**Monthly Progress Report for December 1975
for the
HTGR Safety Studies for the
Division of Technical Review
U.S. Nuclear Regulatory Commission**

J. P. Sanders, Manager

MASTER

OAK RIDGE NATIONAL LABORATORY

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MONTHLY PROGRESS REPORT FOR DECEMBER 1975

for the
HTGR SAFETY STUDIES FOR THE
DIVISION OF TECHNICAL REVIEW
U.S. NUCLEAR REGULATORY COMMISSION

J. P. Sanders, Manager

January 1976

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Regular Progress Reports
HTGR Safety Studies at ORNL for the
Office of Nuclear Reactor Regulation, U.S. NRC

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Bimonthly	March 31, 1973	ORNL-TM-4753
Quarterly	June 30, 1973	ORNL-TM-4741
Quarterly	September 30, 1973	ORNL-TM-4739
Quarterly	December 31, 1973	ORNL-TM-4738
Quarterly	March 31, 1974	ORNL-TM-4736
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Quarterly	September 30, 1974	ORNL-TM-4806
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Quarterly	June 30, 1975	ORNL-TM-5120
Monthly	July 31, 1975	ORNL-TM-5061
Monthly	August 31, 1975	ORNL-TM-5085
Monthly	September 30, 1975	ORNL-TM-5108
Monthly	October 31, 1975	ORNL-TM-5155
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HTGR SAFETY STUDIES FOR THE DIVISION OF TECHNICAL REVIEW U.S. NUCLEAR REGULATORY COMMISSION

I. INTRODUCTION (J. P. Sanders)

This is the sixth in a series of monthly reports describing work performed for the Division of Technical Review, USNRC, concerning the analysis of the transient response of HTGR systems under abnormal conditions. As in the previous report, the work described is for the period from November 15 through December 15 to provide for a timely publication schedule. A list of all of the reports issued in this series is presented on the opposite page.

The report (ORNL-TM-4931) describing the determination of physical properties of mixtures of helium and air has been prepared and submitted for reproduction. This report should be issued early in the next reporting period.

A report (ORNL-TM-5206) has been issued that describes the results of calculations performed to determine the performance of the thermal barrier cover plates at specified temperature limits. Two types of cover plates were analyzed; these were the carbon steel hexagonal plates in the upper plenum and the rectangular plates, which are made of Hastelloy X, in the lower cross ducts of proposed large HTGRs.

The development and evaluation of HEXEREI has continued; HEXEREI is a multi-channel, lumped-node, coupled conduction-convection code to evaluate the thermal conditions in an HTGR core. A preliminary report has been written to provide the results of test cases executed using HEXEREI and the HEATING5 program plus analytical solutions. The ACHERONS calculation has been incorporated with the HEXEREI code to determine the flow provided by the core auxiliary cooling systems.

The complete transient for the specified DBDA for the Fort St. Vrain reactor core has been executed using the HEXEREI code. The output has been compared with the results from GAC and other ORNL calculations.

II. HEXEREI CODE DEVELOPMENT (G. E. Giles, R. M. DeVault)

Test cases for validation of the HEXEREI code were completed, and a rough draft of a report describing the results has been written.

Four simple cylindrical conduction problems were used as input to HEXEREI and to CCCM5, and the calculated values were compared with analytical results. All four test cases used a cylindrical model with homogeneous properties. The first test case fixed the surface temperature at the outer curved surface of the cylinder, and a uniform volumetric heat generation was used. The second test case fixed both the outer surface and centerline temperatures and had no heat generation. The third test case fixed the upper and lower temperatures and retained the heat generation. The first three test cases were used in steady-state calculations. The fourth test case was used in transient calculations. For the fourth test case, the initial temperature of the cylinder was set equal to a uniform temperature, and a time-dependent outer surface temperature was applied. All of the above test cases were repeated for temperature-dependent material property values.

The results of the test cases are within 1.5% of the analytical results for steady-state calculations; and, for the transient analysis, agreement was within 3%.

A fifth test case using the same cylindrical model with uniform volumetric heat generation, coolant flow rate, and coolant inlet temperature was compared to an analogous case for CCCM. Good agreement between the results was obtained.

In addition, work progressed on refining the closed-loop analysis that would use the computed results from the CACS system performance calculations to provide input for the core performance calculations. Certain conditions during the transient prevent the iterative technique from converging.

Calculation of a DBDA for the Fort St. Vrain reactor for comparison with GAC and other ORNL codes reached the first stage. Work is in progress to compare results and to isolate differences in the computational methods.

III. STATUS OF THE FLODIS CODE (D. D. Paul)

The FLODIS code was modified to include the effects of heat conduction in the axial direction within each refueling region. A Crank-Nicholson finite difference technique was used to reduce the resulting differential equations to a set of linear algebraic equations. The algebraic equations are solved using a Gauss-Seidel iteration scheme. These modifications resulted in a 25% increase in computing time.

Calculations were completed for determining the initial conditions for the transient analysis of the Fort St. Vrain DBDA. It was shown in the monthly progress report for October 1975 that a lumped parameter technique could not adequately model the radial space dependency of the temperature at full-power, steady-state conditions due to the temperature gradients across the fuel stick and graphite moderator. Following trip of the reactor, these gradients significantly decrease. For the DBDA accident, the correct initial condition is the volume-averaged nodal temperature at full-power, steady-state conditions. A volume-averaged nodal temperature was calculated using results from the CCCM code. An effective conductive path length was defined in the FLODIS code such that the calculated initial temperatures most nearly matched those calculated by the CCCM code. Agreement between the two models was excellent.

The Fort St. Vrain DBDA was rerun with the correct initial condition and with the effects of axial conduction included. These results and a documentation of the FLODIS code will be available in the near future.

IV. ANALYSIS OF THERMAL BARRIER COVER PLATES (S. K. Iskander)

Work was initiated in the review of GA-LTR-1, the licensing topical report on the analysis of HTGR core cooling capability, to evaluate the performance of the thermal barrier cover plates at specified design limits. The results of this analysis have been reported in ORNL-TM-5206, which is in the process of publication. The abstract of that report is as follows:

Cover plates are used in current designs for high temperature gas-cooled reactors to compress the mineral fiber insulation against the inside of the liner of the prestressed concrete pressure vessel. In the upper plenum, these plates are hexagonal and specified as carbon steel; in the lower cross ducts, the

plates are square and made of Hastelloy X. The General Atomic Company has specified both damage and safety limit criteria for these plates. These plates have been analyzed at these limits using the inelastic finite element computer program EPACA. The results indicate that the total strains for the square plate were within the specified values; however, the maximum deformations at the free corners indicate separation from the insulation and failure to achieve one of the design requirements. Since no material data were available for carbon steel at the limiting temperatures, it was assumed that the hexagonal plates were constructed of 2½% Cr-1% Mo material. Although this material was found to produce satisfactory performance, extrapolation of available information would lead to the conclusion that the performance of carbon steel plates would not be satisfactory at the specified conditions.

V. PHYSICAL PROPERTY VALUES FOR HELIUM-AIR MIXTURES (J. R. Tallackson)

In the evaluation of the performance characteristics of the core auxiliary heat exchanger following a depressurization accident, it is necessary to accurately predict the thermal transport properties of mixtures of air and helium. A report giving both values and correlations developed in this evaluation has been prepared and has been submitted for review and publication. The abstract for this report (ORNL-TM-4931) is as follows:

This report presents, quantitatively, the thermal transport properties of the materials involved in digital computer calculations of heat transfer rates by the Core Auxiliary Heat Exchangers in large HTGR nuclear steam supply systems. These materials are pure helium, mixtures of helium with common gases having molecular weights in the range of 28 to 32, alloy steel tubing, and water.

For use in programmed computations, the viscosity, thermal conductivity, and specific heat are represented primarily by equations augmented by curves and tabulations. Materials supporting the development and selection of property equations are included.

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