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ORNL/TM-5255

**Quarterly Progress Report on High-Temperature
Gas-Cooled Reactor Safety Studies
Sponsored by the NRC Division of
Reactor Safety Research for
October – December 1975**

J. P. Sanders

MASTER

OAK RIDGE NATIONAL LABORATORY

OPERATED BY UNION CARBIDE CORPORATION FOR THE ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

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Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161
Price: Printed Copy \$5.00; Microfiche \$2.25

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Contract No. W-7405-eng-26

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QUARTERLY PROGRESS REPORT ON HIGH-TEMPERATURE GAS-COOLED
REACTOR SAFETY STUDIES SPONSORED BY THE NRC DIVISION
OF REACTOR SAFETY RESEARCH FOR OCTOBER-DECEMBER 1975

J. P. Sanders, Manager

Work funded by the Nuclear Regulatory Commission
under Interagency Agreement 40-495-75

FEBRUARY 1976

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OAK RIDGE NATIONAL LABORATORY
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operated by
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CONTENTS

FOREWORD	v
SUMMARY	vii
1. HTGR SYSTEMS AND SAFETY ANALYSES	1
1.1 REHEATER AND STEAM GENERATOR MODEL DEVELOPMENT	1
1.2 DEVELOPMENT OF THE SYSTEM NUCLEAR STEAM SUPPLY SIMU- LATION CODE (ORTAP) FROM COMPONENT MODELS	1
1.3 CORE SIMULATION FOR EMERGENCY COOLING ANALYSES	10
1.4 CORE AUXILIARY COOLING SYSTEM MODEL DEVELOPMENT	10
1.5 UNIVERSITY OF TENNESSEE SUBCONTRACT	11
2. COMPUTER CODE IMPLEMENTATION	13
2.1 IMPLEMENTATION OF THE OXIDE-3 CODE ON THE IBM 360 COMPUTER	14
2.2 EVALUATION OF THE RECA PROGRAM	15
3. REPORT OF MEETING WITH NRC (RSR)	17
REFERENCES	18

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FOREWORD

This is the sixth in a series of quarterly progress reports on HTGR safety studies conducted at Oak Ridge National Laboratory (ORNL) under sponsorship of the Nuclear Regulatory Commission's Division of Reactor Safety Research.

Funding for these studies began in July 1974; the first quarterly progress report was issued as an internal program report (GCR-RSR: 74-4) and then subsequently reissued as ORNL-TM-4798. For the second, third, and fourth quarters of fiscal year 1974, the report was issued as Vol. IV of a combined quarterly progress report on all reactor safety programs at ORNL sponsored by the NRC Division of Reactor Safety Research. These reports are, in order, ORNL-TM-4805, ORNL-TM-4914, and ORNL-TM-5021. The progress report for the first quarter of fiscal year 1976 is ORNL-TM-5128.

This report covers work pertinent to high-temperature gas-cooled reactors for the quarter Oct. 1, 1975, through Dec. 31, 1975.

Copies of the report are available from the Technical Information Center, Energy Research and Development Administration, Oak Ridge, Tennessee 37830.

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SUMMARY

1. HTGR SYSTEMS AND SAFETY ANALYSES

Further development work was done on the component models and codes used in the overall NSS plant simulation (ORTAP) and on the code used to simulate emergency-cooling shutdown accidents (ORECA). Significant progress was also made in investigations of once-through steam generator models and in preparations for the dynamics tests planned for the Fort St. Vrain (FSV) reactor.

2. COMPUTER CODE IMPLEMENTATION

Evaluation of the RECA program is continuing, with effort given to making the GAC-supplied version of CINDA-3G fully operational plus the evaluation of specific aspects of the assumptions and procedures incorporated in the code. The OXIDE-3 program was received and is being made operational at ORNL. Modifications are still required to render the TAP program fully operational.

3. REPORT OF MEETING WITH NRC (RSR)

J. P. Sanders and S. J. Ball attended a midyear project review meeting in Germantown, Md., on Dec. 17, 1975.

1. HTGR SYSTEMS AND SAFETY ANALYSES

S. J. Ball

Work for the Division of Reactor Safety Research (RSR) under the HTGR Systems and Safety Analyses Program, which began in July 1974, has been reported previously.¹⁻⁵ Emphasis during the period October through December 1975 was put on model and code refinement and documentation.

1.1 REHEATER AND STEAM GENERATOR MODEL DEVELOPMENT

J. C. Cleveland R. A. Hedrick

The BLAST code was modified slightly to increase flexibility in the selection of convenient nodal descriptions. Also, the routines in BLAST used to determine the transport properties of water were improved; these routines now use the formulations presented in the 1967 ASME steam tables. BLAST calculations were performed to determine the dynamic response of the steam generator to a rapid design-basis depressurization accident (DBDA) and to a reactor trip.

Modeling of the Fort St. Vrain (FSV) steam generator with the BLAST code was initiated. The model will first be compared with the linear model developed by the University of Tennessee and then with FSV dynamics test results.

1.2 DEVELOPMENT OF THE SYSTEM NUCLEAR STEAM SUPPLY SIMULATION CODE (ORTAP) FROM COMPONENT MODELS

J. C. Cleveland

Calculations were performed to verify that the core, reheater, steam generator, and turbine plant component simulations have been correctly coupled in the ORTAP code. The techniques used in the helium circulator simulation are being reviewed so that it can be incorporated into ORTAP. Inclusion of the circulator simulation will complete the initial version of this code.

Emphasis was placed on the documentation of the component models used in ORTAP. Calculations for a sample problem to be included in the documentation of the steam generator component model (BLAST), which is used in ORTAP, were completed. This sample problem simulates the response of the reheater and steam generator to a reactor trip transient. Input information necessary for the BLAST calculation was obtained from a sample problem supplied with the General Atomic TAP code. The reheater-steam generator module was represented by 16 helium nodes, 16 water nodes, and 16 tube nodes (each attached to a water node) as shown in Fig. 1. The version of BLAST used in ORTAP allows up to 20 helium nodes, 20 tube nodes, and 20 water nodes and has the option of using more water and tube nodes than helium nodes. Another version of BLAST allows up to 100 of each type of node to provide sufficient detail in calculating helium and water temperatures and pressures and tube temperatures for input to stress analyses.

Based on the information in the TAP sample problem, once the reactor trip is initiated, the feedwater flow is reduced at a rate of 0.5%/sec to 25% flow, as shown in Fig. 2. The helium flow, Fig. 3, is also reduced by the plant control system during the initial part of the transient to ~27% and then increases to a peak at ~370 sec as a result of an increase in circulator rpm as the control system attempts to maintain the main steam temperature at a preset value. Upon initiation of the reactor trip, the high-pressure turbine throttle valve area is decreased, resulting in an increase in main steam pressure as shown in Fig. 4. At 120 sec after the reactor is tripped, a turbine trip occurs and the steam generator outlet pressure remains essentially constant at 18.023 MPa (2614 psia) as the steam is discharged to the main steam bypass system.

The reheater helium inlet temperature is shown in Fig. 5, and the calculated steam generator helium outlet temperature is shown in Fig. 6. This outlet temperature initially decreases with the decreasing inlet temperature and then peaks again at ~370 sec due to the peak in the helium flow rate (Fig. 3). The main steam outlet temperature is shown in Fig. 7. The small oscillations are due to changes in flow regime and the resulting change in the water-side heat transfer coefficients in the evaporator and superheater nodes. Note that wet steam is emitted

Fig. 1. HTGR steam generator nodal arrangement.

from the steam generator from ~385 to ~550 sec, at which time the steam generator is completely flooded.

The above sample problem is used in the ORNL report⁶ documenting the BLAST code. The initial draft of an ORNL/TM report documenting the core model used in ORTAP is approximately 50% complete.

Fig. 2. Steam generator water inlet flow rate following a reactor trip.

Fig. 3. Reheater helium inlet flow rate following a reactor trip.

Fig. 4. Steam generator water outlet pressure following a reactor trip.

Fig. 5. Reheater helium inlet temperature following a reactor trip.

Fig. 6. Steam generator helium outlet temperature following a reactor trip.

Fig. 7. Steam generator water outlet temperature following a reactor trip.

1.3 CORE SIMULATION FOR EMERGENCY COOLING ANALYSES

S. J. Ball

A report titled ORECA-I: A Digital Computer Code for Simulating the Dynamics of HTGR Cores for Emergency Cooling Analyses (ORNL-TM-5159), was written, reviewed, and edited and is to be issued shortly. Draft copies were provided to NRC personnel. The report gives a detailed description of the FSV version of ORECA, which is the most highly developed version. Subsequently, the 3000-MW(t) reactor version (Fulton and CASSAR) was updated, and plotting routines were written to generate temperature profiles and other parameters (as specified) vs time.

A series of DBDA and loss of main-loop cooling (LOMLC) accident simulations were also run for the Fulton (CASSAR) plant design to investigate steam-graphite reaction (corrosion) rates for the core support posts.

1.4 CORE AUXILIARY COOLING SYSTEM MODEL DEVELOPMENT

S. J. Ball

The core auxiliary cooling tower (CACT) air-blast exchanger model was developed further. Detailed analyses of typical water flow, air flow, and water temperature perturbations were made using a CSMP⁷ language simulation. It was determined that a much-simplified model could be used in an overall CACS loop simulation which could still provide accurate histories of the cooling water inlet temperatures seen by the core auxiliary heat exchanger (CAHE). A simplified scheme (and computer program) was developed for accurately simulating the variable-flow transport and thermal-lag characteristics of the long runs of pipe between the CACT and CAHE. Further development work was done on the CAHE model, primarily in investigating the uncertainties in the helium-side heat transfer coefficients.

1.5 UNIVERSITY OF TENNESSEE SUBCONTRACT

T. W. Kerlin
I. Anastasiou Sen-I Chang
Ming-Huei Lee D. G. Renfro
J. G. Thakkar

Work was continued on dynamic modeling of the HTGR steam generator and the steam turbine helium circulator, primarily to assess model accuracy as a function of complexity. Steam generator dynamics studies are continuing for the Fort St. Vrain (FSV) design for comparison with reactor test data. Methods have been developed for automatic fitting of parameters to provide a match between experimental frequency-response data and theoretical models.

The draft of an ORNL report, HTGR Steam Generator Modeling, was completed and reviewed.

1.5.1 Evaluation of Modeling Assumptions for an
HTGR Steam Generator

Various types of models and methods of calculation are being investigated to determine the capabilities and limitations of commonly used once-through steam generator (OTSG) codes.

The most efficient of these models, which utilizes a three-node moving-boundary scheme, has been developed in several forms. The first is a linear model based on the General Atomic Company (GAC) LAP code. The GAC model was revised to allow for replacement of node-boundary state variables by node-averaged state variables (where applicable). Investigations of the relationships between average and boundary properties (i.e., how these are specified) have been of some concern since the choices of these "weighting factors" have significant effects on the results. A new method of weighting helium temperatures was derived which involves fixing both node-boundary and node-averaged steady-state temperatures and allowing nodal weight factors to vary. This resulted in predicted responses which were not very sensitive to the arbitrary choices of weights. Extensive checking of OTSG models and parameter values was also done in

preparation for the FSV tests. The FSV steam generator code is fully operational and predicts both time and frequency domain responses to perturbations in helium flow, helium inlet temperature, and feedwater flow. The effects of including the gas-side dynamics are also being studied.

A nonlinear, three-node, moving-boundary model was also developed; it is operational for the FSV OTSG configuration and is being used in comparative studies with the linear and fixed-node models.

In the development of a detailed multinode, fixed-boundary, nonlinear steam generator model, the steady-state version of the code was completed, is operational, and is being debugged. Subprograms were tested for heat transfer coefficient, pressure drop, and physical property calculations. FSV design data are being factored into the codes.

The FSV steam generator design will be used as a basis for comparison for all of the codes developed at the University of Tennessee as well as for the ORNL model (BLAST). Data from the FSV tests will, of course, be useful in the evaluations.

1.5.2 Fort St. Vrain Reactor Total Plant Simulator

Most of the component models for the FSV reactor nonlinear simulator have been completed, including the OTSG, reactor, and reheater. The helium circulator code is operational and is being debugged, and several improvements were made to the various component models during this quarter. This simulator will be used in FSV test planning and analysis.

1.5.3 Methods for Systematic Interpretation of Dynamic Measurements in an HTGR

The computer code PARAPE has been tested on several large-system examples for optimizing (identifying) parameter values to yield fits of model response to experimental data. It performs very well and was found to be a factor of 10 more efficient than previously developed parameter estimator codes. Matrices of up to 110×110 can be used. An 84×84 system (FSV) has been tested.

2. COMPUTER CODE IMPLEMENTATION

J. P. Sanders

Work has continued on the evaluation of the systems of computer programs used to produce the RECA calculations by the General Atomic Company (GAC). One of the major efforts related to the full implementation of this program has been the modification of the CINDA-3G preprocessor supplied by GAC for operation on the IBM 360 system. Several features of this preprocessor are, by their nature, machine dependent; operations that are specified within the FORTRAN-V statements for the UNIVAC-1108 can only be performed using Job Control Language (JCL) in the IBM 360 system. These modifications have not been completed.

In conjunction with this program modification, efforts have been devoted to the evaluation of the basic assumptions and analytical representations involved in the RECA program. This has been done both through analytical evaluation and by writing individual programs to test the sensitivity of calculated values to calculational procedures.

The card images of the FORTRAN source program for the OXIDE-3 code were received from Brookhaven National Laboratory (BNL). The original version had been obtained by BNL from Los Alamos Scientific Laboratory (LASL) who had, in turn, obtained the code from GAC. BNL also supplied a card deck of sample input and the resultant output from the execution of a CDC version of the code on their computers. The CDC version of the FORTRAN source program was also supplied. The original version is being modified for operation on the IBM 360 system.

System errors that occurred during the execution of the TAP program have been eliminated. At present, internal checks that are built into the TAP program are terminating its execution; these result from differences in hardware between the two computers. Programming is being added to bypass these differences.

2.1 IMPLEMENTATION OF THE OXIDE-3 CODE ON THE IBM 360 COMPUTER

D. D. Paul

The OXIDE-3 code was received from BNL in November 1975. The code was mailed on a seven-track tape with three files containing the following information:

- File 1: Card images for the CDC version of the OXIDE-3 computer program.
- File 2: Card images of the sample input for the OXIDE-3 computer program.
- File 3: Card images for the UNIVAC-1108, FORTRAN V, deck of OXIDE-3 as transmitted to BNL from LASL with permission of GAC.

As additional information, BNL supplied a punched card deck of their sample input and a sample of the output generated by executing the CDC version of OXIDE-3.

The starting point for making the OXIDE-3 code operational on the IBM 360 system was to transfer the card images of the third file to a restricted disk area on the PDP-10 computer where it could be modified using the TECO (Test Editor and Corrector) program. By a simple series of commands, the modified card images can be submitted to the IBM 360 system through a high-speed data link for compilation and execution. Necessary changes to the OXIDE-3 code were carefully noted and cataloged by the following procedure.

1. The incompatible statement is noted and replaced by a comment statement by adding COXXX in columns 1 through 5. Here XXX stands for a unique integer used to catalog the incompatibility.
2. The IBM compatible statement is added, in most cases immediately following the commented incompatible statement, with columns 72 through 80 containing ORNLOXXX. Here the number XXX is the same as that used above.
3. A brief description of the incompatibility and the correction needed for the IBM 360 system are cataloged in a separate booklet according to the number XXX used above.

Although this editing procedure takes considerable effort, the benefits make the extra time spent worthwhile. For instance, complete documentation of all changes made to the code is readily available. Any discrepancies in output data resulting in changes made during the conversion process are easily traceable, and the original GAC FORTRAN statements are immediately available for comparison in comment statements. This documentation will be valuable both during the conversion of the OXIDE-3 code to the IBM 360 computer and for future reference if any problems should arise with the code.

Presently, the ORNL version of OXIDE-3 has been compiled without any diagnostic messages. Input data will be executed and the generated output will be compared to BNL and GAC sample output to verify that the code is executing properly.

2.2 EVALUATION OF THE RECA PROGRAM

D. D. Paul

In attempting to evaluate any computer program, there are two basic approaches. The first is to evaluate the modeling principles and equations used to approximate the system, and the second is to evaluate the sensitivity of key parameters, such as heat transfer coefficients. Since the RECA code is not fully operational at ORNL, the second type of evaluation is impossible at this time. However, in related work directed toward the development of independent calculations for the evaluation of the thermal response of HTGR cores, there have been many investigations on the effects of different model assumptions. Among the more important modeling decisions affecting the final results is the choice of the axial node spacing. Some observations concerning this investigation are useful in evaluating the RECA program.

Axial node spacing affects the results in a number of ways. First, the power distribution in the axial direction varies considerably; the power increases rapidly near the top and bottom reflectors and has a large discontinuity in the middle. Gross lumping methods cannot represent these variations in detail. There is a marked decrease in accuracy of the convective heat transport modeling with large node spacing.⁸ Also,

the pressure drop calculation, used to determine the flow distribution, will tend to give the improper weighting to the temperature feedback effect in calculating frictional losses for large node spacing.⁹ Finally, there are some numerical inaccuracies inherent in finite-difference techniques depending on the node size.

To test the net effect of node size on results, a multiple-channel, lumped-node code written at ORNL was executed for a Fort St. Vrain DBDA analysis. The first calculation contained 100 axial nodes per refueling region and the second contained 10 axial nodes per refueling region (RECA spacing). Results of the study are shown in Table 1.

Table 1. Effect of node spacing

Parameter	Calculation 1	Calculation 2
Nodes per refueling region	100	10
Maximum core temperature		
Value, °C	1273	1239
Value, °F	2323	2263
Time during transient, hr	3.7	3.3
Maximum coolant exit temperature		
Value, °C	1204	1177
Value, °F	2200	2150
Time during transient, hr	5.6	5.2

The differences in peak temperatures of 33°C (60°F) for the core and 28°C (50°F) for the coolant may not seem significant unless the initial conditions for the transient analysis are considered. At steady-state conditions, the peak fuel temperature was 884°C (1623°F) and the peak coolant temperature was 798°C (1469°F). The differences between the two cases now translate into a difference of 9% for the maximum fuel temperature and 7% for the maximum coolant temperature.

This study demonstrates the importance of evaluating the modeling assumptions of the RECA code. Also, the study demonstrates the usefulness of using other codes to make evaluations.

3. REPORT OF MEETING WITH NRC (RSR)

J. P. Sanders S. J. Ball

RSR personnel were briefed on the financial and technical status of the program at a meeting on December 17 in the Germantown, Md., ERDA Headquarters. No midyear operating budget alterations were made. Plans for the following fiscal year were discussed, and a general shift toward safety work generic to steam-cycle, process-heat, and breeder gas-cooled reactors was recommended.

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