**Monthly Progress Report for January 1976
for the
HTGR Safety Studies for the
Division of Systems Safety
U.S. Nuclear Regulatory Commission**

J. P. Sanders

MASTER

OAK RIDGE NATIONAL LABORATORY

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U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161
Price: Printed Copy \$4.50; Microfiche \$2.25

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Contract No. W-7405-eng-26

MONTHLY PROGRESS REPORT FOR JANUARY 1976
for the
HTGR SAFETY STUDIES FOR THE
DIVISION OF SYSTEMS SAFETY
U.S. NUCLEAR REGULATORY COMMISSION

J. P. Sanders, Manager

(This work was performed for the U.S. Nuclear Regulatory
Commission under interagency agreement ERDA No. 40-545-75)

FEBRUARY 1976

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Regular Progress Reports
HTGR Safety Studies at ORNL for the
Office of Nuclear Reactor Regulation, USNRC

TYPE	FOR THE PERIOD ENDING	DESIGNATION
Bimonthly	January 31, 1973	ORNL-TM-4752
Bimonthly	March 31, 1973	ORNL-TM-4753
Quarterly	June 30, 1973	ORNL-TM-4741
Quarterly	September 30, 1973	ORNL-TM-4739
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Quarterly	September 30, 1974	ORNL-TM-4806
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Monthly	November 30, 1975	ORNL-TM-5205
Monthly	December 31, 1975	ORNL-TM-5238
Monthly	January 31, 1976	ORNL/TM-5268

MONTHLY PROGRESS REPORT FOR JANUARY 1976

HTGR SAFETY STUDIES FOR THE DIVISION OF SYSTEMS SAFETY U.S. NUCLEAR REGULATORY COMMISSION

I. INTRODUCTION (J. P. Sanders)

This is the sixth in a series of monthly reports describing work performed concerning the analysis of the transient response of HTGR systems under abnormal conditions. During the first half of this fiscal year, the work was under the guidance of the Division of Technical Review, USNRC. Because of the reorganization of the Office of Nuclear Reactor Regulation (ONRR) during the month of December 1975, the work will now be reported to the Division of Systems Safety.

As in the previous reports of this monthly series, the work described in this report was performed in the interval from December 15, 1975, through January 16, 1976. A short-range projection will be included to indicate the anticipated status at the end of the reporting month. A list of all of the reports issued in this continuing study is presented in the front of this report.

A meeting was held in Washington on December 18, 1975, to review the effect on this work of recent changes in the status of licensing requests for HTGRs and the reorganization of the ONRR. In general, it was concluded that the work would continue during the current fiscal year with the major emphasis being shifted to problems related to the licensing of the Fort St. Vrain reactor and to the completion at ORNL of the development of analysis codes for the large HTGR designs. The code development would be documented, and the computational procedures would be maintained at ORNL for future use.

Two options were established for work during future fiscal periods. The choice of options will depend upon progress in the Fort St. Vrain startup operations, the continuation of the review of the GASSAR-6 document, the review requirements for licensing topical reports, and the consequences of reorganization of the ONRR on the distribution of work and budget assignments.

One of the major undertakings during the past month has been the evaluation of conditions in the Fort St. Vrain reactor following a loss of forced convection cooling accompanied by a reactor trip and loss of PCRV liner cooling-water flow for a limited period. The consequence of this event at 40% power during the rise-to-power program is described in Attachment 2 to a letter¹ from R. F. Walker of the Public Service Company of Colorado. The results of this work have been transmitted by telephone and telecopy to the USNRC.

For this calculation, it was necessary to represent, in detail, the thermal barrier cover plates, the insulation, the PCRV liner, and the concrete of the PCRV in the thermal analysis. Property values for these materials have been obtained and evaluated. Computations have been performed using the general purpose HEATING computer program and the HEXEREI code that was developed for HTGR core thermal analysis.

General development has continued for the FLODIS computer program. Calculations have been performed to evaluate the effect of varying problem parameters. Documentation for this program is being prepared.

The final draft has been prepared for a report that describes the CACHE program that was written to determine the performance characteristics of the core auxiliary heat exchanger bundle. This program, in a modified form, is incorporated in the overall loop calculation for the core auxiliary cooling system. The report for the CACHE program, ORNL/TM-4952, is currently being reviewed and should be published in the near future.

II. THERMAL ANALYSIS OF HTGR SYSTEMS

The general development of computer methods for the analysis of thermal and flow conditions during abnormal transients has continued. A special problem relative to the thermal analysis of loss of cooling in the Fort St. Vrain reactor during the initial phase of operation has been assigned.

II.1 Thermal Analysis for the Fort St. Vrain HTGR (K. W. Childs, W. D. Turner)

A two-dimensional RZ model has been developed as an initial step in the analysis of the Fort St. Vrain HTGR following a loss-of-cooling accident while operating at 40% power during the rise-to-power program. This model, which was executed using the HEATING5 computer code, included the PCRV. The model of the active core consisted of homogeneous concentric cylinders representing a composite of graphite, fuel, and coolant. Each cylinder represented a zone at a different power level. The transient temperature distribution was obtained following a total loss of forced circulation and liner cooling water flow. This analysis is being continued and modified.

II.2 HEXEREI and HEXEREI2 Development (G. E. Giles, R. M. DeVault, B. R. Becker)

The HEXEREI code's capabilities were expanded to include some of the generalized boundary condition capabilities of HEATING5; evaluation of the extension is in progress. The FLODIS routines were run extensively to investigate convergence problems during some of the transient sequences. The root of the problem is that the Newton-Raphson technique requires a close first guess to prevent divergence. Modifications are being developed to overcome this problem. Modification of the coolant routines (EXPAM, EXPAME) to incorporate the core support blocks was achieved and evaluation is in progress. The addition of the ACHERONS routines to HEXEREI, producing HEXEREI2 with complete closed loop capabilities, has progressed to running a test case which has driven the loop for 1 hr of model time. A Fulton plant model will be developed for comparison with previous data provided by GAC.

The addition of the implicit technique has worked for the test cases involving only conduction. The coolant test cases exhibit problems in convergence.

I.2.1 Studies

The HEXEREI code was used to analyze the Fort St. Vrain loss-of-cooling accident studies from 40% power as requested by NRC in January 1976. The adiabatic heat-up of the Fort St. Vrain reactor after loss of forced convection was analyzed and compared with HEATING5 results for a similar model which included an estimate of the heat loss to the PCRV. The HEATING5 model was one of less detail in the core area. The development of the generalized boundary conditions enables the development of a model to simulate the start-up of the PCRV liner cooling water flow after an adiabatic heat-up for 48 hr and subsequent cool-down of the reactor.

HEXEREI was also used to model the cool-down on four circulators for 15 min after sequence 44 in the rise-to-power program (Fig. 1 in Ref. 1). Convergence problems in FLODIS have prevented calculation of a significant portion of the transient time. A similar set of analyses for the circulator speed of 7000 and 8000 rpm was performed and a complete transient can be calculated.

II.3 CCCM Code Development (W. D. Turner, D. C. Elrod)

A few minor modifications and corrections were made to the CCCM computer code. HEATPLOT is a computer code which is used to generate various plots from the CCCM and HEATING computer codes. Modifications are being made to HEATPLOT to enable the same version to execute on any of the large computing systems in Oak Ridge. Some additional capabilities are being added to the code. This includes replacing the contour plotting subroutine with a package (developed on another project) which is both more efficient and more accurate. HEATPLOT, or a modified version of it, should be useful in depicting results from HEXEREI.

II.4. Development of Flow Distribution Code for LOFC and Low-Flow Conditions (K. W. Childs)

Work was begun during this period to analyze the flow in the reactor core during very low coolant flows and loss of forced circulation coolant flow conditions. Under both of these flow conditions, convective loops may develop within the reactor core cavity. In these convective loops, a flow reversal occurs in the hotter channels such that flow is up in these channels and down in the colder channels. Because of this flow reversal and because of the very low flow (or zero flow) which occurs in this situation, existing computer codes developed to calculate the flow distribution among refueling regions have not been successful in analyzing this case. Therefore, a new computer program will be developed to model these convective loops in the reactor core. This program will be written so that it can be incorporated into HEXEREI as a subroutine.

III. AXIAL NODE SPACING IN THE FLODIS CODE (D. D. Paul)

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.² Also, the pressure drop calculation, used to determine the flow distribution, will tend to give the improper weighting of the temperature feedback effect in calculating frictional losses for large node spacings.³ 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, the FLODIS code was executed for a Fort St. Vrain DBDA analysis with two circulators operating at 8000 rpm. The first calculation contained 100 axial nodes per refueling region and the second calculation 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 maximum 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 maximum fuel temperature was 884°C (1623°F) and the maximum coolant temperature was 798°C (1469°F). The differences between the two cases now translate into a 9% difference for the maximum fuel temperature and a 7% difference for the maximum coolant temperature.

This investigation demonstrates the importance of evaluating modeling assumptions to ensure accuracy. Also, this study should be useful in the review of the RECA program.

A topical report describing the FLODIS code is currently being assembled. The report will contain the following information:

- I. INTRODUCTION
 - A. Description of the Fort St. Vrain HTGR
 - B. Capabilities of the Code
- II. MATHEMATICAL MODEL
 - A. Core Model
 - B. Flow Distribution Model

III. CORRELATIONS

- A. Heat Transfer Correlations
- B. Fluid Flow Correlations
- C. Useful Design Evaluation Correlations

IV. MATERIAL PROPERTIES

- A. Physical Properties of Helium
- B. Physical Properties of Graphite and the Composite Core

V. FORT ST. VRAIN DBDA ANALYSIS

- A. Description of the Accident
- B. Results
- C. Conclusions

VI. APPENDICES

- A. Listing of Code
- B. Code Input
- C. Code Output

A rough draft of the report will be available for review sometime in February 1976.

IV. THE PERFORMANCE CALCULATION FOR THE CAHE (J. R. Tallackson)

In the evaluation of the performance characteristics of the core auxiliary heat exchanger, it is necessary to determine the performance characteristics of the tube bundle. A report describing the calculation techniques and the input parameters used in this evaluation has been prepared and has been submitted for review and publication. The abstract for this report (ORNL/TM-4952) is as follows:

An extended BASIC program, CACHE, has been written to calculate steady state heat exchange rates in the core auxiliary heat exchangers (CAHEs), designed to remove afterheat from high-temperature gas-cooled reactors (HTGRs). Computationally, these are unbaffled counterflow shell and tube heat exchangers. The computational method is straightforward. The exchanger is subdivided into a user selected number of lengthwise segments, heat exchange in each segment is calculated in sequence and summed. The program takes the temperature dependencies of all thermal conductivities, viscosities and heat capacities into account providing these are expressed algebraically. CACHE is easily adapted to compute steady-state heat exchange rates in any unbaffled counterflow exchanger. As now used, CACHE calculates heat removal from high temperature helium with liquid water.

A second program, FULTN, is described. FULTN computes the geometrical parameters required as input to CACHE. As reported herein, FULTN computes the internal dimensions of the Fulton Station CAHE. The two programs are chained to operate as one.

Complete user information is supplied. The basic equations, variable lists, annotated program lists, and sample output with explanatory notes are included.

REFERENCES

1. Letter from R. F. Walker, Vice President for Engineering and Planning of the Public Service Company of Colorado, to Mr. R. P. Denise, Assistant Director for Advanced Reactors, Division of Reactor Licensing, USNRC, dated December 15, 1975 (Docket No. 50-267).
2. J. P. Sanders, Monthly Progress Report for November 1975 for the HTGR Safety Studies for the Division of Technical Review, U.S. Nuclear Regulatory Commission, ORNL-TM-5205 (December 1975).
3. J. P. Sanders, Monthly Progress Report for September 1975 for the HTGR Safety Studies for the Division of Technical Review, U.S. Nuclear Regulatory Commission, ORNL-TM-5108 (October 1975).

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