circulator-turbine is provided by the main steam bypass system, which includes a desuperheater and a flash tank. The dynamic response of these components is included in ORTAP.

The present version of ORTAP is developed specifically for the FSV plant, although with changes in input and minor program modifications, ORTAP could be used to simulate other HTGRs. The extent of the modifications required would, of course, depend on the new plant design but would not be difficult if the changes primarily involved scale-up.

Since the initial development of the ORTAP code, numerous improvements have been made that have resulted primarily from application of the code to specific problems. Changes have been made in the following areas:

1. Improvements were made in the turbine, feedheater, circulator turbine, and main steam bypass systems simulations that resulted in better convergence and run-time efficiency, especially for low-flow and off-normal operating conditions.[12]

2. A correction was made to the ORCIRT circulator turbine subroutine; this correction had little effect on the cases run to date but would have made a significant difference for a depressurization accident.

3. 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 was also done to make the code more readily exportable.

4. The code was modified to account for the 152-s rod-bank insertion time on a scram and for several variations of the minimum assumed scram reactivity available. A standard GA-supplied scram-power-vs-time curve had been used previously. In the updated version, the reactor core power derived from the kinetics equations is used until the power level falls to the point where it is equal to the standard scram curve, which accounts for afterheat. Thereafter, the scram curve is used. This modification resulted in higher predicted peak fuel temperatures for sample accident cases.

5. The ORTAP code was modified to simulate turbine-trip transients in which the reheat-steam temperature control system, which uses reactor power level as a dependent variable, functions to keep the plant operating. Several successful example turbine-trip runs were executed.

6. The circulator-turbine model (ORCIRT) was modified to simulate the turbines operating on wet steam. The main steam-bypass system subroutine was also modified to dampen the computation noise in the calculated bypass flow seen during parts of the shutdown transients.

7. Simulation of plant operation at low-power (~25%) conditions was achieved. Substantial adjustment of control system parameters resulted in good agreement with most plant operating characteristics.

8. Improvements to several steam property subroutines and the feedwater-heater subroutine resulted in substantial decreases in computer running time for certain transients.

9. Alternative models and subroutines were developed for the steam lines in the turbine plant. The original ORTAP steam line model requires small computation time steps because it uses a simple explicit integration method. This steam line time step is the smallest one and tends to limit the maximum value for the overall code.

This model includes the (1) main line, (2) high-pressure turbine, (3) cold reheat steam line, (4) hot reheat steam line, (5) main steam bypass line, and (6) reheat steam bypass line. In all, five plenums are considered: the steam generator, cold reheat, helium circulator, hot reheat, and flash tank plenums. Heat and mass balances were maintained in each plenum. Turbine steam flow is calculated by a simplified turbine model. Pipe pressure loss is considered using a constant turbulent-flow friction factor, and pipe heat capacity is accounted for as a lump mass.

This system has about 30 state variables and therefore 30 differential equations that are coupled. These are solved by MATEX2, a modification of the MATEXP code,[13] which accounts for full coupling of state variables. As a result, calculation time is proportional to the square of the number of state variables. The MATEX2 code eliminates insignificant coupling terms.

In a sample transient calculation, a time step of 0.4 s was applied, and the calculation (CPU) time for a 100-s turbine plant transient was 15 s on the IBM 360/91.

10. The version of ORTAP in use during the past years [and as sent to Rheinisch-Westfalischer Technischer Uberwachungs-Verein e.v. (RWTUV) and Ishikawajima-Harima Heavy Industries Company, Limited (IHI)] appeared to execute properly on the ORNL computer system. However, when this version was run at the Oak Ridge Gaseous Diffusion plant (ORGDP) computer site, errors occurred. The execution errors at ORGDP were traced to the ORTAP subroutine SUPORT, which calculates the average reactor-core outlet gas temperature. The SUPORT code was modified to eliminate errors related to underflows and differences in variable initialization techniques. A few minor corrections to the subroutine logic were also implemented to account properly for coolant flow through the control-rod guide tubes.

The corrected code now executes properly at both the ORNL and ORGDP computer sites without the necessity of masking underflows. Card decks of the revised subroutine SUPORT have been transmitted to RWTUV and IHI.

The present status of the ORTAP code is that of a fully operational simulation of the overall FSV plant, and it has been and can be applied to a variety of postulated accident and transient studies. One of the code's major limitations is that its setup for a particular transient study usually requires coding changes as well as input data changes. Future development of ORTAP is planned to make the code more readily adaptable to a variety of prescribed transients by means of input data changes only. Other efforts are planned in incorporating auxiliary control-loop simulations, generally improving the code's running efficiency and developing alternate models (e.g., a simplified core or steam generator) that more nearly match the needs of specific user problems.

Descriptions of applications of the ORTAP code to specific analyses are given in other sections of this report.

4.3.2 The CORTAP code[7]

The CORTAP code simulates the HTGR core thermal and neutronic response to normal operational transients and to postulated accident conditions. This response is determined by coupling the neutron kinetics equations to the heat transfer equations for the fuel, moderator, and coolant in a representative region of the reactor core. The model represents a unit cell consisting of a fuel stick, the surrounding graphite moderator, and coolant channels in the average power region. The code also has the capability to determine conservative values of fuel, moderator, and coolant temperatures in the "hot" fuel region.

The major features of CORTAP follow:

1. Up to 60 nodes can be used to represent an average or hot fuel stick, the surrounding graphite and coolant channels, the top and bottom reflector elements, and the core-support block. The model includes the temperature dependence of the fuel and moderator conductivity, density and specific heat, and the helium transport properties. Therefore, up to 60 first-order, nonlinear, inhomogeneous differential equations are used to represent the core thermal response.

2. Heat transfer from the graphite to the coolant is calculated based on the helium flow regime (turbine-transitional-laminar).

3. The neutron kinetics behavior of the core is modeled using space-independent neutron kinetics equations with six groups of delayed neutrons. The "prompt jump" approximation is not made.

4. Fuel and moderator temperature coefficients of reactivity are considered temperature dependent.

5. The neutron kinetics equations are coupled to the heat transfer equations through a rapidly converging iterative technique, so that correct fuel and graphite temperatures are used in determining the feedback reactivity rather than temperatures existing at the end of a previous time step.

6. A smaller computational time step is used for the solution of the neutron kinetics equations than is used for the solution of the heat transfer equations, because the response of the reactor power to reactivity changes is much faster than the response of fuel and moderator temperatures to changes in core power.

7. For transients involving a reactor trip, the core heat-generation rate is determined from an expression for power decay following a scram.

8. Input to the code includes the coolant flow rate and inlet temperature as functions of time. Axial relative power-peaking factors are input and assumed constant during transients. The time dependence of the component of the reactivity change caused by control rod motion must also be input.

9. The CORTAP code can be used alternatively as a stand-alone code for analysis of HTGR at-power core transients or as a subroutine of the ORNL overall HTGR plant dynamics code ORTAP.[9]

The CORTAP code was developed both as an aid in the evaluation of the GA system transient analysis code TAP[5] and as an independent method of analyzing transients affecting the HTGR core. Reference 7 describes the techniques used in the CORTAP simulations, comparisons of CORTAP results with results obtained by GA, input instructions, and sample input. Secion 4.6.2 of this report describes comparisons of CORTAP results with FSV transient data. The only additional follow-up work presently planned for CORTAP development is generation of an alternative simplified model that can be substituted for CORTAP (optionally) in the overall systems code. One version of this model has already been developed and checked out in a DOE-sponsored advanced-HTGR simulation program. However, this version uses a simulation language, Continuous System Modeling Program (CSMP), that is not really adaptable to ORTAP coding.

4.3.3 The ORECA code[11]

The ORECA code was developed to predict the 3-D transient thermal-hydraulic behavior of HTGR cores for specified accident and emergency shutdown conditions. It was modeled after GA's RECA code[6] to (1) provide a better understanding of the relative importance of mechanisms for afterheat removal and (2) enable independent evaluation of GA analyses. Reference 11 describes the ORECA modeling and solution techniques and examples of transient calculations. The report shows predictions of several types of accident transients to be in good agreement with the results of RECA calculations. The relatively small computation times required for ORECA make it convenient to use for model and parameter sensitivity studies.

The original development of ORECA included three versions, one each for FSV, the 3000-MW(t) LHTGR, and the intermediate-size commercial HTGR [2000 MW(t)]. Most of the later development of ORECA was concentrated on the FSV version in response to specific licensing requirements.

The original ORECA code for the FSV reactor had the following characteristics and capabilities:

1. Each of the 37 refueling regions and 18 side-reflector blocks was represented by 8 axial nodes (a total of 440 nodes for the core simulation). Of each group of 8 axial nodes, 6 represented the active core, 1 the top reflector, and 1 the bottom reflector and core-support blocks.
2. Coolant heat transfer coefficient calculations included the effects of changing flow regimes (turbulent-transition-laminar) and of helium conductivity and viscosity variations with temperature.
3. Inputs included total helium flow, inlet pressure and inlet temperature vs time, and the total reactor power vs time following a typical scram curve. Input values of axial and radial power-peaking factors are assumed constant throughout the run.
4. The flow calculated for each channel depends on friction losses, acceleration losses, buoyancy effects, and empirically derived entrance and exit loss and orifice pressure drop coefficients. Calculated friction losses depend on the flow regime.
5. Channel flows are calculated at each time step by an iterative scheme that determines the overall core pressure drop and the proper total flow rate (within a specified error). Reverse flows are also accommodated.
6. To investigate the consequences of postulated LOFC accidents, a model was developed for predicting the maximum temperatures of the prestressed concrete reactor vessel (PCRV) thermal-barrier cover plates in the core-inlet plenum. In such accidents, the cover plates would be exposed to hot plumes emanating from those refueling regions in which reverse (or upward) flow occurs. This model was incorporated into the stand-alone ORECA code as an optional feature.
7. A most helpful review of the ORECA code was done by P. G. Kroeger,[3] who noted some problem areas and made several good suggestions for improvement. Appropriate coding changes were made as a result. The major improvement made was to the internode heat-conduction algorithm. Previously, the effective thermal conductance between a given node and its neighbors was assumed to be a function of only that node's temperature. In the corrected version, each conductance term is based on the average temperature of the given node and its neighbor. Corrections were made to the acceleration pressure-drop term and the orifice coefficient temperature multiplier for cases in which flow is reversed. Improvements were also made in the algorithms that account for the ratios of conduction areas and directional conductivity relationships for axial vs radial conduction between refueling region blocks.
8. Because a more detailed analysis was required of the core-support region, the axial node structure was changed to include two nodes for the bottom reflector and one for the core-support block, for a total of 10 axial nodes per region.

9. The ORECA code was used in investigations of the FSV oscillation problem[14] (see also Sect. 4.5.3 of this report). To confirm the GA jaws model for explaining the large oscillations in core-region outlet temperatures, a special version of ORECA was developed to back calculate the region flow changes that would be required to produce the observed outlet temperature perturbations. Some problems were encountered with noise in the resulting region flow calculations because the measured temperature output is a lagged function of flow caused by both the region heat capacity and the thermocouple assembly. For both of these effects, the lower the flow, the slower the response and, thus, the more sensitive the back-calculated flow is to temperature changes. To minimize the flow fluctuations, the temperature data were fitted and smoothed using a third-order Lagrangian interpolator. The code was also modified so that measured and calculated steam-generator helium inlet temperatures could be compared. This feature also allowed an alternative calculation of the apparent changes in effective cold bypass-flow fraction for the 12 core sectors corresponding to each of the 12 steam-generator modules.

Further development work planned for the ORECA code includes continued model enhancement based on comparisons with FSV transient data and special test results. Specific improvements planned include incorporation of a detailed PCRV liner cooling system simulation for modeling long-term LOFC accidents and a reworking of the input data structure so that a wider variety of transient types can be run by changing only input data. Presently, some coding changes are usually required.

Applications of the ORECA code to specific FSV licensing problems are described in Sect. 4.5, and a description of model verification efforts is given in Sect. 4.6.1.

4.3.4 The BLAST code[8]

The BLAST code simulates the HTGR reheater-steam generator module with a multinode, fixed-boundary homogeneous flow model. The original version of BLAST solved the time-dependent conservation of energy, mass, and momentum equations for the helium and water flow and the conservation-of-energy equation for the tubing with an implicit integration technique. The code calculates helium and water temperature, pressure, and flow rate, as well as tube bulk and wall temperatures for a user-specified computational mesh of up to 20 nodes each for the water, helium, and tube sides.

The BLAST code has great flexibility for not only arbitrary computational geometry but also a variety of options for two-phase heat transfer correlations.

A copy of BLAST was sent to RWTUV in 1979. The RWTUV's plans involve the use of BLAST to analyze transients as required by the German licensing process. The RWTUV also completed the BLAST model for the THTR reheater and steam generator. This model will be incorporated into a plant simulation of the thorium high-temperature reactor (THTR) by the Institut fur Reaktorentwicklung at Kernforschungsanlage Jülich (KFA). The RWTUV also completed a model of the AVR steam generator with BLAST in preparation for BLAST verification activities that would compare the code with measured data from AVR transients.

The RWTUV has modified BLAST significantly and has made these modifications available to ORNL. These versions include several improvements such as (1) a modification allowing a restart after the initial steady-state calculation or during the transient, (2) a modification in the subroutine for computing two-phase flow multipliers to extend the pressure range, (3) a more rapid method of solving the system of differential equations, and (4) a separate version of BLAST with input and output in SI units. Current plans are to use these versions in comparing BLAST predictions with measured data obtained from FSV for selected transients. These modifications provide very significant improvements in BLAST capability and represent considerable effort by RWTUV.

Besides the additional code verification work planned for BLAST, more development work will be done to improve the initial steady-state convergence routine and the helium and two-phase flow and heat transfer algorithms.

Discussions of comparisons made between BLAST and other code predictions and BLAST vs experimental data are given in Sects. 4.2 and 4.6.3.

Additional research in the area of steam generator modeling was done under a subcontract with the University of Tennessee.[15] Activities included development of (1) a few-node linear model, (2) a very detailed nonlinear model, and (3) model verification test plans.

4.3.5 The ORTURB code[16]

The electrical turbine and feedwater heater simulation ORTURB evolved from a model developed by Delene.[17] Delene's coding had been modified for use in ORTAP; however, this modification used an excessive amount of computer time. Work was initiated to revise the computer simulation for the dynamic behavior of the steam turbine components so that sufficient accuracy would be obtained with minimum computer time. The present version of ORTURB[16] is the result. This model uses governing equations similar to those presented for Delene's original steam turbine model, but it uses a modeling and iteration scheme that minimizes floating point exponentiation, the major consumer of computer time. The ORTURB code was developed and debugged independently of ORTAP and therefore required the inclusion of FORTRAN statements to provide the necessary transient plant-behavior parameters. In ORTAP, these parameters are supplied by the appropriate plant component simulator.

The ORTURB code is divided into three main parts: (1) a driver subroutine that provides plant operating parameters and conditions; (2) turbine subroutines to calculate the pressure-flow balance of high-, intermediate-, and low-pressure turbines; and (3) feedwater heater subroutines. This feedwater heater model is substantially modified from the feedwater heater model developed by Delene.[17] The necessary steam property subroutines were taken from this same reference and modified slightly.

The ORTURB code has two important limitations: (1) the turbine shaft rotates at rated speed (3600 rpm) and (2) no energy or mass storage is accounted for in the high-, intermediate-, and low-pressure turbines. These limitations exclude the use of ORTURB during a turbine transient such as start-up from zero power or very light turbine flows.

Turbine transients that represented normal and upset turbine conditions were simulated with ORTURB. The calculated results were appropriate in all cases, and when plant data were available, calculated turbine conditions agreed quite well with the data.

4.3.6 The FLODIS code[18]

The FLODIS code is a 3-D thermal-flow analysis code that represents the FSV core in a very detailed fashion. Its present use to ORNL is mainly for comparison with ORECA, which is much less expensive to use. The original funding for FLODIS was provided by the Division of Technical Review, NRC.

The FLODIS code approximates each of the 31 seven-column refueling regions with 4 rectangular subregions and each of the 6 five-column refueling regions with 3 rectangular subregions. Therefore, FLODIS can calculate both the intraregional and interregional flow distribution. The core is divided into 20 axial nodes.

The original version of FLODIS was substantially revised by D. D. Paul but not completed before he left ORNL. The revision has been debugged, and preliminary documentation has been prepared.[19] Corrections for heat transfer coefficients and reactor afterheat have been modified to agree with those used in ORECA to compare results from both codes.

A paper entitled "Thermal-Flow Performance of the Fort St. Vrain High-Temperature Gas-Cooled Reactor Core During Two Design-Basis Accidents" was presented at the American Nuclear Society/American Society of Mechanical Engineers topical meeting on Nuclear Reactor Thermal-Hydraulics held on October 6–8, 1980, in Saratoga, New York. The sensitivity of the interregional core flow distribution caused by the position of the flow control orifices was investigated with FLODIS. The calculated results showed that the effect of temperature on helium viscosity is an important factor in the interregional and intraregional flow redistribution subsequent to both accidents.

4.4 Analysis of FSV Reactivity Insertion Accidents for Comparison with FSAR Results

The FSV FSAR[20] discusses possible sources of reactivity accidents together with safeguards designed to prevent or limit the severity of these accidents. The discussion includes cases in which the accidents are followed by postulated failures in the safety system. Sources of reactivity accidents investigated by GA are:

1. excessive removal of control poison;
2. loss of fission product poisons;
3. rearrangement of core components, including fuel-loading accidents;
4. introduction of steam into the core; and
5. sudden decrease in reactor temperature.

General Atomic Company concluded that the maximum reactivity insertion and the largest credible reactivity insertion rate would result from the accidental withdrawal of a control rod pair. The rod pair withdrawal accident is postulated to occur as a result of a malfunction of a control rod drive or an operator error resulting in unintentional removal of control poison from the core. The plant control system design allows withdrawal of only one rod pair at a time; therefore, the withdrawal of several rod pairs simultaneously is not considered credible. Furthermore, the control rod drive penetration closures are designed so that a control rod ejection accident is not considered a credible occurrence. Because the rod pair withdrawal accident leads to greater reactivity insertions and insertion rates than others listed previously, detailed analyses of the other reactivity accidents are not presented in the FSAR.

To provide an independent assessment of the FSAR results and at the same time obtain comparisons for benchmark accident cases, the ORTAP code was used to analyze postulated rod pair withdrawal accidents for a variety of assumptions about reactor parameters and plant protection system response.[21]

An overall plant simulation was required for analysis of this accident to account for variations in core-inlet temperature and flow caused by the response of other nuclear supply system components and control system action. An overall plant simulation was also necessary to determine when certain scram signals, such as the scram on high reheat steam temperature, would be initiated.

Several important safety-related conclusions were reached as a result of the analysis:

1. The results indicated that the most severe temperature transient would occur if the accident were initiated from full power at the beginning of the equilibrium cycle.
2. In a rod pair withdrawal accident, several plant control and safety systems must be inoperative for fuel temperatures to exceed 1600°C. According to GA, fuel remaining below this temperature will not fail.
3. Considerably more time is required to reach the high reheat steam temperature scram setpoint than the 140%-power scram setpoint. If the scram is initiated by the rise in reheat steam temperature, transients initiated by the accidental withdrawal of a maximum-worth rod pair from the half-inserted position result in earlier scrams and less severe temperature conditions than transients initiated by the accidental withdrawal of the same rod pair from the fully inserted position.
4. For the reference case, the use of lower core-heat conductance values results in the prediction of more severe core temperatures.
5. The action of the plant control system in reducing helium flow in an attempt to maintain the main steam temperature delays the reheat steam temperature scram because it reduces the rate of increase of the reheat steam temperature. This delay results in more severe peak temperatures in the core but less severe temperatures in the reheater and steam generator.

Comparisons of the results of the ORTAP and FSAR analyses indicated generally very good agreement.

For the reference case, no significant difference existed between the calculated time of scram and GA's reported value. The maximum fuel and gas temperature predictions were also in good agreement. Sample results of the study are shown in Table 1.

Table 1. Sample results of the FSV rod pair withdrawal accident study

Parameters	Reference case RHST scram[a]		140%-power scram		Rods initially half inserted power scram	Core conductances increased 50% RHST scram
	ORTAP	FSAR	ORTAP	FSAR		
Time at scram initiation, s	102.5	105	39.2		76.2	98.3
Power level at scram initiation, %	282		140		210	291
Maximum core average fuel temperature, °C	1195	1225	861	870	1098	1147
Maximum mixed mean core outlet temperature, °C	994	1062	804	796	927	993
Region experiencing withdrawal: peak fuel centerline temperature, °C	3057	2870	1137	1183	2082	2981
Peak region outlet temperature, °C	1654	1650	862	914	1224	1604

[a] A reheat steam temperature (RHST) scram occurs when the measured reheat temperature exceeds the rated temperature by 42°C (75°F).

4.5 FSV Reactor Licensing Support

4.5.1 NRR technical support for initial rise-to-power licensing questions

A variety of relatively small tasks were done by project personnel to assist with licensing tasks both for FSV and for generic licensing questions in the 1974–77 period. These tasks include:

1. review of various GA LTRs,
2. Summit reactor licensing review,
3. study of the effects of derating the Pelton-wheel circulator drive performance during a DBDA,
4. extension of the ORECA models to calculate heat-up of concrete near the PCRV liner during an extended LOFC event,
5. studies relating to the FSV moisture ingress problem,
6. review of predicted vs measured performance of the primary circulator,
7. response to a Division of System Safety, NRC request to provide information on anticipated accident transients,
8. a study of the FSV 2-loop dump incident,
9. a study of the control rod overheating problem, and
10. a review of emergency depressurization procedures.

4.5.2 FSAR design-basis accident reevaluation

In April 1978, ORNL was requested by NRC to provide independent calculations of both DBDA and LOFC accidents to assist in evaluating a 100%-power operating license application for FSV. Reactor operating parameters were supplied by GA, and worst-case equilibrium core conditions were assumed for the reference analyses. Some cases were also run using initial core nuclear parameters to evaluate consequences of a postulated LOFC accident occurring before installation of planned pumping capacity that would be used in the subsequent cooldown.

4.5.2.1 Design-basis depressurization accident. The reference case DBDA analysis using ORECA assumed a 5-min delay in the start-up of the emergency cooling system and then assumed the availability of only one loop (two of the four circulators).

Of primary concern with the DBDA is the overheating of the steel liners and ducting to the steam generators. This outweighs the concern for fuel damage because peak predicted fuel temperatures are well below 1600°C. Calculation of the steel liner temperatures is complicated primarily by uncertainties in the estimates of "streaking factors," which relate the maximum gas temperatures impinging on the liners to the maximum refueling-region gas exit temperatures. Using a conservative value of the streaking factor derived from GA air model tests, the predictions indicated that the 1093°C damage limit would not be reached. Figure 3 shows some results of the reference-case ORECA predictions. Sensitivity studies were also performed to determine the effects of various assumptions on peak temperatures, and no surprises were encountered. Table 2, which gives comparisons of ORECA results with those generated by GA (RECA3

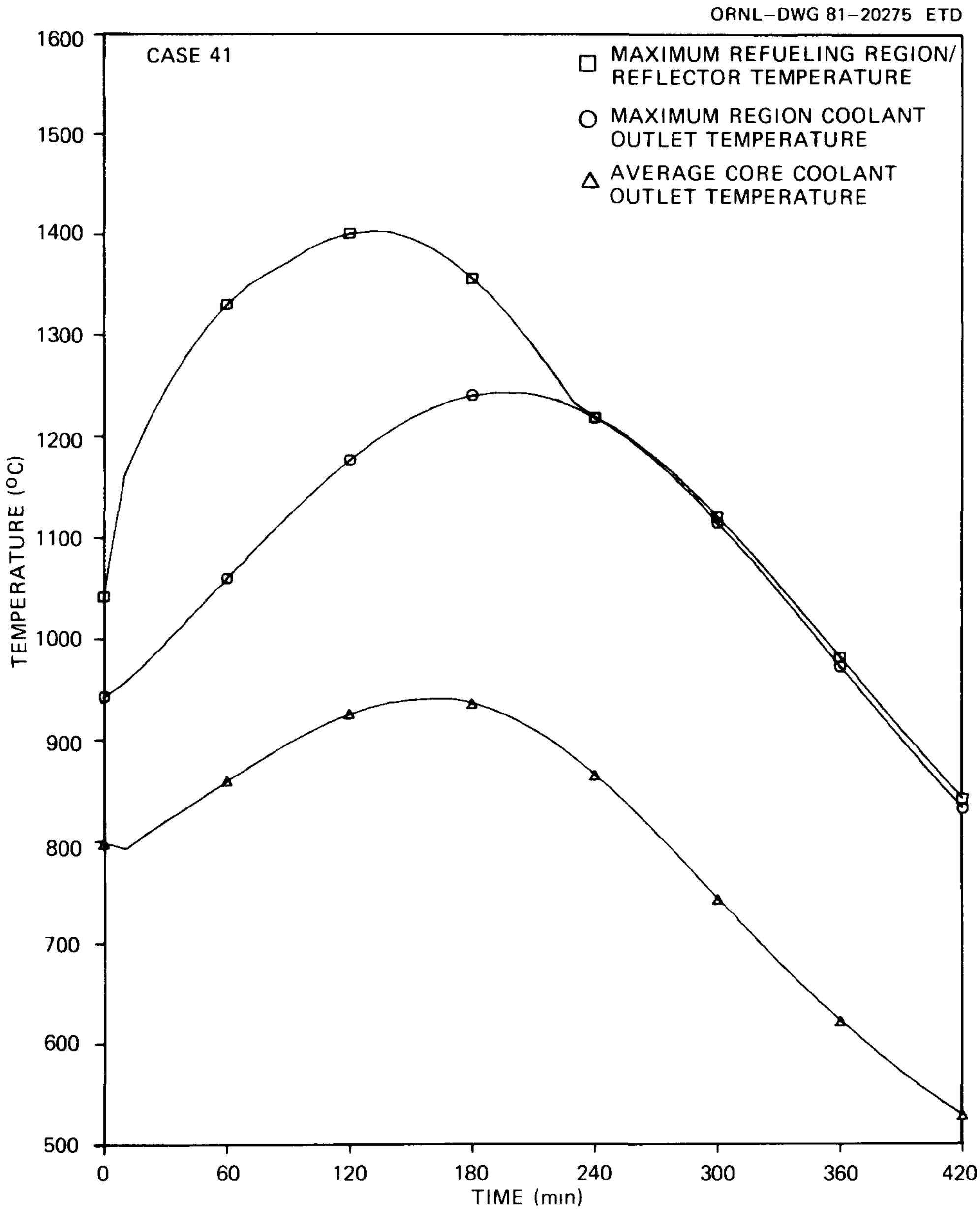

Fig. 3. Sample results of a postulated FSV reactor DBDA using the ORECA code.

Table 2. Results of FSV DBDA study

Reference DBDA	GA/RECA3	ORNL/ORECA
Peak fuel temperature, °C	~1427	1403
Maximum average core outlet temperature, °C	~927	940
Maximum refueling-region outlet temperature, °C	~1288	1243

code), indicates generally good agreement. Note that FSAR afterheat equations were used in both analyses.

4.5.2.2 LOFC - firewater cooldown (FWCD) accidents. The LOFC accident calculations centered on the question of how much time the operators would actually need to start up the emergency cooling system following a postulated design-basis earthquake. In this case, an LOFC is followed by use of the earthquake-proof firewater system to provide both the motive force for the circulator's Pelton-wheel drives and the cooling water for the steam generators. Calculations were done both for worst-case initial and equilibrium cores, the latter giving the higher peak temperatures.

The main concern during the LOFC period is the ability of the carbon steel upper-plenum thermal-barrier cover plates to withstand the heat from the hot plumes that emanate from refueling regions experiencing reverse flows. Calculations were done for postulated delays of up to 2 h in initiation of the FWCD system. Subsequent analyses by NRC indicated that the assumption of a 1.5-h delay was sufficiently conservative. A major uncertainty in the model is the effective plume heat transfer coefficient (h-plume), and a detailed model of the plumes and cover plates was added to the ORECA code. Depending on h-plume assumptions, the calculations indicated that some of the cover plates would be likely to exceed failure limit temperatures for extensive LOFC periods.

The major problem following initiation of the FWCD system is, like the DBDA, possible damage to the steam generator inlet ducts. As before, using the GA-derived streaking factor, the damage limit was not exceeded for any of the cases analyzed.

Sample results of an LOFC/FWCD ORECA calculation are shown in Fig. 4, and comparisons with some GA results for the case of no delay in initiation of FWCD (as in the FSAR) are shown in Table 3. As in the case of the DBDA analyses, the comparisons are generally very good.

4.5.2.3 Evaluation of bounding consequences of other FSAR-postulated accidents. The FSAR's major accident scenarios include consideration of reactivity insertion accidents, loss-of-heat removal capability accidents, and chemical reaction accidents from air or water ingress.

A thorough study was made of the reactivity accidents,[21] and a follow-up study[22] showed that while fuel damage would probably not occur, some verification data analyses should be done to check the BLOOST/CORTAP-type code accuracy. Another point that was raised by the analysis was that if the main steam-temperature control system were in manual, significant overheating of the steam generators could result [providing that several levels of defense failed (see Sect. 4.4)].

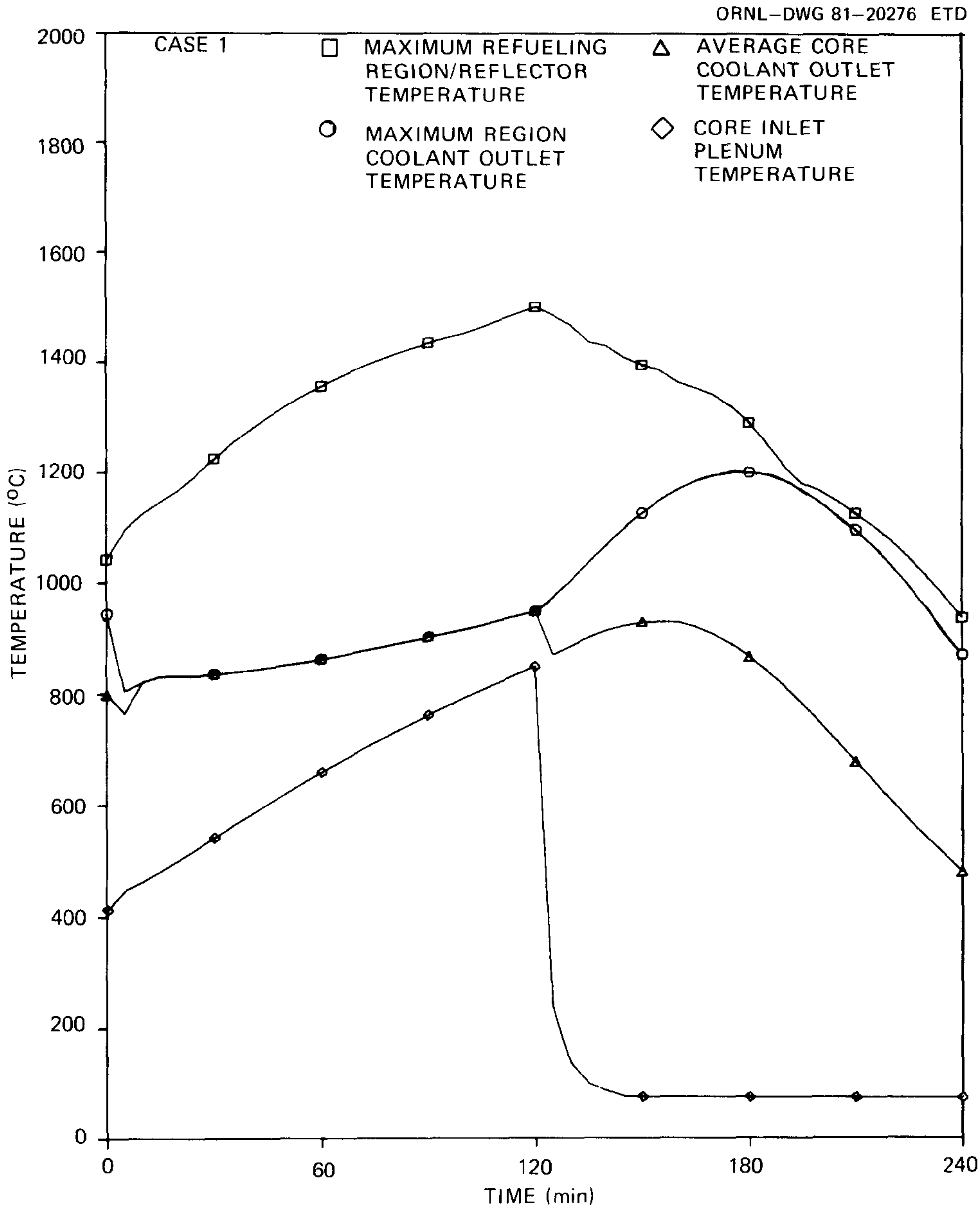

Fig. 4. Sample results of a postulated FSV reactor LOFC-FWCD accident using the ORECA code.

The failure of heat removal capability was investigated by J. P. Sanders,[23] who addressed the problems of heat removal from the core in the event of (1) depressurization and total LOFC or (2) a permanent LOFC without depressurization. Such scenarios result in eventual deterioration of PCRV integrity, which would be especially unattractive if the reactor has not been successfully depressurized.

In summary, the scenarios that go beyond the severities of the FSAR assumptions are those that would be addressed in the severe accident sequence analysis (SASA) studies noted in Sect. 5.

Table 3. Results of FSV LOFC/FWCD study

Zero-delay FWCD	GA/RECA3	ORNL/ORECA
Equilibrium core		
Maximum average core outlet temperature, °C	~829	821
Maximum refueling region outlet temperature, °C	~1038	1023
Initial core		
Maximum average core outlet temperature, °C	~816	804
Maximum refueling-region outlet temperature, °C	~1038	1038

4.5.3 Investigations of the FSV oscillation problem

At the request of NRC, project personnel became involved in assessing the unique safety and licensing questions that came up because of the oscillations observed in the FSV core temperature and neutron detector measurements. At certain operating conditions, fluctuations in individual neutron channels would be as much as ±5%, and helium temperature excursions from individual refueling regions and to steam generator modules would be as large as 200 and 100°F, respectively. Fluctuations could be initiated at higher levels of power and core flow resistance (which can be adjusted by the refueling-region orifice valves) and terminated by reducing the power and flow. The fluctuations had a random character, but a dominant periodicity of ~10 min was observed in many instances.

The oscillations first occurred on Oct. 31, 1977, and they have since occurred at power levels ranging from 30 to 68%. Tests run during both Cycles 1 and 2 have resulted in just over 100 h spent in an oscillation mode. Subsequent installation of region constraint devices (RCDs) to the top layer to plenum elements during the October–November 1979 outage appears to have been successful in stopping the oscillations for power levels up to 70%; however, more tests at higher powers are planned for 1981.

The ORNL involvement in the oscillation problem included (1) technical support during the initial stages of analysis, (2) assessment of related safety analyses and test program plans by GA and PSC, (3) noise analyses of various core instrumentation signals, (4) review of the special in-core instrumentation [Instrumented Control Rod Drives (ICRDs)], and (5) safety assessments of the proposed fixes, including the RCDs. Program involvement is continuing through the post-RCD tests and 70 to 100% power tests.

The program's major analytical effort was an evaluation of the jaws theory, which postulated that periodic tilting of fuel element blocks near the top of the core will open up alternate flow paths through the jaws so formed (Fig. 5) and that the resulting additional flow through a region's coolant channels could cause a substantial and rapid decrease in its outlet temperature.[14]

General Atomic postulated that the largest temperature fluctuation observed in the most significant oscillation event (November 4, 1978, ~0410 h) could have been caused by a 38% change in the region flow and that such a flow change was compatible with the jaws model.

To confirm the jaws model, the ORECA code[11] was modified to take the 37 measured refueling-region outlet temperatures, the reactor power, the core differential pressure (ΔP), and the core inlet temperature vs time as input. The 37 region flows (and effective orifice coefficient changes equivalent to the jawing) vs time would then be back calculated as required to produce the observed outlet temperature perturbations.

Refueling-region (measured) outlet temperatures for the 20-min period beginning at 0410 h on November 4 are shown in Fig. 6 for regions 12 and 13, the two most active regions. Region 12 temperature increases ~90°C (160°F), while that for adjacent region 13 drops ~100°C (180°F). Figure 6 also shows the results of the modified ORECA calculation of the measured

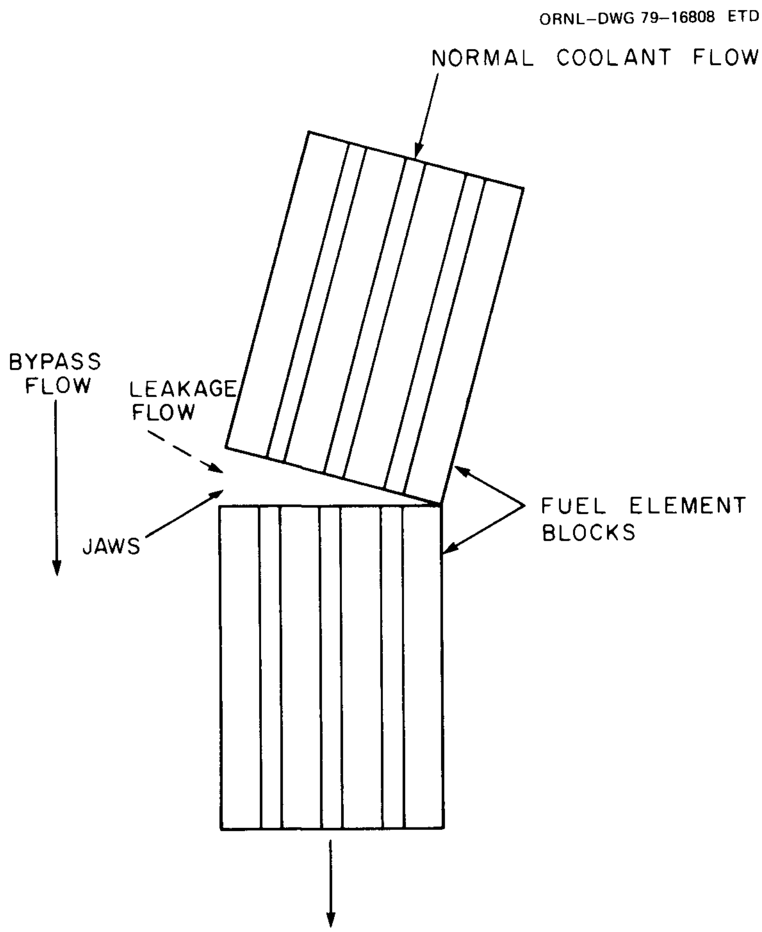

Fig. 5. Jaws theory model.

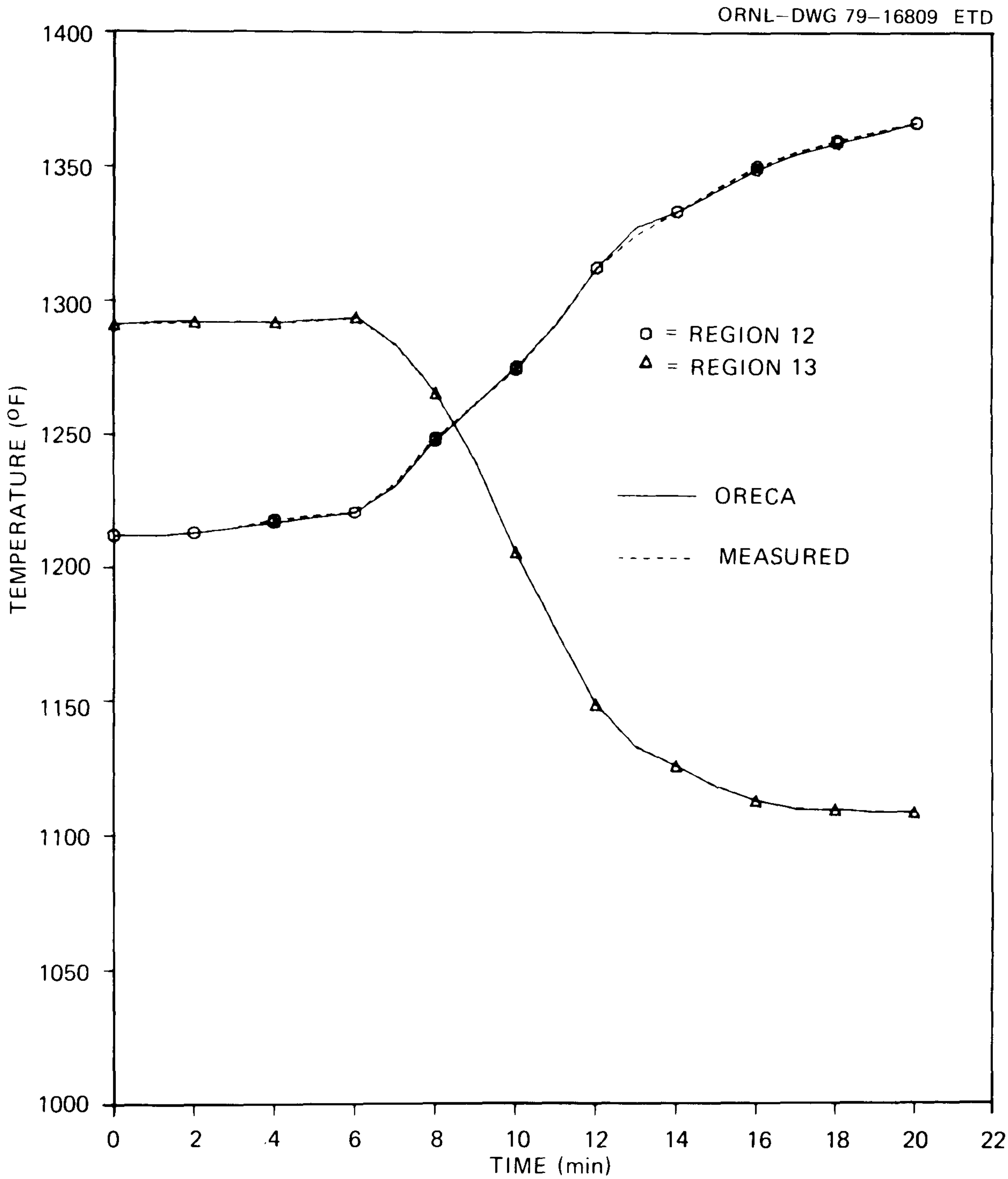

Fig. 6. Calculated vs measured FSV core outlet temperature from regions 12 and 13 during oscillation transient. Nov. 4, 1978, 0410 h.

outlet temperature transients (dashed curves), indicating that the back calculations of those region flows succeeded in matching the observed temperatures. The back-calculated flows for regions 12 and 13 are shown in Fig. 7. A flow increase of ~70 to 80% was needed to cause the rapid drop in temperature for region 13, and a flow decrease of ~50 to 75% was required to give the large temperature increase in region 12. Note that the erratic nature of the region 12 flow is probably partially caused by the sensitivity of the calculation to abrupt changes in temperature slope. The measured temperature "output" is a lagged function of flow caused by both the region heat capacity and the thermocouple assembly time constants. For both of these effects, the lower the flow, the slower the response and, thus, the more sensitive the back-calculated flow is to temperature changes. To minimize the flow fluctuations, the temperature data were fitted and smoothed using a third-order Lagrangian interpolator. Even so, the flows shown are probably more erratic than would actually be required to produce the observed temperature swings. Some of the other regions with much smaller outlet temperature fluctuations also had very large computed flow transients.

In summary, sufficient uncertainties were noted in the accuracy of the data, the ORECA models, and the method used to back calculate region flow transients to make any conclusion for this analysis tentative. The ORECA-derived flow variations of ~$\pm$75%, however, are probably larger than could reasonably be expected from jaw-type leakage; thus, the assumption that other phenomena are also influencing the behavior of the region outlet thermocouples is suggested. The conclusion was made that the most likely phenomenon is the biased readings caused by bypass-flow leakage into the thermocouple assembly sleeve. Measurements support the contention that this effect would be more pronouned in regions near the side reflector.

4.6 Comparisons of Code Predictions with Experimental Data

4.6.1 ORECA[11,24]

Most of the experimental verification efforts to date have been comparisons of ORECA code predictions with data from FSV scrams. Four different scrams from power levels between 30 and 50% have been used. General Atomic Company has supplied the necessary input data, including circulator inlet temperature, core flow, pressure, and power (afterheat) vs time after the scram. The initial conditions are the 37 measured refueling-region outlet temperatures and estimates of each region's power-peaking factor. Comparisons are then made of the computed and measured outlet temperature transients for all regions. Although no unique combination of models and parameters exists that will produce a good fit to the data (and therefore no guarantee that an optimized model is valid), optimization schemes do suggest areas that may need improvement.

The original ORECA best-estimate calculations of the scram tests were usually in reasonably good agreement with the data; however, there were some significant differences. In Fig. 8, a typical ORECA prediction of measured region outlet gas temperature (TGO) is seen to be low after 20 to

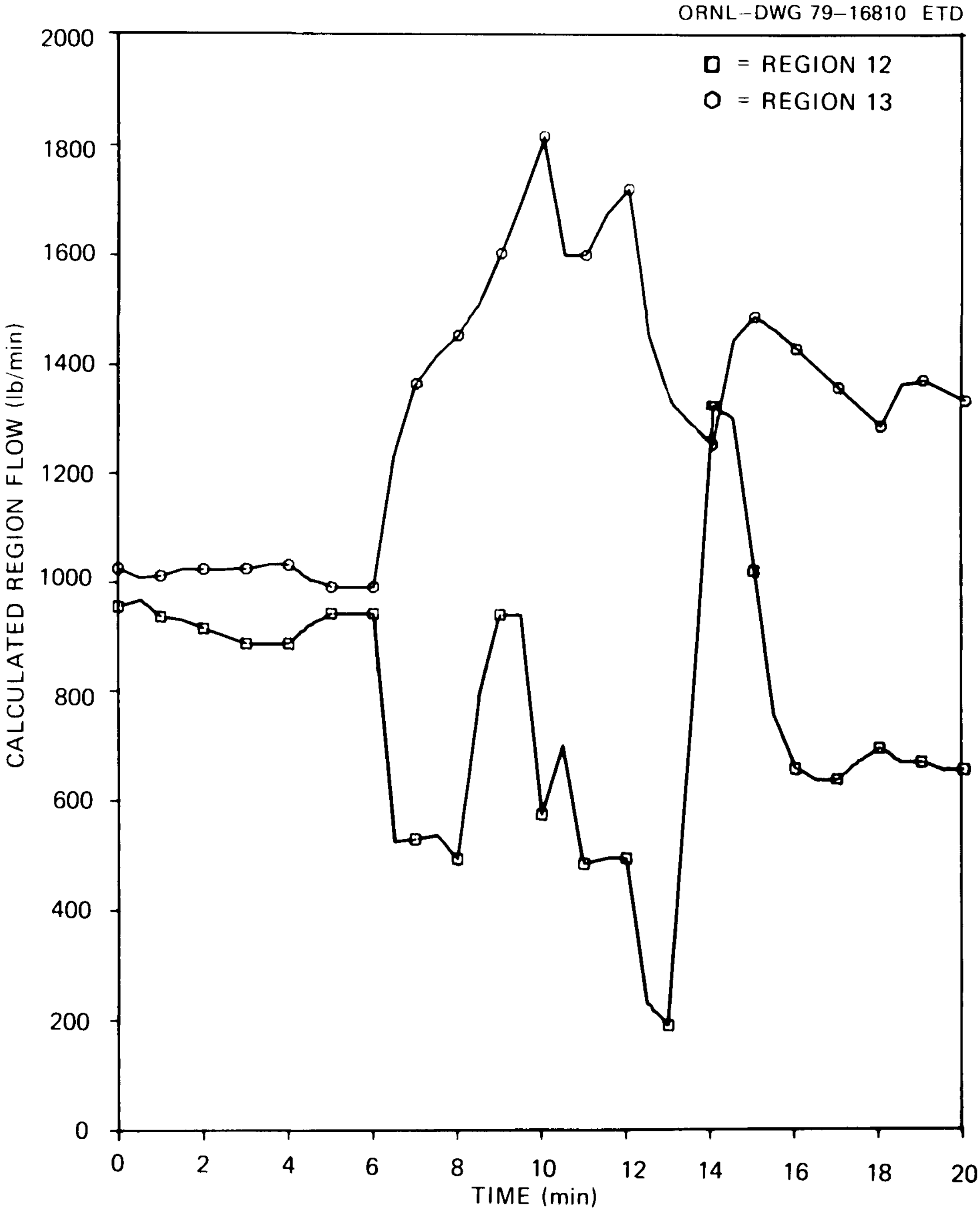

Fig. 7. Calculated FSV core flows for regions 12 and 13 during oscillation transient. Nov. 4, 1978, 0410 h.

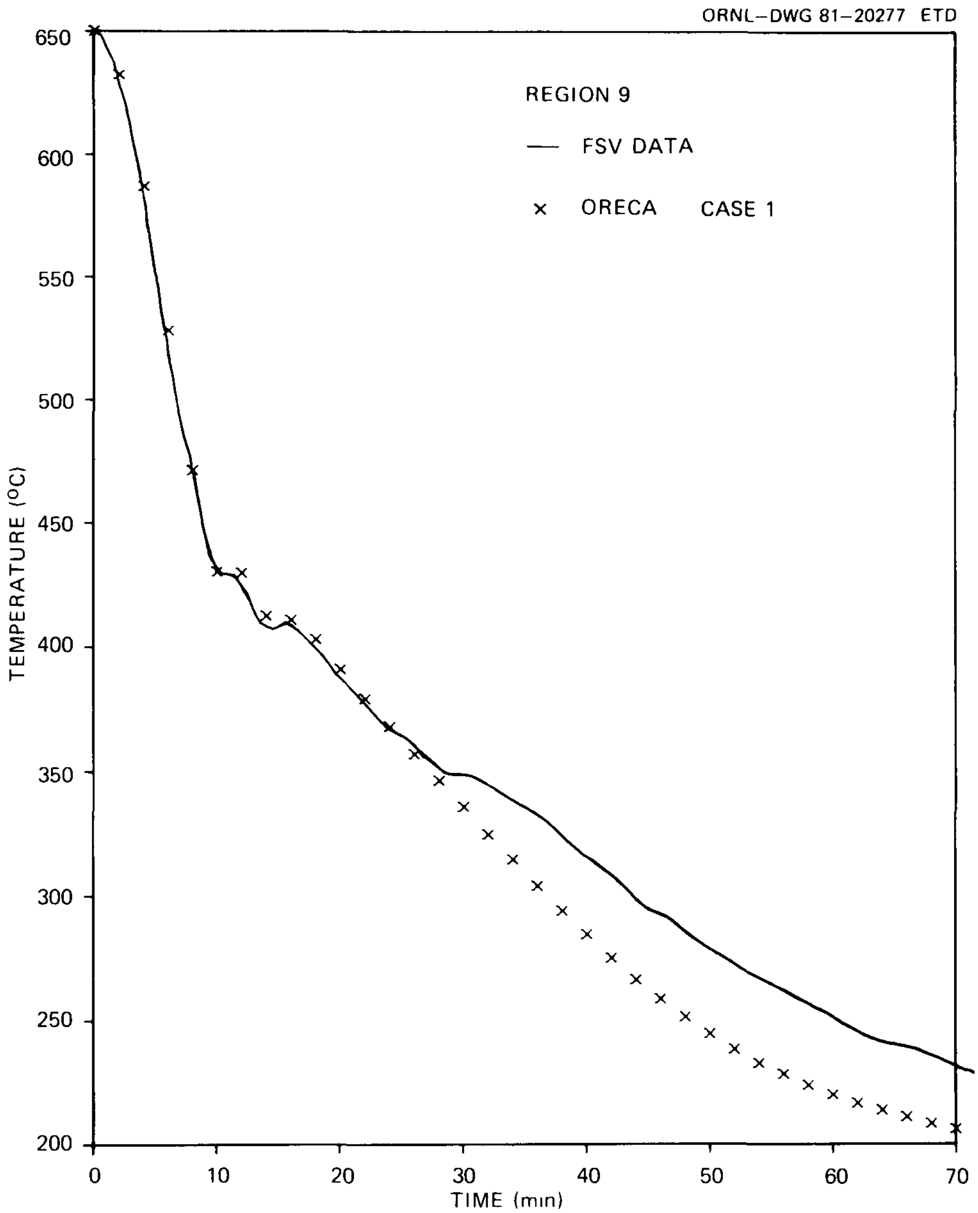

Fig. 8. FSV scram test of Aug. 6, 1977, from 28% power — comparison of best-estimate reference ORECA code predictions of measured gas outlet temperature from region 9 vs plant data.

30 min following the scram. This has been attributed both to an overestimate of core flow (or underestimate of bypass flows) and to deficiencies in the original dynamic model of the TGO thermocouples.

The core bypass flows include those through gaps in the refueling region and side reflector block as well as flows bypassing the core barrel entirely. None of these is directly measurable.

The TGO thermocouples are in large graphite sleeves and have time constants of ~2 min at rated conditions. Several versions of the thermocouple model have been used subsequently, with the most significant improvement being the addition of T^4 radiation effects. To properly account for these effects, however, ORECA had to be revised to model the lower part of the core-support blocks separately (rather than lumped with the lower reflector as before), because the support blocks cool down much more slowly after a scram than does the rest of the core.

Other modifications required to produce a good fit were adjustments in the assumed peaking factors for many of the regions, especially those near the outer ring, and adjustments of the assumed temperature rise of the helium between circulator inlet (measured data) and the core inlet. This rise is caused by both the heat of compression from the circulators and by heat transfer to structures between the circulators and the core inlet plenum.

An optimization code was used to find the ORECA parameters that give the best least-squares fit to the data. The optimization code uses the differences in the responses generated by ORECA for several selected parameter-variation cases. By comparing these responses with the FSV data, the optimization code computes a set of optimized parameters. This set is limited by what are judged to be reasonable uncertainty ranges. After these parameter adjustments are incorporated into the ORECA code, the agreement is generally excellent, with typical results shown in Fig. 9. One discrepancy still remaining (especially in the higher-power tests) is a distinct difference in the shape of the curve for several regions adjacent to the side reflector (Fig. 10). These differences are thought to be a result of interactions with the side reflectors that are not yet explicitly modeled. Work on the optimization is still in progress.

4.6.2 CORTAP[7,25]

The CORTAP code calculates the reactor power and representative fuel, moderator, and coolant temperatures. Inputs are (1) coolant temperature, (2) flow and pressure at the core inlet, and (3) control-rod reactivity. The CORTAP code was used here as an independent calculation of core response rather than as a subroutine of the plantwide simulation ORTAP.

The CORTAP code calculation of reactor power transients resulting from control-rod movement was verified by comparison with operating data taken during control-rod influence tests at FSV. Each of the tests consisted of a brief control-rod insertion or withdrawal followed by constant control-rod position throughout the remainder of the test. The two transients used were a 6-s withdrawal of region 1 control rods and a nominal 24-s insertion of region 6 rods, where the control-rod speed was 2.5 cm/s in both cases. Control-rod worths were such that a 15-cm change in the region 1 control-rod position changed reactivity more than the 61-cm change in region 6 rod position.

The reactor power transient was recorded for each of the six neutron detectors. Data from the six detectors were averaged for comparison with the CORTAP calculation of reactor power response. No attempt was made to compare region outlet temperatures with CORTAP calculations, because the time response of these thermocouples is not known with sufficient accuracy.

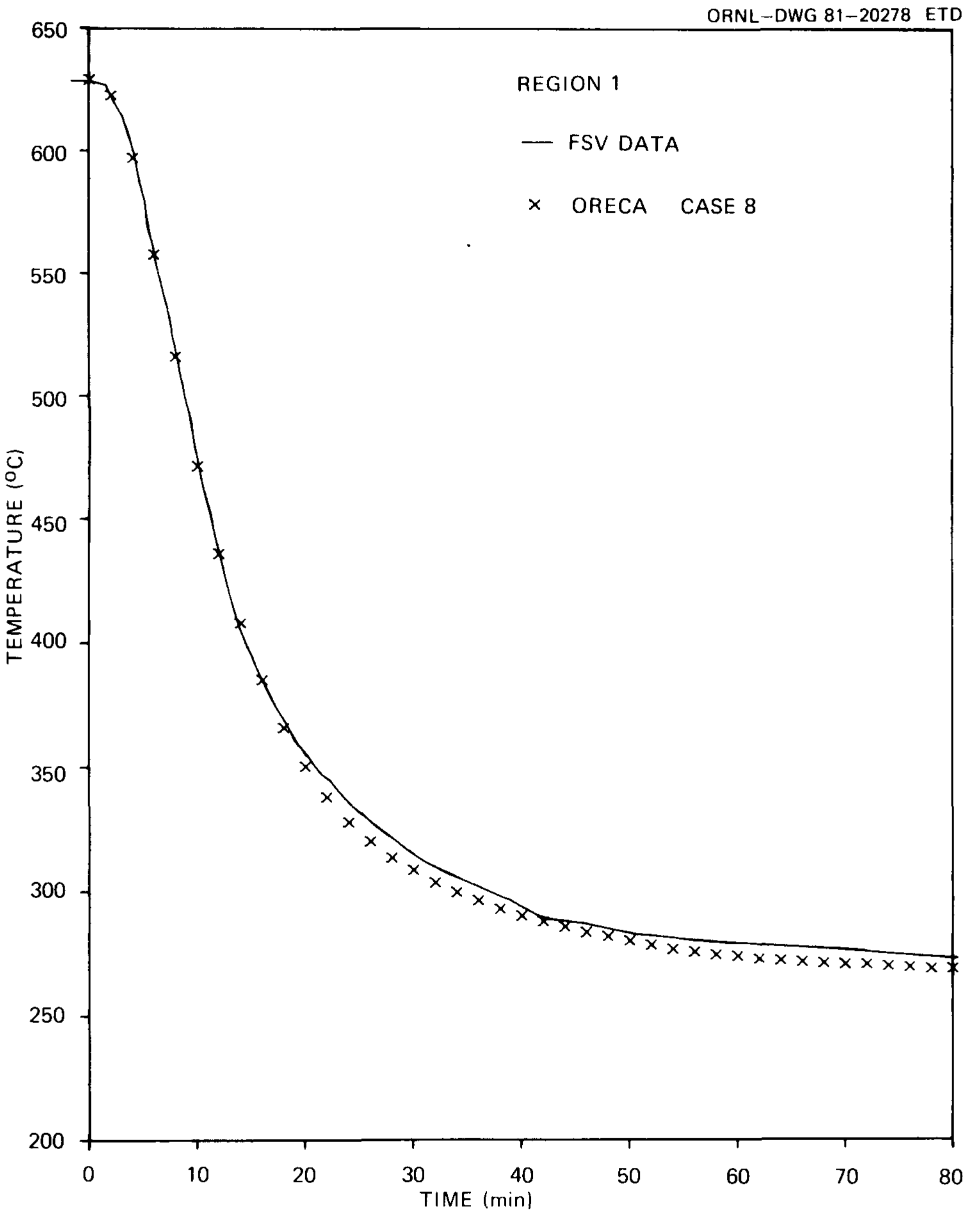

Fig. 9. FSV scram test of Oct. 25, 1977, from 40% power — comparison of optimized ORECA code predictions of measured gas outlet temperature from region 1 vs plant data.

The important CORTAP input parameters that were used are summarized in Ref. 25. To get a good comparison with the data (January 1978), beginning-of-cycle (BOC) inital-core kinetics data were used. Core flow was calculated from steady-state core inlet and outlet temperatures reported in the test data. Because control-rod travel was short in comparison with the 4.5-m active core length, constant differential rod worth was assumed for input to CORTAP. Core flow and inlet temperature were assumed to remain constant throughout each test.

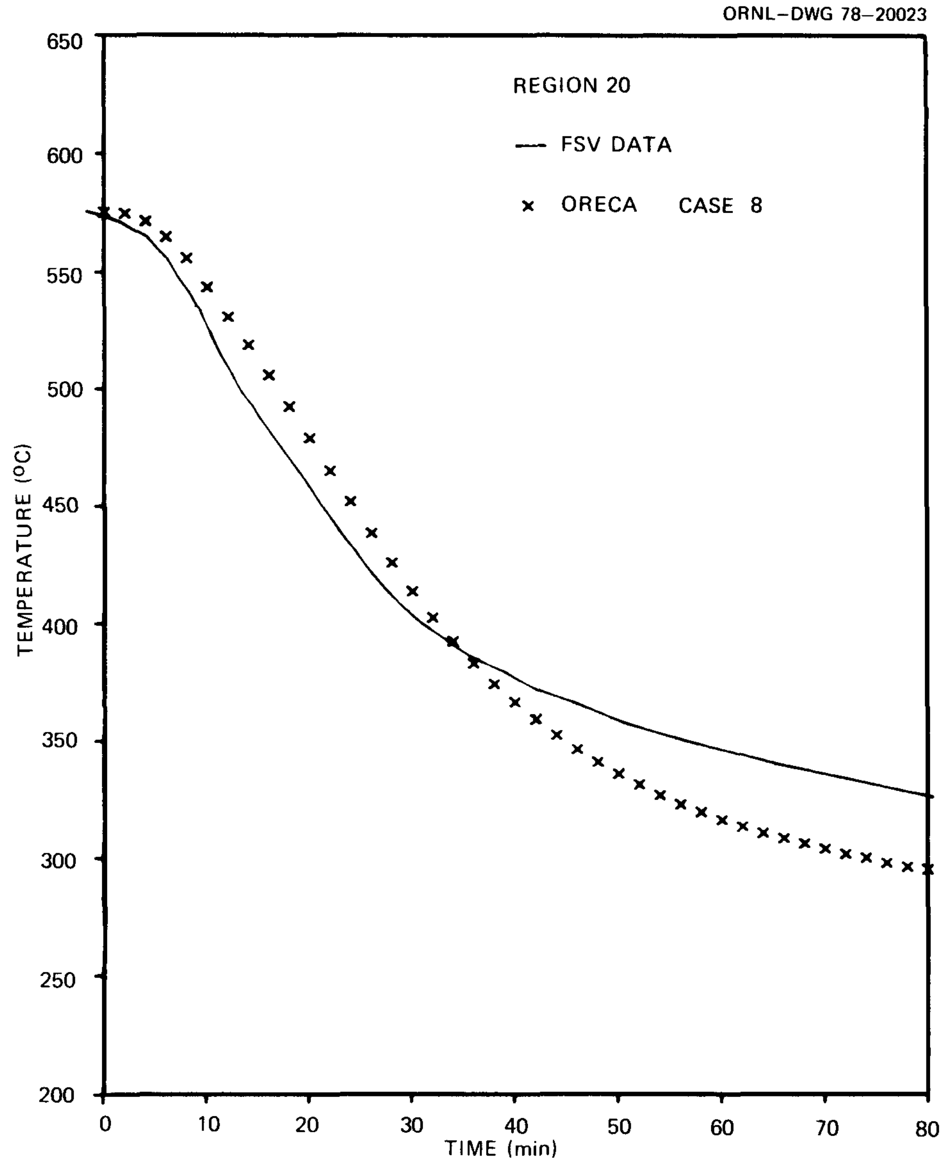

Fig. 10. FSV scram test of Oct. 25, 1977, from 40% power — comparison of optimized ORECA code predictions of measured gas outlet temperature from region 20 vs plant data.

To set up the CORTAP code to calculate the transient, the total reactivity added by the control rods must be known; however, this was not reported for either test. By means of a sensitivity study, we found that the reactivity addition could be inferred from the data, because while the control rods are in motion, the sensitivity of the power response to control-rod reactivity is five to ten times greater than to any of the other parameters. After the control rods stop moving, the other parameters become more important. The fuel and moderator specific heats have an effect on reactor power during the dynamic part of the transient but no effect on the final steady-state power level reached. The Doppler coefficient and coolant flow have a significant effect on both the dynamic portion and final steady-state power change. If the other parameters are known reasonably well, then the control-rod reactivity can be inferred by simply matching experimental and calculated responses during the first 6 s (or during the first ~24 s for the rod insertion transient). This procedure was used to calculate control-rod reactivity for the comparisons reported.

Results of the CORTAP calculation of reactor power and the corresponding plant data are shown in Fig. 11 for the 15-cm control-rod withdrawal test. The agreement between experiment and prediction is good, both for the transient portion and for predicting the final steady-state power change. This was true of the insertion test as well. For both rod insertion and withdrawal, CORTAP calculated that the power change at ~90 s would slightly undershoot the final steady-state power change. The plant data show very little tendency to undershoot the equilibrium power level. The same data were used by GA for validation of the BLOOST code,[26] with the result that BLOOST predicted an undershoot very similar to the CORTAP calculations. The reason for the undershoot phenomenon remains unexplained; however, because the magnitude of the discrepancy is small, no concern that the model limitation has any safety significance exists.

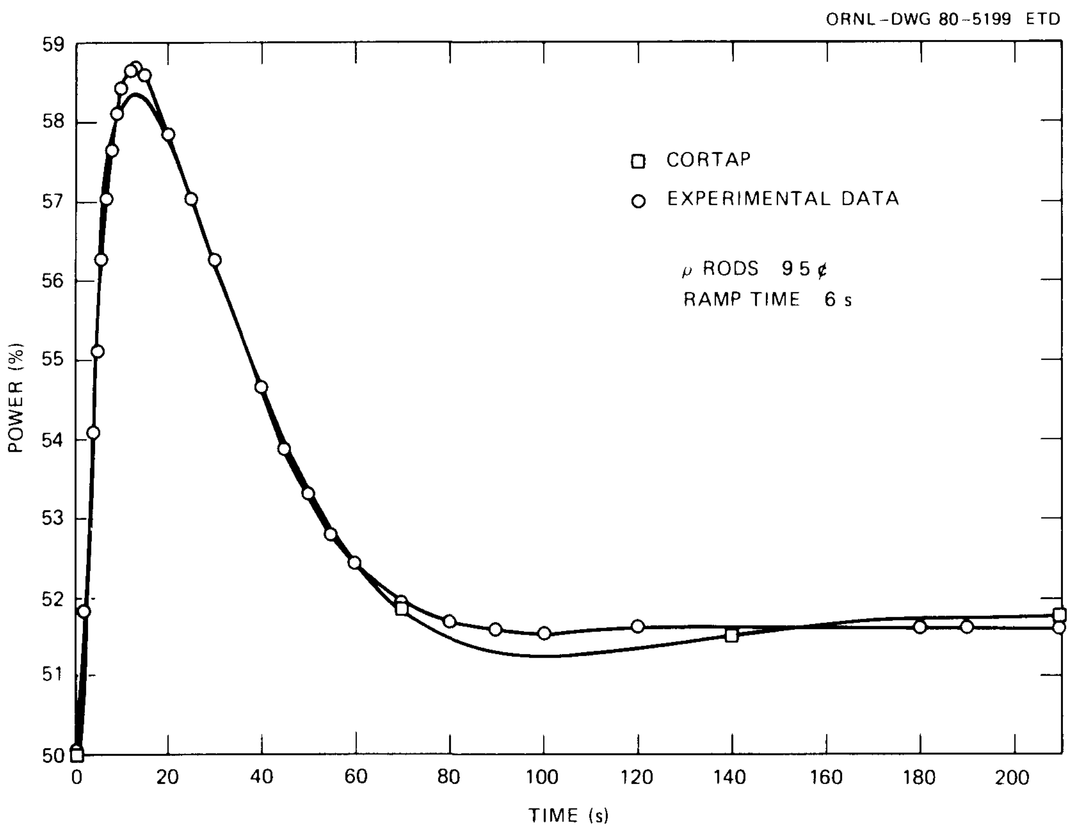

Fig. 11. Reactor power response following 9.5¢ reactivity insertion.

4.6.3 BLAST*

Comparison of the BLAST code predictions with measured plant data is proceeding in two areas. The BLAST predictions were compared with FSV transient data obtained during an oscillation test transient that caused a rapid decrease in helium inlet temperature to a steam generator module in loop 1. Also, comparison of BLAST predictions with data obtained from the AVR steam generator were made in 1980. (More information on this effort is given in Sect. 4.7.)

The FSV oscillation test transient of November 4, 1978, involved a large [~44°C (80°F)] rapid decrease in helium inlet temperature to one of the 12 steam generator modules (B-1-1 in loop 1) and resulted in a drop in main steam subheader temperature of ~68°C (122°F) for this module. The purpose of this analysis was to make a direct comparison of BLAST predictions with the measured plant response. Analysis of the transient is not complete, but the following is intended to indicate the nature of the preliminary results obtained to date.

For the oscillation transient, GA provided measured data for reactor power, loop 1 feedwater flow, total core helium flow, module B-1-1 helium inlet temperature, loop 1 inlet and outlet reheat-steam temperature, and module B-1-1 subheader outlet steam temperature vs time. Some inputs required for BLAST (e.g., feedwater temperature and pressure, reheat-steam flow and pressure, and main steam pressure) were not provided and have been estimated by interpolating from steady-state conditions expected at 25 and 100% power. Additionally, loop 1 feedwater flow and reheat-steam flow were assumed to be distributed equally among the six steam generator modules in loop 1.

The model used in analyzing this transient uses ten water nodes, ten tube nodes, and seven helium nodes.

Figure 12 shows a comparison of the change in exit steam temperature from the main superheated steam section (superheater II) as computed by BLAST vs the change in measured subheader outlet temperature for module B-1-1.* The flow-dependent lag associated with the steam temperature measurement has been incorporated into the BLAST prediction. As is shown, the calculated drop in steam temperature resulting from the 44°C drop in helium inlet temperature was ~67°C (121°F) and compares very well with measured steam temperature drop. However, the measured data showed that the initial drop in steam temperature was followed by a 25°C (45°F) increase in steam temperature, which is not reflected in the BLAST calculations. The increase may result from differences between actual conditions (e.g., transient feedwater flow, feedwater inlet temperature, and main steam pressure for module B-1-1) and estimated input to BLAST (the estimated values used in BLAST for these parameters were assumed to remain constant during the transient). Furthermore, while the computed temperature changes during the transient compare fairly well with measured data, an offset occurs at the time of transient initiation between the computed superheater-II exit temperature and the measured subheader temperature of ~42°C (75°F) (based on current inputs to BLAST, some of which are assumed values), with the computed temperature higher than the measured value.

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*Superheater II exit temperature is not measured for module B-1-1.

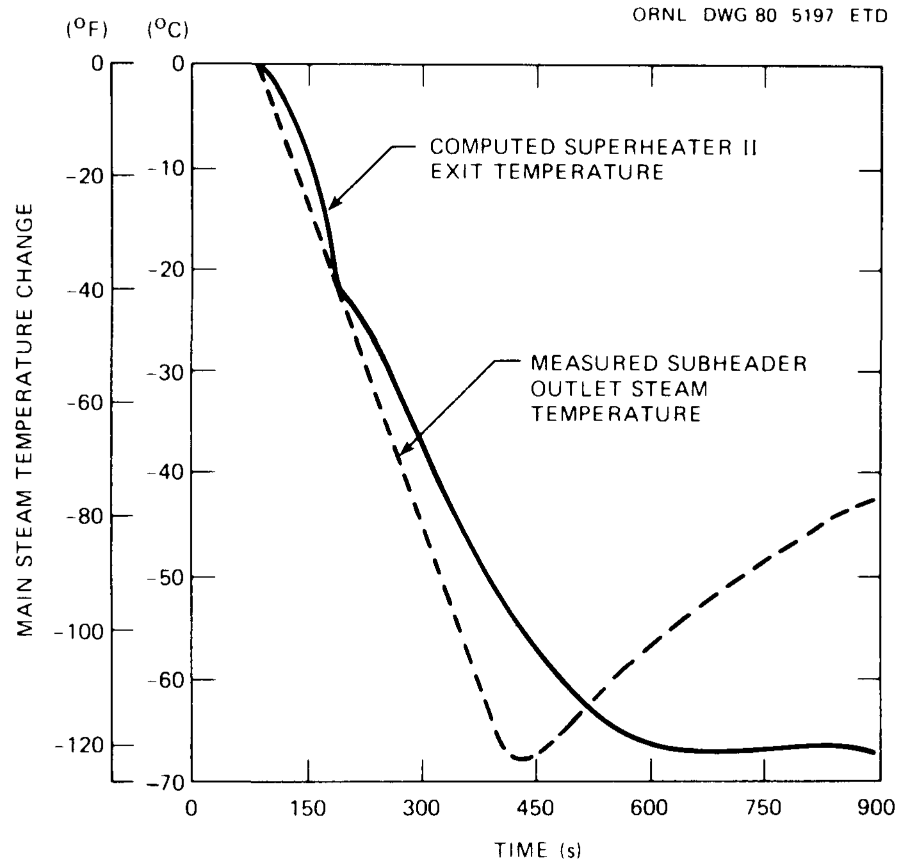

Fig. 12. Comparison of computed superheater II exit steam temperature with measured subheader outlet steam temperature for module B-1-1.

Reasons for this offset have not been explored in depth to date. A significant portion of this offset could possibly be attributed to regenerative heating, which causes the subheader-outlet main steam temperatures to be lower than superheater II exit temperatures. Another likely source of error is the omission of reheater attemperation flow from the model. Adding attemperation would decrease the calculated superheater helium inlet temperature and thus lower the steam exit temperature.

The first step in examining the reasons for the differences between the preliminary BLAST computation and measured data for the oscillation transient is to attempt to obtain data for those input values that have had to be estimated for these initial BLAST calculations. Specifically, data are needed for (1) feedwater inlet temperature, pressure, and flow (to module B-1-1), (2) main steam pressure, and (3) reheat-steam flow and pressure during the transient.

4.7 HTGR Safety Information Foreign Exchange Programs

The ORNL program has had the good fortune to benefit significantly from two foreign exchange programs. A guest scientist from Japan,

M. Hatta, was sponsored by his home company (IHI) and Japan Atomic Energy Research Institute for a one-year visit to ORNL to work on the RSR-sponsored program (July 1977 to June 1978). Mr. Hatta brought with him a wealth of experience relevant to HTGR safety and made numerous and substantial contributions to our program. Other significant benefits resulted from attendance (by S. J. Ball) at the HTGR Safety Technology Conference[27] in Fuji, Japan, in November 1978, and the subsequent information exchange that resulted from the trip.

The RSR program has also had a number of useful exchanges with the West Germans. Initially, RWTUV acquired the BLAST code and ORTAP for use in their licensing studies of the THTR. Since then, ORNL and RWTUV have had much fruitful correspondence, and several RWTUV-developed improvements have been incorporated into BLAST. More recently, one of the ORNL program staff members received a one-year assignment to KFA in Jülich to work on HTGR safety problems.

Regular correspondence and information exchanges have also been set up with British [Central Electricity Generating Board (CEGB)] as well as Japanese and German HTGR researchers.

4.8 Investigations of Overheating of FSV Upper-Plenum Cover Plates During Extended LOFC Accidents

A major uncertainty in the prediction of the consequences of sustained 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. 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 its full pressure of ~4.8 MPa (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 with a maximum temperature limit of ~815°C (1500°F). Simulations of 2-h LOFC accidents have indicated that this temperature limit might be exceeded, depending largely on the assumptions of plume heat transfer.

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

The testing with a scaled low-temperature, low-pressure air model of the actual FSV upper plenum during an LOFC would have to be based on the assumption that certain scaling laws would apply. The object, then, was to scale the model such that both the Reynolds and Grashof numbers would be roughly equivalent. Such a comparison is shown in Table 4.

Preliminary scoping tests of a scaled air model were run initially. Subsequently, a small-scale plume experiment was built and tested (Fig. 13). The purpose of the experiment was to demonstrate the applicability of the assumed scaling laws and iron out procedural and measurement problems that would be encountered in a full-scale model.

Table 4. Comparison of HTGR plume and air model parameters

Parameters	HTGR helium plume	Model air plume
Temperature, °C (°F)	1093 (2000)	93 (200)
Pressure, MPa (psia)	4.8 (700)	0.1 (14.7)
Density, kg/m^2 (lb_m/ft^3)	1.76 (0.11)	0.96 (0.06)
Viscosity, kg/ms ($lb_m/ft \cdot h$)	5.8×10^{-5} (0.14)	2.2×10^{-5} (0.053)
Mass flow, kg/s (lb_m/min)	0.11 (15)	0.045 (5.9)
Equivalent orifice diameter, m (in.)	0.43 (17)	0.43 (17)
Velocity, m/s (fps)	0.46 (1.5)	0.32 (1.04)
Reynolds No.	6000	6000
Grashof No.	7×10^8	4.3×10^8

In the test, a rotameter measures air flow through a heater assembly and into a nozzle with adjustable diameters, which directs the heated air up to a thin flat plate mounted above and representing an FSV upper-plenum cover plate. Mounted in the plate is a thin metal can insulated on the top and sides and partially filled with water that serves as a calorimeter to measure the rate at which heat from the plume is transferred to the plate area. The height of the plume, as well as its (nozzle) temperature and flow, are all adjustable. Material considerations limit the nozzle temperature to ~315°C (600°F). A curtain is used to shield the plume from extraneous drafts.

An on-line computer is used to monitor temperatures of the plume, calorimeter water, and ambient air and to calculate the heat transfer rate, heat transfer coefficients (Nusselt numbers), Reynolds number at the nozzle, the Grashof number, and other data that indicate the statistical accuracy (confidence level for a prescribed accuracy or error tolerance). The program written to acquire and analyze the data is set up to control the duration of the run based on the run statistics.

An example set of results of several runs on the reverse-flow plume experiment with a nominal nozzle exit temperature of 205°C (400°F) and a 0.305-m (12-in.) plume height are shown in Fig. 14. Of these runs, the two that are classified as variant runs, 6 and 8, are characterized by both higher-than-average values of Nusset and higher plume-top temperatures. This variation is postulated to be caused by a basic plume instability problem; that is, under similar conditions of nozzle exit temperature and flow, plume height, and ambient temperature, quasi-stable plumes could be established with significantly different top temperatures and

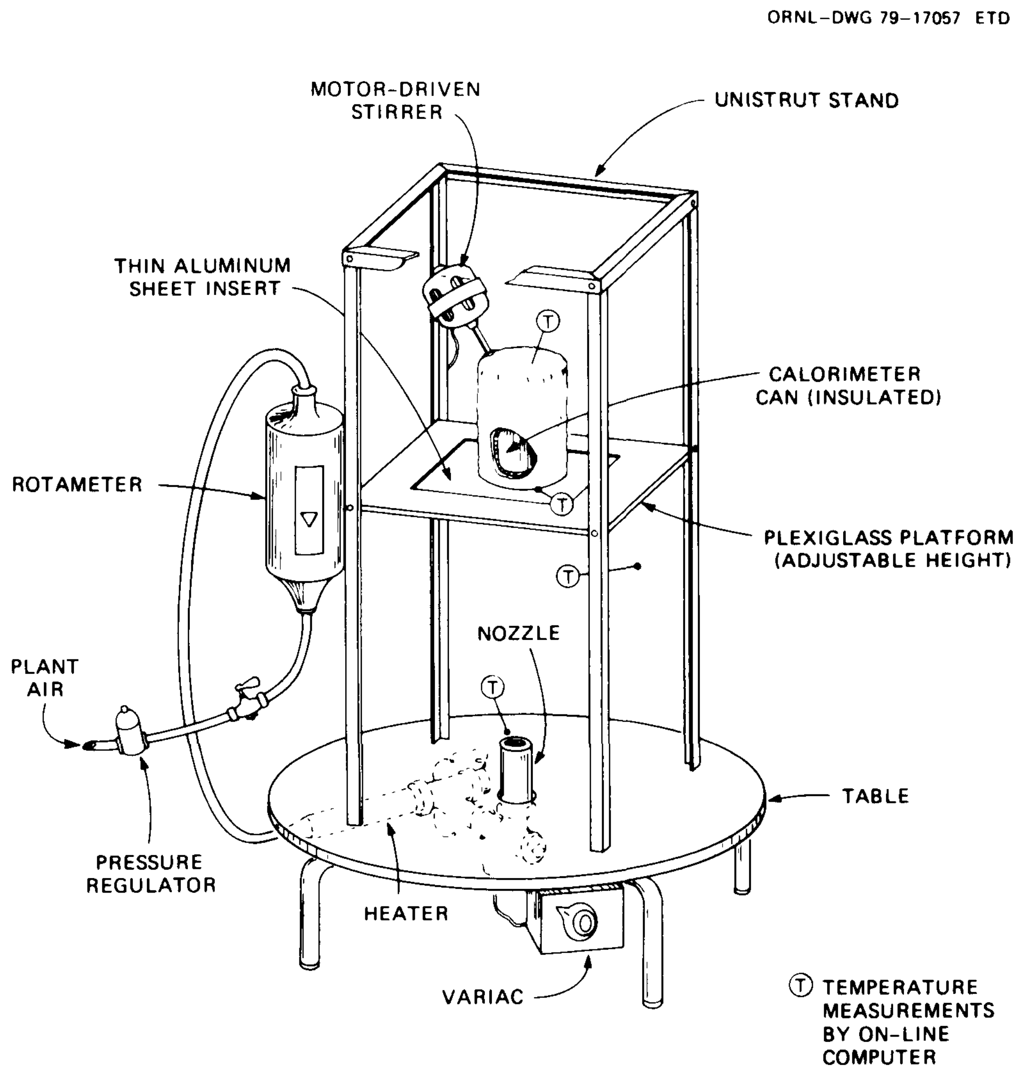

Fig. 13. Small-scale heated-plume experiment assembly.

rates of heat transfer to the calorimeter. The instability is believed to be related to the extent of the mixing of the hot plume with the surrounding ambient air.

Several other more qualitative tests showed that exposing the plume to nearly still ambient air by opening the protective curtain would essentially destroy the plume, even for relatively short plume heights and typical accident-case nozzle velocities. Therefore, we may postualate that more subtle plume disturbances could occur even within the protective shroud of the curtain, which could result in the formation of quasi-stable plumes. (In all cases, the plume-top temperatures were quite stable over the data-taking period, ~1/2 h.)

General Atomic Company postulated that the plumes established following an LOFC accident at FSV would be well mixed; therefore, a mixed mean

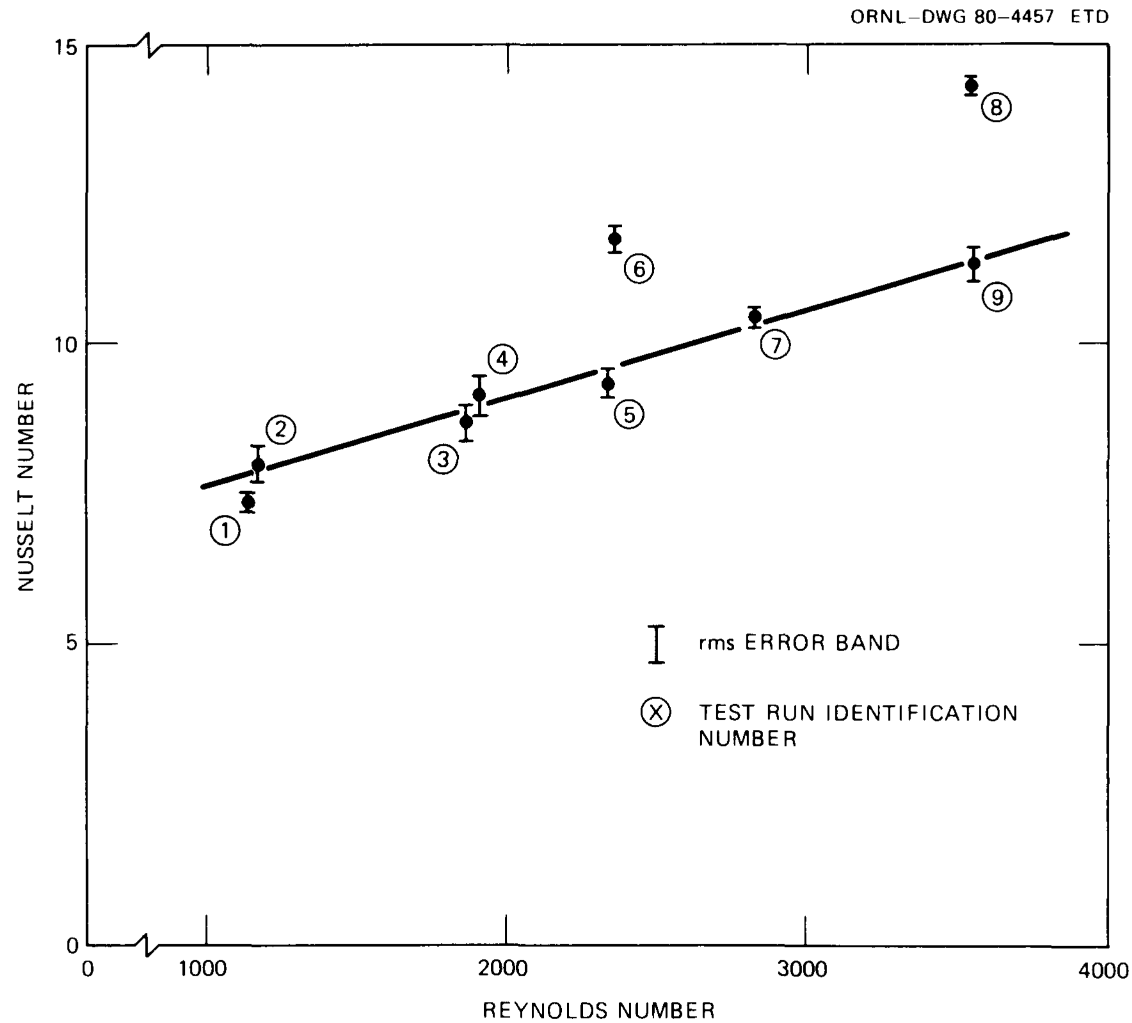

Fig. 14. FSV plume tests with nozzle temperature of ~205°C (~400°F) and plume height of 0.305 m (12 in.), November 1979.

upper-plenum gas temperature, rather than individual hot plume temperatures, should be used in the prediction of cover plate temperatures.

The conclusions drawn from these observations were that (1) the mixing of adjacent plumes in the FSV upper plenum, where plumes would have no protective shrouds, is expected to be large and thus help to substantiate GA's claim; and (2) these apparent plume instabilities would make difficult the conclusive demonstration of the applicability of the similarity laws for Reynolds and Grashof scaling. Such a demonstration depended on obtaining the same values of Nusselt for given values of Reynolds, Grashof, and H/D, the plume height-to-nozzle-diameter ratio; typically, the tests showed no such relationships. The model testing was therefore abandoned, and the conclusion was that upper-plenum plume instabilities would make cover plate damage during the postulated LOFC unlikely.

4.9 Calculations of Postulated FSV Reactor LOFC/FWCD Accidents for Core Thermal Stress Evaluations

Results of previous analyses of the postulated design-basis earthquake LOFC accident followed by an FWCD were used by LANL to calculate thermal stresses in parts of the core-support structure. These stresses result from large temperature differences between adjacent refueling regions caused by preferential heating and cooling of the regions during the LOFC and FWCD phases of the accident. Recent LANL calculations of maximum stresses in the core-support blocks indicated that the stresses were large enough for some concern about possible crack formation and propagation in the support blocks. Several significant uncertainties, however, in both the thermal analyses (ORECA code) and the stress analyses (LANL calculations) required refinements in both analyses.

Sensitivity studies were completed using the ORECA code to determine the effect of changes in the reference-case assumptions. These studies indicated that a reduction of initial core power and an increase in the firewater booster pump output could significantly reduce the maximum region-to-region temperature difference. The reference-case analyses had been done assuming a 105% operating power level; thus, further analyses at the current FSV operating limit (~70%) were recommended as an interim means of alleviating concern about the safety of present operation, at least until more detailed analyses of the full-power case were available. Booster pump tests had shown that the FWCD flow estimates used in the analysis were conservative. The sensitivity studies also showed that the problem was less severe for high-flow-resistance cores because the redistribution of the coolant flow in the FWCD phase is less sensitive to the hot coolant-channel flow resistance.

Further refinements to the ORECA code in the core-support region were also found to be needed to provide more detailed information for LANL's stress analysis code inputs. Output was sent to LANL from a revised ORECA model that had ten axial nodes per refueling region — one for the upper reflector, one for each of the six fueled regions, two equal-sized nodes for the bottom reflector, and one for the core-support block. The revised ORECA version also provided outputs of heat flows into selected nodes via conduction and convection.

5. HTGR SAFETY GOALS

5.1 Near-Term Safety Goals

All HTGR near-term safety goals deal primarily with safety issues related to the operation and licensing of the FSV reactor. The most current and important of these is the technical support and assistance to the NRC NRR licensing staff on specific licensing questions. Recent assistance has concerned approval of PSC's request to operate FSV up to 100% power. Specifically, the program provided input on (1) questions of excessive thermal stresses in the core-support regions following a postulated design-basis earthquake accident, (2) GA RECA code verification analyses after installation of the region constraint devices, and (3) adequacy of the proposed PSC test plans (RT-500K) for oscillation testing between 70 and 100% power. Some follow-up work and post-test analyses are expected on all but the thermal stress issue.

Other near-term safety goals that we have proposed to NRC include the following:

1. FSV experiments and analyses to resolve questions about core bypass-flow fraction, hot streak phenomena, and region reverse flow behavior;
2. analyses and tests of interaction between the control and safety system;
3. tests of region-outlet thermocouple dynamic response (to resolve questions about identification of safety-related parameters from test data); and
4. consideration of SASA similar to those being addressed in the LWR safety programs.

5.2. Long-Term Safety Goals

The longer-term safety goals deal more with generic HTGR problems, which in turn depend on the status of the DOE-sponsored development programs. Presently, the status of DOE funding is uncertain, so setting priorities for longer-range safety work is difficult. Of the advanced HTGR concepts, those that generate process heat as well as electrical output appear most likely to succeed; therefore, new safety problems will relate to dealing with higher (process heat) temperatures, both for steady-state operation and accident transients, and with more complex system interaction and interdependence that result from having a process heat system coupled to the reactor primary coolant loops.

Specific areas in which longer-term HTGR safety work would be appropriate are:

1. environmental release — thermal discharge and radioactivity discharge;
2. safeguards [strategic nuclear materials (SNM) diversion)]; and
3. the risk to population from plant accidents — safety systems reliability, inherent plant safety features, normal plant equipment reliability, and the ability of containment to withstand and contain radioactivity from severe core damage.

For a given proposed advanced plant design, a good way to categorize potential safety issues is to make a list of accidents or types of accidents that should be considered. Calculations are then performed using computer models as necessary to determine if the proposed reference plant is likely to be able to meet the desired safety goals. On completion of this task, the researcher can specify ways in which the plant may be deficient for a given type of accident, and *also* what types of design modifications would be required to meet the safety goals.

The key to acceptability will be an assessment of the risk to the general population. This will require not only the calculation of the failure probabilities of plant equipment, but also deterministic calculation of the consequences of accidents resulting from plant failures. In a long-range program of safety research, risk assessment should be conducted for a variety of design alternatives to firmly establish regulatory requirements for future HTGR plants.

To be able to perform the risk assessment task mentioned above, the NRC safety researchers should have good analytical tools — computer programs that allow accurate prediction of the consequences of postulated accidents. Predicting (without further study) everything that will have to be calculated via computer models is impossible, but a logical assumption is that safety analysis research will have to concentrate on accidents that can have severe consequences.

The types of events that the computer codes will need to be able to handle include the following:

1. loss of forced circulation,
2. loss of coolant (leakage of helium or steam),
3. loss of process gas (if applicable),
4. excessive reactor core heat-up,
5. failure of normal and/or safety systems to operate as designed, and
6. water and other impurity ingress into the primary system.

The basic task of most of the system models will be calculation of pressures, temperatures, and flow rates by solution of the equations for conservation of mass, energy, and momentum. To complete the thermal-hydraulic calculation, in some accidents the ability to predict the chemical reactions that an HTGR core can undergo when exposed to unusual atmospheres will be necessary. For this example, however, use of only the basic conservation relations is insufficient, and the results of laboratory experimentation will have to be factored into the computer models. If applicable experimental data are not available, then one output of the safety assessment would be a recommendation that laboratory testing be performed to develop the needed data.

Examples of specific calculational capabilities required for the advanced HTGRs include the following:

1. very-high-temperature fuel behavior, including chemical behavior when exposed to air or water vapor, and radioactivity release;
2. PCRV cooling systems (for total loss of heat sink events);
3. post-accident radioactivity transport;
4. containment response (where applicable); and
5. process gas combustion or explosion.

In summary, development of computer models for HTGR safety research clearly should be coupled with a sincere effort to find out what major safety issues lie in the way of future licensing of HTGRs. This research will assure that the computer models developed will have all the capability required to help in the resolution of the important safety issues.

5.3 Possible Application of an ORNL Experimental Facility, the CFTL, to HTGR Safety Problems

The original design of the core flow test loop (CFTL) was developed for and funded by the DOE gas-cooled fast reactor (GCFR) program to test simulated fuel bundles in a high-temperature, high-pressure helium environment. Following the cancellation of the GCFR program in 1980, the CFTL construction was completed by DOE funding to make it available as an HTGR component flow test loop (also CFTL). The CFTL is a closed-circuit helium circulating system designed with a variable electrical heat supply and an air-cooled heat sink, capable of both steady-state and transient operation over a wide range of temperatures and pressures. It is also provided with a large and powerful on-line data acquisition and computer system, as well as a direct digital computer control system. A flow diagram of the CFTL is shown in Fig. 15. Potential HTGR safety applications presently under consideration of the CFTL include testing of primary-loop heat exchange

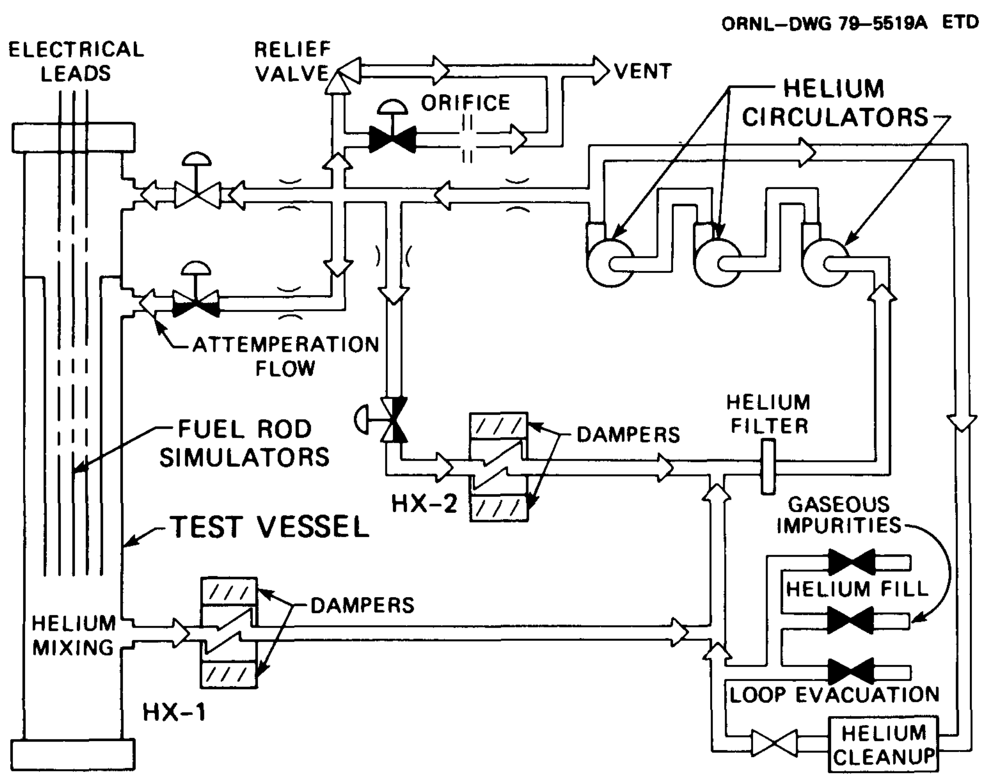

Fig. 15. Flow diagram — CFTL.

equipment and core cavity component performance. The major characteristics of the CFTL are as follows:

1. design pressure — 11.8 MPa (1715 psia);
2. operating pressure — ambient to 10.6 MPa (1540 psia);
3. power — 0 to 4 MW, controlled by 13 independent power supplies;
4. temperature — 260 to 600°C. Attemperation flow arrangement allows stainless steel melting (~1350°C) in the test section;
5. transient — full power to zero power in 1 s, full flow to zero flow in 1 s, fully pressure to approximately ambient in less than 1 min. All transients fully program controlled;
6. helium circulators — centrifugal type with gas-lubricated bearings, hermetically sealed;
7. working media — designed for helium with impurity control;
8. flow rate — 0 to 3.2 kg/s (circulators in series), to 9.6 kg/s (circulators in parallel);
9. flow measurement — wide range, high-accuracy vortex shedding flow meters;
10. data acquistion — high-speed (10 kHz), 640 channels (expandable), computer controlled;
11. location — Building 9201-3, Y-12 Plant, Oak Ridge, Tennessee 37830;
12. operator — Engineering Technology Division, ORNL;
13. sponsor — DOE, GCFR program (formerly), HTGR program (currently); and
14. availability — second half of 1981 (partial capability).

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