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MASTER

**High-Temperature Gas-Cooled Reactor
Safety Studies for the Division
of Reactor Safety Research
Quarterly Progress Report,
April 1-June 30, 1980**

S. J. Ball
J. C. Cleveland J. C. Conklin
R. M. Harrington

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Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
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HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF REACTOR SAFETY RESEARCH QUARTERLY PROGRESS REPORT, APRIL 1-JUNE 30, 1980

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R. M. Harrington

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CONTENTS

	<u>Page</u>
PRIOR HTGR SAFETY REPORTS	v
FOREWORD	vii
ABSTRACT	1
1. HTGR SYSTEMS AND SAFETY ANALYSIS	1
1.1 Development of the Turbine Plant Dynamics Code ORTURB	1
1.2 Implementation of ORTAP and Subsidiary Codes	2
1.3 Development of the Steam Generator Code BLAST	2
1.4 Assistance with BNL Code Reviews	5
1.5 Preliminary Investigation of Process Heat HTGR Dynamics and Safety	5
2. CONFERENCE ATTENDED UNDER PROGRAM SPONSORSHIP: ANS/ENS TOPICAL MEETING ON THERMAL REACTOR SAFETY, KNOXVILLE, TENNESSEE, APRIL 7-11, 1980	6
REFERENCES	7

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PRIOR HTGR SAFETY REPORTS

Quarterly Progress Reports

<u>Ending date</u>	<u>Designation</u>
September 30, 1974	ORNL/TM-4798
December 31, 1974	ORNL/TM-4805, Vol. IV
March 31, 1975	ORNL/TM-4914, Vol. IV
June 30, 1975	ORNL/TM-5021, Vol. IV
September 30, 1975	ORNL/TM-5128
December 31, 1975	ORNL/TM-5255
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March 31, 1979	ORNL/NUREG/TM-336
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September 30, 1979	ORNL/NUREG/TM-366
December 31, 1979	ORNL/NUREG/TM-383
March 31, 1980	ORNL/NUREG/TM-397

Topical Reports

S. J. Ball, *ORECA-I: A Digital Computer Code for Simulating the Dynamics of HTGR Cores for Emergency Cooling Analyses*, ORNL/TM-5159 (April 1976).

T. W. Kerlin, *HTGR Steam Generator Modeling*, ORNL/NUREG/TM-16 (July 1976).

R. A. Hedrick and J. C. Cleveland, *BLAST: A Digital Computer Program for the Dynamic Simulation of the High Temperature Gas Cooled Reactor Reheater-Steam Generator Module*, ORNL/NUREG/TM-38 (August 1976).

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FOREWORD

High-temperature gas-cooled reactor (HTGR) safety studies at Oak Ridge National Laboratory (ORNL) are sponsored by the Division of Reactor Safety Research, which is part of the Office of Nuclear Regulatory Research of the Nuclear Regulatory Commission (NRC).

This report covers work performed from April 1 to June 30, 1980. Previous quarterly reports and topical reports published to date are listed on p. v. Copies of the reports are available from the Technical Information Center, U.S. Department of Energy, Oak Ridge, TN 37830.

HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR
THE DIVISION OF REACTOR SAFETY RESEARCH QUARTERLY
PROGRESS REPORT, APRIL 1-JUNE 30, 1980

S. J. Ball, Manager
J. C. Cleveland J. C. Conklin
R. M. Harrington

ABSTRACT

Development of the new steam turbine plant model ORTURB was completed. Further development, implementation, and verification work was done on the BLAST steam generator code.

1. HTGR SYSTEMS AND SAFETY ANALYSIS

S. J. Ball

Work for the Division of Reactor Safety Research under the High-Temperature Gas-Cooled Reactor Systems and Safety Analysis Program began in July 1974, and progress is reported quarterly. Work during the present quarter included further code development and implementation.

1.1 Development of the Turbine Plant
Dynamics Code ORTURB

J. C. Conklin

Development of the improved turbine plant model ORTURB (Fig. 1) has been completed. A turbine runback from 100 to 25% power was simulated, and excellent agreement with published heat balances was obtained for high- and low-pressure turbines and feedwater heaters. In addition, transients representing a loss of condenser vacuum, loss of a feedwater heater, and high pressure turbine exit pressure fluctuations were simulated. The ORTURB model yielded appropriate responses for all transients. The first draft of formal documentation for ORTURB has been completed.¹

During development of ORTURB, the earlier feedwater heater model² was observed to calculate shell-side liquid temperatures greater than saturation. The problem was corrected by modifying the system of differential equations so that shell vapor temperature as a function of mass, rather than vapor mass in the heater shell, was represented as a state variable.

The ORTURB model was also successfully incorporated into the ORTAP-FSV code,³ replacing the previous turbine model. Sample transients run on the complete system showed results agreeing very well with expected steady-state end point conditions.

Fig. 1. Fort St. Vrain Reactor turbine-generator plant flow diagram.

1.2 Implementation of ORTAP and Subsidiary Codes

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

Frequent discussions with Public Service Company of Colorado (PSC) have been held regarding their implementation of the ORTAP,³ BLAST,⁴ and ORECA⁵ codes. The PSC has had some difficulties in using these codes because of apparent differences in their computer systems and changes in the input data sets.

1.3 Development of the Steam Generator Code BLAST

J. C. Cleveland J. C. Conklin

The PSC has recently provided Oak Ridge National Laboratory (ORNL) with more detailed design information for the Fort St. Vrain (FSV) steam generators than previously available at ORNL. Based on this information, a revised FSV steam generator model has been prepared for the BLAST code.⁴

Certain problems in the steady-state search routines of BLAST have been encountered both at ORNL and at PSC while using the new design information. A modification has been planned that should solve this convergence problem as well as another convergence problem encountered occasionally at low-power conditions. Programming of this modification is under way.

Additional steam generator transient data have been requested from General Atomic Company (GA) for the response of one of the steam generator modules during a particular oscillation transient encountered at FSV. Also, measured steady-state data for the highly instrumented steam generator module were requested. GA has contacted ORNL indicating that requested data will be supplied. This data will be very beneficial in efforts to benchmark BLAST against measured data.

A review of coding and input requirements for BLAST resulted in a new method of formulating the input, which is recommended to improve physical and mathematical modeling as well as decrease computational costs.

The BLAST code was originally written for the large (3000-MWt) HTGR design, which had a tunnel to carry helium from the reheater exit to the steam generator inlet. Since the FSV steam generators have no tunnels, the tunnel node can be eliminated from BLAST in the following manner (refer to Ref. 4):

1. The first helium node should have the upstream node number on card series 80KK designated as 201.
2. The last helium node should have the downstream node number on card series 80KK designated as 251.
3. Helium nodes connecting the first and last nodes should not have either an upstream or downstream node number between 201 and 251.
4. Any and all input entries for a tunnel should be ignored. This specifically includes deletion of card series 9001 from the input deck.

By deleting the tunnel entries from the input deck, large portions of coding are bypassed, and the number of equations that need to be solved is decreased. However, the subroutine OUTLNK must be slightly modified to avoid printing undefined variables concerning the reheater outlet and steam generator inlet for the helium side.

It is also recommended that the node-numbering scheme have the relatively lower helium node number exchange heat with a corresponding lower water node number. This change minimizes the bandwidth of the A-matrix, a desirable objective when using a banded matrix solver. It is not necessary to number the nodes, either water or helium, sequentially with the fluid flow direction. A banded matrix equation solver has been tested, resulting in a decrease of computation cost as compared with a general matrix equation solver. However, software for the banded solver is problem-specific presently. The general matrix equation solver using Gaussian elimination is adequate for general use at this time.

Tube length data based on construction (as built) for the steam generator geometry has been received from PSC and is presented in Fig. 2 along with an example of the node-numbering scheme previously mentioned.

Fig. 2. Revised nodal arrangement for BLAST simulation of FSV steam generator.

1.4 Assistance with BNL Code Reviews

S. J. Ball

Brookhaven National Laboratory (BNL) has acquired HTGR system and component dynamics codes developed at ORNL and has begun implementing and reviewing them. Background information and code development notes have been supplied to BNL to assist them in evaluation of the ORECA code,⁵ which models 3-D core temperature and flow distributions in the FSV reactor.

1.5 Preliminary Investigation of Process Heat HTGR Dynamics and Safety Characteristics

S. J. Ball	J. C. Cleveland
J. C. Conklin	R. M. Harrington

A brief, preliminary study was made of HTGR plant concepts for generating process heat as well as electricity, using a steam-methane reformer HTGR plant (R-HTGR) design as a reference. The plant is expected to have some unique licensing features due to proximity of process gas systems to the reactor. Development of a plant dynamics simulation capability is being funded by DOE. Because of unique features of the R-HTGR, it was concluded that NRC research involvement in the program should begin in FY 81 or FY 82.

2. CONFERENCE ATTENDED UNDER PROGRAM SPONSORSHIP: ANS/ENS
TOPICAL MEETING ON THERMAL REACTOR SAFETY,
KNOXVILLE, TENNESSEE, APRIL 7-11, 1980

S. J. Ball J. C. Cleveland
J. C. Conklin R. M. Harrington

The subject conference included many sessions on various aspects of reactor safety and licensing; although there were none devoted specifically to HTGR's, much of the information was generic enough to be useful to the HTGR safety program. Sessions of particular interest were: Class 9 Accidents and Containment, Reactor Event Reporting and Analyses, Computational Methods, Statistical Assessment of Accident Potential, Verification of Computational Methods, Issues with Respect to Improved Safety, and Unresolved Safety Issues. Also, several sessions on the TMI-2 accident and its aftermath were related to changes in the approaches that will be used in licensing reactors.

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3. J. C. Cleveland et al., *ORTAP: A Nuclear Steam Supply System for the Dynamic Analysis of High-Temperature Gas-Cooled Reactor Transients*, ORNL/NUREG/TM-78 (September 1977).
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