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# High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research Quarterly Progress Report, July 1–September 30, 1979

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

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, JULY 1–SEPTEMBER 30, 1979

S. J. Ball, Manager
J. C. Cleveland   J. C. Conklin



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

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

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

J. C. Cleveland, CORTAP: A Coupled Neutron Kinetics—Heat Transfer Digital Computer Program for the Dynamic Simulation of the High Temperature Gas Cooled Reactor Core, ORNL/NUREG/TM-39 (January 1977).

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

S. J. Ball et al., Evaluation of the General Atomic Codes TAP and RECA for HTGR Accident Analyses, ORNL/NUREG/TM-178 (May 1978).

# FOREWORD

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 July 1 to September 30, 1979. 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, JULY 1—SEPTEMBER 31, 1979

S. J. Ball, Manager
J. C. Cleveland   J. C. Conklin

## ABSTRACT

Further development work was done on the ORTAP and BLAST codes. A new and improved model of the Fort St. Vrain (FSV) reactor turbine-generator plant (ORTURB) was developed for use both as a stand-alone code and as a part of the ORTAP system code. Additional work was done on FSV licensing questions. The intermediate heat transfer experiment for investigating FSV upper-plenum reverse-flow plumes was assembled and checked, and an on-line computer was set up to acquire and analyze the data.

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## HTGR SYSTEMS AND SAFETY ANALYSIS

S. J. Ball

Work for the Division of Reactor Safety Research (RSR) under the High-Temperature Gas-Cooled Reactor (HTGR) Systems and Safety Analysis Program began in July 1974, and progress is reported quarterly. Work during the present quarter included assistance to the Nuclear Regulatory Commission (NRC) on Fort St. Vrain (FSV) reactor licensing questions and further work on code development.

### Development of the FSV Nuclear Steam Supply System Simulation Code ORTAP-FSV

J. C. Conklin

A new computer simulation of the high-pressure turbine (HPT) has been written which improves the accuracy of the predicted performance and uses less computer time than the earlier HPT model. This new model combines two empirical relationships derived for noncondensing turbines. The HPT first-stage pressure is assumed to be a function of the desired turbine load only and not a function of the conditions in the main steam line. An empirical plot[1] is used to determine the first-stage shell pressure, given the desired turbine load. A modified Stodola equation[2] is then used to determine the HPT flow rate. With the Stodola relationship, the turbine exit pressure influences the turbine flow rate. Also, the performance of the HPT is calculated directly, with no iterative solution techniques.

Steady-state turbine plant data at two different operating levels are necessary to set parameters such as thermal efficiencies and other constants in the Stodola equation. Straight-line interpolations are used when necessary.

The regenerative Rankine cycle of the intermediate- and low-pressure turbine (ILPT) at FSV utilizes five feedwater heaters (FWHs) and a deaerator. The FWH model used in the earlier version of ORTAP[3] is also used in the new simulation. This model uses nine differential equations per FWH, and the equations are solved using the matrix exponential method.[4]

The pressure and flow distribution in the new ILPT simulation model is solved iteratively because of the coupling between the turbine extraction points and the FWHs. In the analytical model for the ILPT, the assumption is made that, for condensing turbines at a given temperature, the pressure preceeding any section of a turbine is directly proportional to the flow through that section of the turbine.[1] The pressure ratio across each stage is assumed constant, subject to temperature corrections. This relationship between the pressure, flow, and temperature is used to determine stage pressure. The ILPT is divided into seven segments, and each segment ends with an extraction point, with the last segment terminating in the condenser. The flows from the turbine extraction points to the FWHs are assumed proportional to the square roots of the pressure differences.

Inputs to the ILPT model are inlet pressure and condenser outlet enthalpy. The flow and pressure distribution of the ILPT then are found by iteration, using the constant pressure ratio and extraction flow relationships. Empirical constants are also derived from manufacturer's data at 100% power conditions.

This model of the steam turbine has yielded reasonable results when compared with published heat balances[5] and uses ∿15 s of central processing unit (CPU) time for a 200-s turbine transient. It presently is being incorporated into ORTAP. A report on the new steam turbine model subroutine (ORTURB) is being drafted.

The version of ORTAP in use during the past few years [and as sent to Rheinisch-Westfalischer Technischer Uberwashungs-Verein e.V. (RWTUV), West Germany, and Ishikawajima-Harima Heavy Industries Co., Ltd. (IHI), Japan] executed properly on the Oak Ridge National Laboratory (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 subroutine SUPORT, which calculates the average reactor core outlet gas temperature. SUPORT was modified to eliminate the errors, which were related to underflows (masked at ORNL but not at ORGDP) and the differences in the way variables are initialized. A few minor corrections to the subroutine logic were also implemented to account properly for coolant flow through the control rod guide tubes. All corrections to SUPORT were marked by appropriate comment cards.

The corrected code now executes properly at both 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.

ORTAP is also undergoing a general renovation, with the dual objectives of decreasing computation costs while maintaining an appropriate level of accuracy.

## ORECA Code Calculations of Postulated FSV Reactor LOFC/FWCD Accidents for Core Thermal Stress Evaluations

S. J. Ball

Professor Theophanous of Purdue University [who was working on related FSV thermal analysis problems for NRC Division of Project Management (DPM)] noted that, during the firewater cooldown (FWCD) phase of a postulated 90-min loss-of-forced-convection (LOFC) accident, the predicted differences between certain adjacent core regions' lower reflector and support block temperatures were very large, some being as great as ∿840°C (1500°F). There was some concern that these gradients may cause large thermal stresses in the support block regions. These large temperature differences at the bottom of the core result from the uneven region temperature profiles that are generated during the LOFC. The regions with higher region peaking factors (RPFs) experience reverse (upward) flows, which transport the core heat up toward the core top. After the FWCD begins, the forward flow drives the heat downward, temporarily raising the temperature at the base of the core to a much greater degree than those low RPF regions that had downflow during the LOFC phase of the accident.

Calculations of the predicted thermal performance were made using the ORECA[6] code and forwarded to NRC and to Dr. Charles A. Anderson [Los Alamos Scientific Laboratory (LASL)], who is to perform the stress analyses. A modified version of the "standard" ORECA core nodal approximation was used for the calculations. Usually, the lower reflector and core support block for each radial region are represented by one node. In the present case, the lower 0.38 m (15 in.) of each core support block is represented by a separate node (axial node 9), while the rest of the support block plus the reflector are lumped as one node (axial node 8). This model change had been made previously in a version of the ORECA code used in comparisons of FSV scram data with predictions.[7] The lower portion of the support block, which has considerably less heat transfer area than the average region element, was found to cool down much more slowly after a scram. The resulting higher values of $T^4$ radiation to the region outlet thermocouple assemblies yielded much better agreement with the scram test data.

Sensitivity studies have also been done to see if variations of the presumed "worst-case" accident assumptions would lead to instances of larger thermal gradients. For example, the pessimistically low assumed value of FWCD helium circulation flow was increased, but the low flow was found to give the largest gradients.

Figure 1 shows the temperature difference between the typically hottest node in axial region 8 (reflector and upper support block) and its coolest neighbor, radial regions 19 and 35, respectively. The difference peaks at ∿7200 s (120 min) after the start of cooling.

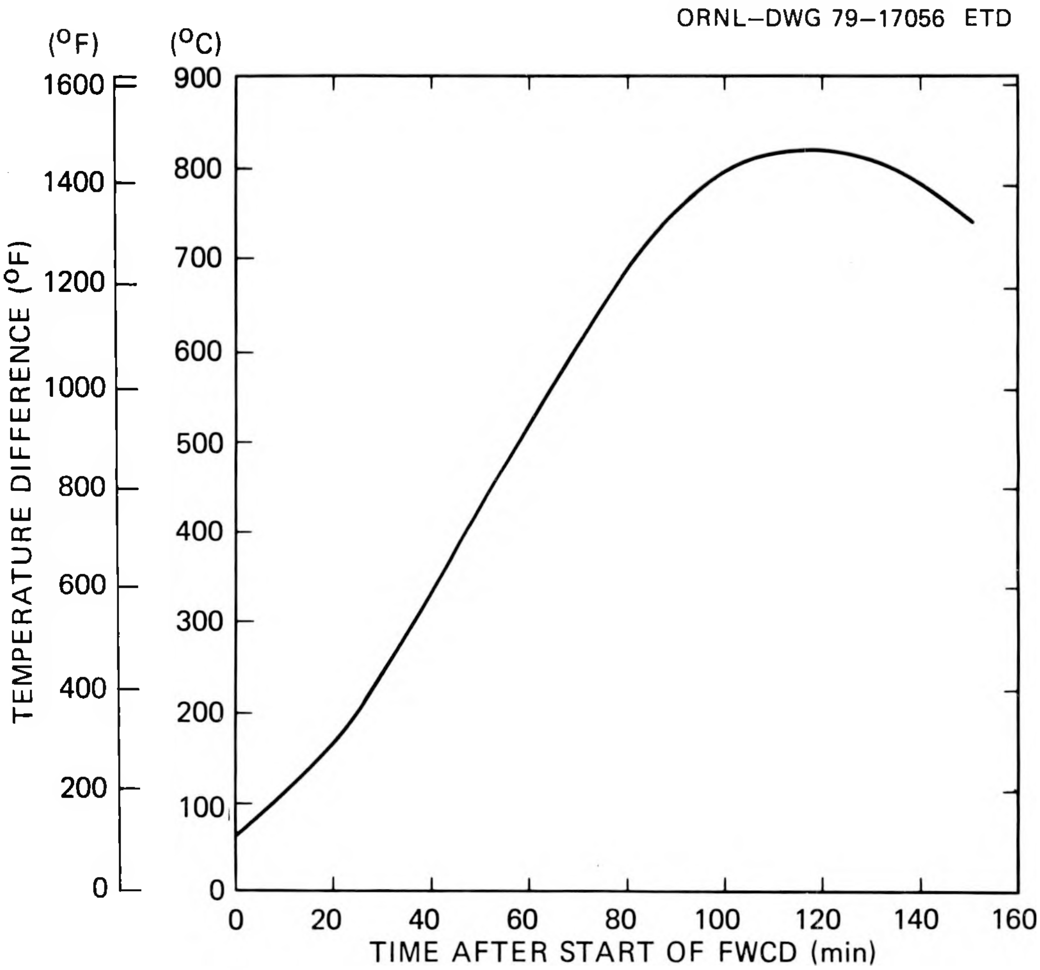


Fig. 1. Temperature difference between regions No. 19 (hot) and No. 35 (cool) after FWCD at axial region 8 elevation (lower reflector plus top of support).

Consultations with LASL are continuing as to their needs in evaluating the stresses.

## Development of the Steam Generator Code BLAST

J. C. Cleveland

Informal discussions were held with H. Schuldt of RWTUV concerning both RWTUV's plans to use the BLAST[8] code in Thorium High-Temperature Reactor (THTR) simulations and ORNL's currently planned HTGR safety activities. The RWTUV's plans involve the use of BLAST to analyze transients, as required by the West German licensing process, during 1980. RWTUV completed the BLAST model for the THTR reheater and steam generator. This model will be incorporated into a plant simulation of the THTR by the Institut fur Reaktorentwicklung at Kernforschungsanlage (KFA). The RWTUV also completed a model of the Arbeitgemeinshaft Versuch Reaktor (AVR) steam generator with BLAST in preparation for BLAST verification activities, which would compare BLAST with measured data from AVR transients.

A suggested change in the BLAST technique for calculating main and reheat steam outlet enthalpy from conditions in the last main steam and reheater nodes was discussed with H. Schuldt of RWTUV. This potential change will be tested by RWTUV with their THTR steam generator model. Results of RWTUV's test will be provided to ORNL for examination.

Implementation of modified versions of BLAST made available to ORNL by RWTUV continued. These versions include several improvements such as a modification allowing a restart after the initial steady-state calculation or during the transient, a modification in the subroutine for computing two-phase flow multipliers to extend the pressure range, a more rapid matrix inversion technique, and 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 the BLAST capability and represent considerable effort by RWTUV.

## FSV Upper-Plenum Reverse-Flow Plume Experiments

S. J. Ball   D. J. Fraysier

The final design of the intermediate reverse-flow plume experiment was completed, all the components were acquired, and assembly and installation were completed. The purpose of the experiment is to determine the validity of using Reynolds (Re) and Grashof (Gr) similarity relationships to extrapolate from low-temperature, low-pressure air plume models to the HTGR case, that is, high-temperature high-pressure helium.

The main features of the intermediate plume experiment are shown in Fig. 2. A rotameter measures air flow through a heater assembly and into a nozzle (with adjustable diameters), which directs the heated air to a thin flat plate mounted above 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 which serves as a calorimeter

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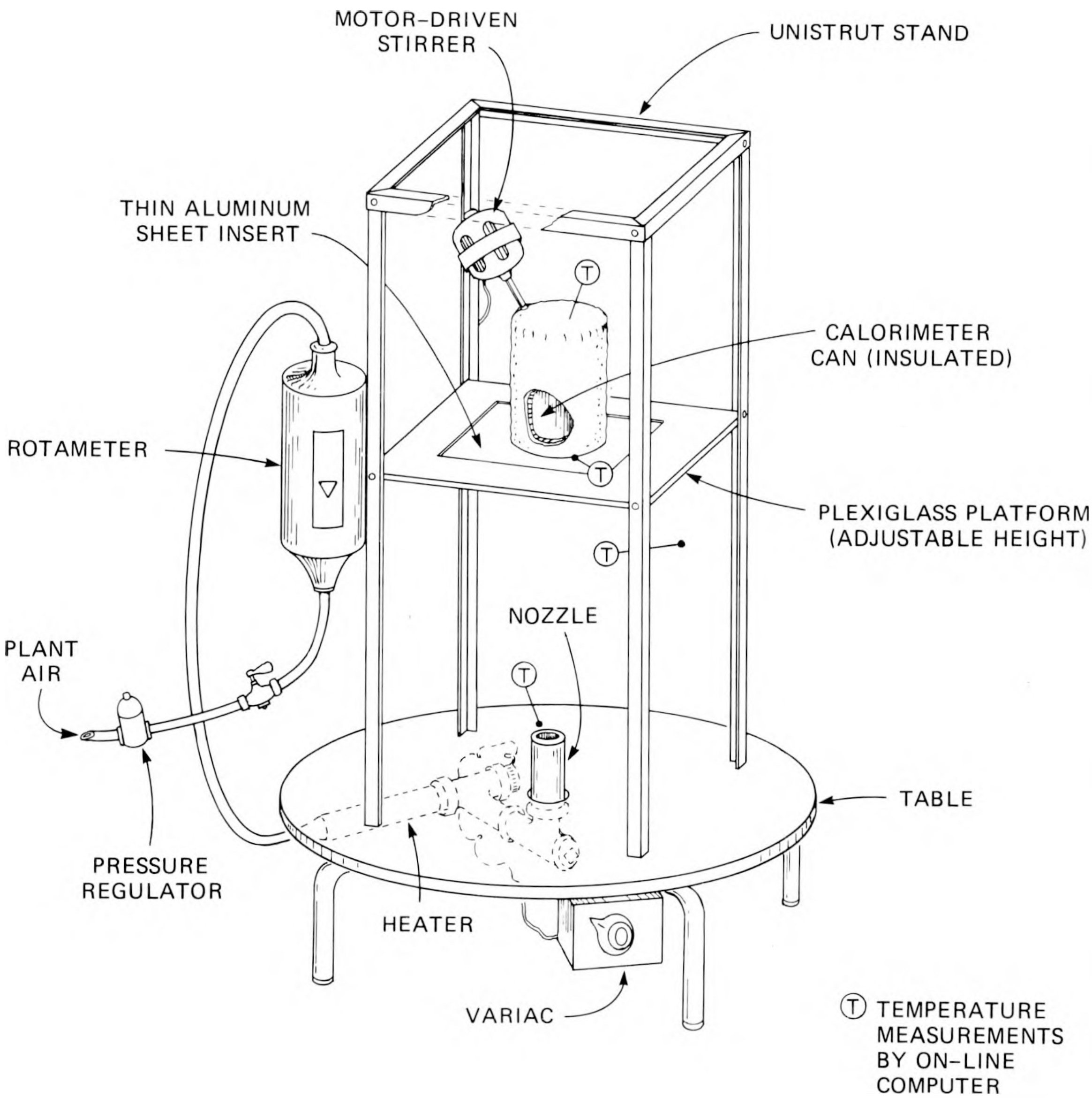


Fig. 2. Intermediate heated-plume experiment assembly.

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 360° 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.

Initial results have been obtained; however, problems with repeatability and Nusselt number accuracy prevented drawing any conclusions from the data obtained during the quarter. (The repeatability problems were subsequently overcome.)

## Implementation of the JAERI Code SCOTCH

S. J. Ball

As a result of discussions with M. Ezaki of the Japan Atomic Energy Research Institute (JAERI) in Tokai, we received a copy of the SCOTCH code, which was developed for simulations of the Japanese VHTR core. Because the code may be useful in investigations of FSV postulated accidents, we plan to try and implement it on the local IBM computers. The abstract of the report accompanying the code (JAERI-M 8292) is as follows:

SCOTCH: A Program for Solution of the One-Dimensional, Two-Group, Space-Time Neutron Diffusion Equations with Temperature Feedback of Multi-Channel Fluid Dynamics for HTGR Cores

Masahiro Ezaki, Tamotsu Ozawa,* and Susumu Mitake
Division of Power Reactor Projects, JAERI

* Visiting scientist from Kawasaki Heavy Industries, Ltd.

The SCOTCH program solves the one-dimensional (R or Z), two-group reactor kinetics equations with multi-channel temperature transients and fluid dynamics. Subprogram SCOTCH-RX simulates the space-time neutron diffusion in the radial direction, and subprogram SCOTCH-AX simulates the same in the axial direction.

The program has about 8,000 steps of FORTRAN statements and requires about 102 K words of computer memory.

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