**Summary Progress Report for Fiscal Year
1976 and the Transition Quarter
Describing Technical Assistance Work
for the Division of Systems Safety,
U.S. Nuclear Regulatory Commission**

Prepared for the
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Under Interagency Agreement ERDA 40-544-75

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Printed in the United States of America Available from
National Technical Information Service
U.S. Department of Commerce
5235 Port Royal Road, Springfield, Virginia 22161
Price: Printed Copy \$4.50, Microfiche \$3.00

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Contract No. W-7405-eng-26

Engineering Technology Division

**SUMMARY PROGRESS REPORT FOR FISCAL YEAR 1976
AND THE TRANSITION QUARTER DESCRIBING
TECHNICAL ASSISTANCE WORK FOR THE
DIVISION OF SYSTEMS SAFETY,
U.S. NUCLEAR REGULATORY COMMISSION**

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Manuscript Completed - January 3, 1977
Date Published - January 1977

Prepared for the
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Under Interagency Agreement ERDA 40-54 1-75

Prepared by the
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UNION CARBIDE CORPORATION
for the
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

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PREVIOUS PROGRESS REPORTS IN THIS SERIES

Type	For the period ending	Designation
Bimonthly	Jan. 31, 1973	ORNL/TM-4752
Bimonthly	Mar. 31, 1973	ORNL/TM-4753
Quarterly	June 30, 1973	ORNL/TM-4741
Quarterly	Sept. 30, 1973	ORNL/TM-4739
Quarterly	Dec. 31, 1973	ORNL/TM-4738
Annual	Dec. 31, 1973	ORNL-4975
Quarterly	Mar. 31, 1974	ORNL/TM-4736
Quarterly	June 30, 1974	ORNL/TM-4737
Quarterly	Sept. 30, 1974	ORNL/TM-4806
Quarterly	Dec. 31, 1974	ORNL/TM-5062
Quarterly	Mar. 31, 1975	ORNL/TM-5075
Quarterly	June 30, 1975	ORNL/TM-5120
Sesquiannual	June 30, 1975	ORNL-5183
Monthly	July 31, 1975	ORNL/TM-5061
Monthly	Aug. 31, 1975	ORNL/TM-5085
Monthly	Sept. 30, 1975	ORNL/TM-5108
Monthly	Oct. 31, 1975	ORNL/TM-5155
Monthly	Nov. 30, 1975	ORNL/TM-5205
Monthly	Dec. 31, 1975	ORNL/TM-5238
Monthly	Jan. 31, 1976	ORNL/TM-5268
Monthly	Feb. 29, 1976	ORNL/TM-5326
Monthly	Mar. 31, 1976	ORNL/NUREG/TM-11
Monthly	Apr. 30, 1976	ORNL/NUREG/TM-20
Monthly	May 31, 1976	ORNL/NUREG/TM-32
Monthly	June 30, 1976	ORNL/NUREG/TM-42
Monthly	July 31, 1976	ORNL/NUREG/TM-50
Monthly	Aug. 31, 1976	ORNL/NUREG/TM-54
Summary	Sept. 30, 1976	ORNL/NUREG-6

RELATED TOPICAL REPORTS

1. J. P. Sanders, W. D. Turner, S. J. Ball, R. M. DeVault, G. E. Giles, and D. D. Paul, *Evaluation of Thermal Response in Fort St. Vrain Reactor Primary System to a Design-Basis Depressurization Accident Followed by Cooling with Two Pelton Wheel Drives Operating at 7000 rpm*, ORNL/TM-5140 (December 1975).
2. R. L. Reid and J. P. Sanders, *INGRES: A Computer Code for the Rate of Air Ingress into an HTGR Following a Design-Basis Depressurization Accident*, ORNL/TM-5144 (December 1975).
3. S. K. Iskander, C. W. Collins, and J. P. Sanders, *Inelastic Analysis of Two Plates Under Deformation Dependent Loads*, ORNL/TM-5206 (February 1976).
4. J. R. Tallackson, *The Thermal Transport Properties of Helium, Helium-Air Mixtures, Water, and Tubing Steel Used in the CACHE Program to Compute HTGR Auxiliary Heat Exchanger Performance*, ORNL/TM-4931 (February 1976).
5. J. R. Tallackson, *CACHE - An Extended BASIC Program which Computes the Performance of Shell and Tube Heat Exchangers*, ORNL/TM-4952 (March 1976).
6. G. E. Giles, R. M. DeVault, W. D. Turner, and B. R. Becker, *HEXEREI - A Multichannel Heat Conduction Convection Code for Use in Transient Thermal Hydraulic Analysis of High-Temperature Gas-Cooled Reactors*, K/CSD/TM-1 (May 1976).
7. D. D. Paul, *FLODIS: A Computer Model to Determine the Flow Distribution and Thermal Response of the Fort St. Vrain Reactor*, ORNL/TM-5365 (June 1976).
8. R. L. Whaley and J. P. Sanders, *Thermal Response of Core and Central-cavity Components of a High-Temperature Gas-Cooled Reactor in the Absence of Forced Convection Coolant Flow*, ORNL/TM-5354 (September 1976).
9. D. D. Paul, *Depressurization Accident Analyses for the Fort St. Vrain Reactor*, ORNL/NUREG/TM-58 (December 1976).
10. I. I. Siman-Tov and W. D. Turner, *HEATUP - A Computer Program for the Thermal Analysis of a LOFC Accident in an HTGR*, ORNL/CSD/TM-15 (November 1976).
11. W. D. Turner, *HEATINGS - An IBM 360 Heat Conduction Program*, ORNL/CSD/TM-15 (in publica . . .

12. W. D. Turner and J. P. Sanders, *CCCM - A Single-Channel, Coupled, Conduction-Convection Model for Use in Thermal Analysis in High-Temperature Gas-Cooled Reactors*, ORNL/CSD/TH-23 (in preparation).
13. G. E. Giles, W. D. Turner, K. W. Childs, R. M. DeVault, and B. R. Becker, *HEXEREIS: Computer Program for the Transient Thermal Hydraulic Analysis of High-Temperature Gas-Cooled Reactors*, K/CSD/TH-3 (in preparation).

FOREWORD

This summary report replaces the fifteenth in a series of monthly reports describing technical assistance provided to the Division of Systems Safety of the USNRC. It summarizes the accomplishments made during FY 1976 (July 1, 1975 to June 30, 1976) and the transition quarter (July 1, 1976 to September 30, 1976) and thereby fulfills the requirements for an annual report covering this period. A complete list of all the regular progress reports and annual summaries, plus those topical reports completed during the summary period, is given at the front of this report.

This work was initiated in November 1972 at the request of the Gas-Cooled Reactors Branch of the Directorate of Licensing of the USAEC and consists in supplying technical assistance and making analyses in support of licensing procedures for high-temperature gas-cooled reactors (HTGRs). The work continued under this branch during the change from the USAEC to the USNRC. In July 1975, the management of the assistance contract was transferred to the Division of Systems Safety of the Office of Nuclear Reactor Regulation of the USNRC, and the work continued under this division through FY 1976 and the current transition quarter. Because of the elimination of attempts to commercialize the HTGR plant, beginning with the cancellation of the Summit Power Station in October 1975, support of this effort has been significantly reduced, and it will be terminated at the end of the transition quarter. This report, therefore, represents the final report in this series.

SUMMARY

1. Introduction

This is a summary report describing technical assistance provided to the Division of Systems Safety of the USNRC during FY 1976 plus the transition quarter; it also represents the final monthly report required for the work. This presentation reviews briefly the analytical methods that were developed during the course of the task. The features of these analytical methods are compared with methods used to perform similar analyses, and examples of the use of these methods are cited.

2. HEATUP - A Computer Code for the Thermal Analysis of an LOFC Accident in an HTGR

The HEATUP code was written to determine the temperatures developed in the core of a high-temperature gas-cooled reactor (HTGR) following a loss-of-forced-cooling (LOFC) accident coinciding with a reactor trip. No convective transport of heat is represented in the calculation. Through the use of the total facilities built into the code, it is possible to determine the temperatures developed in the various refueling regions and relate them to the fuel age (or exposure to irradiation).

3. HEATING5 - An IBM 360 Heat-Conduction Code

HEATING5 is the most recent modification of a general-purpose heat conduction code developed at Oak Ridge National Laboratory (ORNL). It formed the basis of several other specialized codes developed for this task. Modifications were made or features were added to HEATING5 to serve the needs of these specialized codes. To reflect this work, it is cited in this summary report.

4. CCCM - A Coupled Conduction-Convection Model for Core Thermal Analysis

The CCCM is a particular model developed using the facility of the HEATING5 code. It limits the geometry options to a two- or three-dimensional cylindrical coordinate system and then incorporates a coolant channel model along the z-axis. This coupled convection-conduction model can represent either a nominal or "hot" coolant channel in the HTGR core. With the use of the three-dimensional R-Z option, the effect of blocked coolant channels in the vicinity of the open coolant channel can be studied.

5. FLODIS - A Computer Model to Determine the Flow Distribution and Thermal Response of the Fort St. Vrain Reactor

To make use of the single-channel analysis provided by CCCM, it was necessary to specify the flow of coolant in the channel. During a transient situation in an HTGR core, the distribution of coolant flow among the various regions will depend upon the total amount of coolant flow and the temperatures in the core. The FLODIS code was written to provide this information. It began as a relatively simple, multichannel model with only one solid node represented per channel. When it was found that convection transport of heat down the channel was an important effect, the number of axial nodes was easily expanded to a maximum number of 100 per channel. When the effect of axial conduction was questioned, this feature was added to the computation. The final refinement was to represent the conduction between nodal points in planes perpendicular to the coolant flow. This was done using a rectangular mesh rather than the hexagonal mesh of similar codes. By placing four mesh points within each refueling region, intraregional effects could be represented. The configuration represented by FLODIS was specifically the geometry of the Fort St. Vrain core; FLODIS is not a general-purpose calculation.

6. HEXEREI2 Code Development

While FLODIS was developed to provide a quick response to specific questions about the Fort St. Vrain review procedure, a more general multi-channel, coupled conduction-convection program, HEXEREI2, was developed for the designs of large HTGRs. The core model, which was originally called HEXEREI, will generate the representation of the HTGR core for a range of design operating powers from the simple designation of overall core characteristics. The conductive relationships are expressed in terms of a hexagonal grid and rely on the numerical techniques represented by the subprograms of HEATING5. The accuracy of this representation of both the conductive and convective heat transfer characteristics of the core was evaluated by executing simplified problems whose solutions could be compared with analytical or numerical results from proven methods. HEXEREI was coupled with a group of subroutines representing the performance characteristics of the core auxiliary heat exchanger (CAHE) and the circulator of the core auxiliary cooling system (CACS) plus the subroutines required to determine the distribution of coolant flow in the core, and this total code is called HEXEREI2. Although the existence of the CACS component models makes the HEXEREI2 code directly applicable to current large commercial HTGR designs, it can be adapted for use in evaluating transients for the primary system of the Fort St. Vrain reactor.

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1. INTRODUCTION

J. P. Sanders

At the beginning of FY 1976, the assistance contract for the high-temperature gas-cooled reactor (HTGR) safety studies with the Office of Nuclear Reactor Regulation, USNRC, was transferred for administrative guidance from the Division of Reactor Licensing to the Division of Technical Review (DTR). After the transfer, some of the previous work in the development of general analytical techniques was continued, and some of the existing studies were completed; however, the general scope and direction of the work was altered to meet the needs of the DTR.

Later, in December 1975, the Office of Nuclear Reactor Regulation was extensively reorganized, and the function of the DTR was assigned to the Division of Systems Safety (DSS). The work continued under the guidance of personnel of that division through the remainder of the fiscal year plus the transition quarter.

Changes in the licensing status of HTGR plants also affected the course of the work during this reporting period. At the beginning of the period, preliminary safety analysis reports (PSARs) and requests for construction permits had been submitted for two plants - the Summit Power Station¹ of the Delmarva Power and Light Company (DP&L) and the Fulton Generating Station² of the Philadelphia Electric Company (PEC). In addition, various licensing topical reports (LTRs) were under review, and a generic PSAR,³ designated GASSAR-6, had been submitted. The Fort St. Vrain plant was in the final stages of license review before going to power. At the same time, there were indications that requests for construction permits would soon be submitted for several more HTGR units.

The overall picture changed drastically when the General Atomic Company (GAC) announced in October 1975 that the contract with DP&L for the Summit Power Station would be terminated. After this announcement, work directed toward the licensing of the Fulton Generating Station was also suspended, and plans for the other HTGR plants were discontinued or indefinitely postponed. The remaining licensing work related to the Fort St. Vrain plant and to the review of GASSAR-6 and the LTRs.

These changes in program objectives plus the changes in program guidance due to the reorganization of the Office of Nuclear Reactor Regulation have resulted in a continuing restructuring of the tasks reported in this summary. As a final consequence of the decreasing need for technical support, it was announced that the assistance contract would be terminated at the end of the transition quarter.

The final work under this program consisted mainly in an effort to complete and document the computational methods developed through the course of the entire study. This final report briefly summarizes those computational methods and cites the topical reports that describe them in detail. Where applicable, examples of the use of these methods are given as illustrative of the work performed during the reporting period.

2. HEATUP - A COMPUTER CODE FOR THE THERMAL ANALYSIS OF AN LOFC ACCIDENT IN AN HTGR

W. D. Turner

2.1 Introduction

In the safety evaluation of the transient response of active cores of HTGRs under abnormal operating conditions, it is important to obtain the transient temperature distribution following a loss-of-forced-cooling (LOFC) accident that is recognized as one that would lead to a maximum hypothetical fission-product release (MHFPR). Thus, the computer code HEATUP was designed specifically to study the problem of the transient accumulation of heat in the active core of an HTGR for use in the analysis of an MHFPR arising from a total LOFC accident and simultaneous reactor trip. It calculates the maximum temperature as a function of time as well as the fractions of active core where temperatures exceed specified values as a function of the age of the fuel in the various refueling regions. It was used in the thermal analysis of LOFC accidents involving both the Summit Power Station and the Fulton Generating Station.

The HEATUP code, a modification of the general, multidimensional computer program HEATING5, was designed for the thermal analysis of an LOFC accident in an HTGR. The code is designed to calculate the transient temperature distribution for two-dimensional RZ and three-dimensional R0Z models of the active core of an HTGR after an LOFC accident. The calculations assume there is no coolant flow in the active core during the transient. The thermal properties may be dependent on temperature and position. The effective thermal conductivity may also be anisotropic in the active core, and this function is calculated using input data on the fuel, graphite, and coolant. The power in the reactor is a function of the radial and axial power-peaking factors as well as time. HEATUP is designed to calculate the percentages of the active core where temperatures exceed three criteria, which are a function of the time the fuel has been in the reactor (or the fuel age). If the respective temperature criteria for each age group represent the same fuel condition, then the code is capable of computing the total percentage

of active core where the temperature exceeds the criterion describing the same condition of fuel in all age groups.

2.2 Basic Computation

The user's manual for the HEATUP code was completed during this reporting period. It includes (1) a description of the basic problem; (2) a review of results related to the choice of thermal properties, boundary conditions, and initial conditions; (3) a full description of the three-dimensional R0Z model and a limited one of the two-dimensional RZ model; (4) the special computations that were added to HEATING5 in the development of HEATUP; (5) input preparation; and (6) a sample problem consisting of the three-dimensional model of a postulated LOFC accident for the Fulton Generating Station.

2.3 Relationship to Other Codes

The computational methods used in the HEATUP code are the same as those used in the HEATING5 code.

HEATUP is similar to the CORCON code⁴ of GAC, but HEATUP permits a more detailed description of the HTGR core. HEATUP allows both two-dimensional RZ and three-dimensional R0Z models, whereas CORCON allows only two-dimensional RZ models. Also, HEATUP, as compared to CORCON, describes the fuel condition in more detail. CORCON yields the fraction of core above specified temperatures as a function of time, whereas HEATUP can depict in addition the fuel condition as a function of lifetime in the active core.

2.4 Documentation of Code

The report⁵ documenting the HEATUP code has been reviewed and should be distributed by the end of November 1976.

The FORTRAN source deck and related sample problems will be sent to the DSS for further distribution.

2.5 Example of Use

The HEATUP code is designed to analyze the specific problem of the transient accumulation of heat in the active core of an HTGR arising from a total LOFC accident and simultaneous reactor trip. Specific examples of its use are described in the HEATUP report.⁵

3. HEATING5 - AN IBM 360 HEAT-CONDUCTION CODE

W. D. Turner

3.1 Introduction

HEATING5, a modification of the generalized heat-conduction code HEATING3, is designed to solve steady-state and/or transient heat-conduction problems in one-, two-, or three-dimensional Cartesian or cylindrical coordinates or one-dimensional spherical coordinates. The thermal conductivity, density, and specific heat may be both spatially and temperature-dependent. The thermal conductivity may be anisotropic. Materials may undergo a change of phase. Heat-generation rates may be dependent on time, temperature, and position, and boundary temperatures may be dependent on time. The boundary conditions, which may be surface-to-boundary or surface-to-surface, may be fixed temperatures or any combination of prescribed heat flux, forced convection, natural convection, and radiation. These boundary-condition parameters may be dependent on time and/or temperature. The mesh spacing can be variable along each axis. The code is designed to allow a maximum of 100 regions, 50 materials, and 50 boundary conditions. The maximum number of lattice points can be easily adjusted to fit the problem and the available computer memory. The requirements on an IBM 360 machine range from approximately 250,000 bytes for one lattice point to 1,256,000 bytes for 6000 lattice points.

3.2 Basic Computation

The development of the HEATING5 code was partially funded by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. Some work on development of the code was continued during this reporting period, but most of the work was directed toward documentation of the code.

3.3 Relationship to Other Codes

The point successive-overrelaxation-iterative method and a modification of the "Aitken ² extrapolation process" are used to solve the finite-difference equations, which approximate the partial differential equations for a steady-state problem.

The transient problem may be solved using any one of several finite-difference methods. These include (1) an implicit technique that can range from Crank-Nicholson to the classical implicit procedure, (2) an explicit method that is stable for a time step of any size, and (3) the classical explicit procedure that involves the first forward time difference. The solution of the system of equations arising from the implicit technique is accomplished by point successive-overrelaxation iteration and includes procedures to estimate the optimum acceleration parameter. Transient problems involving materials with change-of-phase capability cannot be solved using the implicit technique with this version of HEATING5.

HEATING5 writes a data set that is used by the HEATPLOT computer code to generate plots containing isotherms and temperature profiles. HEATPLOT is a general plotting program that allows the user considerable flexibility in generating various plots of results from the heat-transfer calculations.

The HEATING5 code is similar to many other general heat-transfer codes. Some of the other codes and the laboratories where they were written are: TAC2D and TAC3D from General Atomic Company; TRUMP from Lawrence Livermore Laboratories; SAHARA from Sandia Laboratories, and TAP-B and TRACK from Westinghouse Astronuclear Laboratory. Most of the general heat-transfer codes are similar in nature. Some features of HEATING5 that may not be available in other general heat-transfer codes are (1) a nodal point generator; (2) independent solution techniques for transient and steady-state problems; (3) several techniques, both explicit and implicit, for solving transient problems; (4) a variety of boundary conditions; (5) nonlinear problems to include the source term; and (5) an output map of the temperature distribution. Features found in some of the codes mentioned above that are not available in HEATING5 are (1) the ability to model problems other than those described in Cartesian, cylindrical, and spherical coordinate systems; (2) coupled mass-flow calculations; (3) material property

libraries; (4) free-form input; and (5) input data specifically designed to facilitate parametric studies.

3.4 Documentation of Code

The report^c documenting the HEATING5 code, which is in the final stages of preparation, should be ready for publication in late 1976.

The FORTRAN source deck and related sample problems will be placed in the Argonne Code Center.

3.5 Example of Use

The HEATING5 computer code may be used for a wide variety of diffusion-type problems that can be modeled in Cartesian, cylindrical, or spherical coordinates.

4. CCCM - A COUPLED CONDUCTION-CONVECTION MODEL FOR CORE THERMAL ANALYSIS

W. D. Turner

4.1 Introduction

The coupled conduction-convection model (CCCM) computer code, a modification of the general heat-transfer code HEATING5, has been especially modified to provide localized temperatures in the fuel, graphite, and coolant channel of an HTGR during normal operation as well as during accident conditions involving reduced flow and loss of forced cooling. It is designed to calculate the steady-state and transient temperature distributions for two-dimensional RZ and three-dimensional R6Z models coupled to a single, one-dimensional, axial coolant channel. In addition to the input data required by HEATING5, CCCM requires the following information about the coolant: inlet temperature, mass velocity, composition, and pressure. It also requires certain information concerning the coolant channel for the HTGR in question.

4.2 Basic Computation

A few minor refinements were made to CCCM during the reporting period. During the first quarter, the calculations concerning the evaluation of the thermal response in the Fort St. Vrain Reactor arising from a design-basis depressurization accident (DBDA) followed by cooling with two Pelton wheel drives were completed. A number of cases were studied using the CCCM code, and the results are presented in ORNL/TM-5140 (Ref. 7).

4.3 Relationship to Other Codes

The CCCM code is a modification of the HEATING5 computer code and, as such, uses the same numerical techniques for the heat-conduction analysis. The convection calculations are based on a one-dimensional axial model, where the coolant temperature is based on the mass velocity, the upstream temperature, and the heat flow between the solid and coolant. The heat

flow between the solid and coolant is a function of the flow regime and the channel configuration.

The CCCM code requires input information concerning the coolant, which can be obtained from codes such as FLODIS.

4.4 Documentation of Code

The CCCM code has been documented in a report⁸ currently in publication. The FORTRAN source deck and related sample problems will be transmitted to DSS for further distribution.

4.5 Example of Use

The CCCM computer code can be used to study localized temperatures within the core of an HTGR for a variety of situations, provided accurate inlet coolant information is available. Some examples of situations where it can be used are: normal operation; scram of the reactor; accidents such as design-basis depressurization and loss of forced cooling; blocked coolant channels; variations in power, inlet coolant temperature, and inlet mass velocity; and closed orifices.

5. FLODIS - A COMPUTER MODEL TO DETERMINE THE
FLOW DISTRIBUTION AND THERMAL RESPONSE
OF THE FORT ST. VRAIN REACTOR

D. D. Paul

In February 1975, the Nuclear Regulatory Commission asked Oak Ridge National Laboratory to perform an independent analysis of the thermal response of the Fort St. Vrain reactor core and primary coolant system following a postulated design-basis depressurization accident (DBDA). The FLODIS code was developed in response to that request and has since been revised several times.

FLODIS is a combined fluid-flow and heat-transfer analysis calculation written specifically for the core of the Fort St. Vrain reactor. It is a lumped-node representation of the 37 refueling regions in the active core, the side reflector blocks, and the prestressed concrete reactor vessel (PCRVR). Features of the FLODIS code include models to represent (1) inter-regional conduction of heat, (2) both forced and natural convection flows, (3) intraregional distribution of coolant flow, and (4) the PCRVR-liner cooling system. The calculation uses the specified operating conditions for the reactor at power to determine appropriate loss coefficients for the variable orifices in each refueling region. Flow distributions following reactor trip and a reduction in coolant pressure and flow are determined assuming that the orifice coefficients remain constant. Iterative techniques are used to determine the distribution of coolant flow as a function of time during the transient.

The latest version of FLODIS uses a rectangular grid to model the entire core of the Fort St. Vrain reactor. A typical refueling region is approximated by four rectangular mesh spaces. Heat is allowed to flow between all adjacent mesh spaces. Note that all current models used to represent multiple regions in HTGR cores (HEXEREI, ORECA, RECA) are based on hexagonal grid spacing. Results from the rectangular model will provide a valuable check of the shape factors used in models with hexagonal grid spacing.

The flow distribution model has been expanded to include both forced and natural convection flow. Concurrent with this improvement, the flow is allowed to redistribute itself within a refueling region according to

the finer mesh spacing. Intraregional distribution of coolant flow is not represented by HEXEREI, ORECA, or RECA. This effect is important during some postulated accident sequences where there are large temperature gradients across a refueling region.

Finally, the core model has been expanded to include the side reflector, gas annulus, and PCRV-liner cooling system. These improvements were necessary to determine the capability of the PCRV cooling system to remove afterheat during some postulated accident sequences.

Figure 5.1 shows the results of a DBDA analysis using the FLODIS code. Also shown on the same graph are results reported by GAC using their RECA code.

The FLODIS code was documented in a topical report⁹ issued in June 1976. Subsequent modifications of the code are described in a more recent document,¹⁰ which also presents the results of analyses of certain specified DBDAs for the Fort St. Vrain reactor.

A copy of the FORTRAN symbolic program plus a set of sample input was prepared for submission to the Argonne Code Center. The symbolic language used is compatible with the IBM FORTRAN IV, level H compiler, release 21.6. A sample set of the job control language (JCL) for the HASP system on the IBM 360, model 91 was also supplied.

ORNL DRG 76-1852

Fig. 5.1. Comparison of FLODIS and RECA temperatures during a postulated Fort St. Vrain DBDA with 7000 rpm circulator speed.

6. HEXEREI2 CODE DEVELOPMENT

G. E. Giles K. W. Childs

6.1 Introduction

HEXEREI2 is a three-dimensional, coupled, conduction-convection heat-transfer and multichannel fluid dynamic analysis computer code for both steady-state and transient problems. The program was developed to provide thermal fluid dynamic analysis of high-temperature gas-cooled reactors (HTGRs) to satisfy licensing procedures for the Nuclear Regulatory Commission (NRC).

The development of HEXEREI2 was commissioned by the NRC to provide an analytical capability that was independent of those used by the reactor vendor and was not available at Oak Ridge or other sites. The General Atomic Company (GAC) is the only active HTGR vendor in the United States, and its designs represent a departure from both those of the European gas-cooled reactors and the normal light-water reactors. The development of HEXEREI2 was restricted to the current commercial HTGR design of GAC and was initiated in June 1974, with the goal of producing a code with safety analysis capability. Improvements were possible on GAC's RECA technique,¹¹ since the project was starting with a previously developed code⁶ (HEATING5) and the developers had the benefit of recognizing the deficiencies of existing codes. Another major goal was to develop a code that would be compatible with other computer systems, thus allowing NRC personnel to use it on other systems to develop expertise and confidence in the safety analysis of HTGRs.

The gross model of the HTGR is a right circular cylinder with fuel of nonuniform characteristics dispersed regularly throughout an inner region (a right circular cylinder shorter and of smaller radius than the model). Coolant flow is distributed in a large number of axial coolant channels extending from the top to the bottom of the core model. Thus, heat transfer in the HTGR core involves both three-dimensional heat conduction and convection to the axially flowing coolant in the coolant channels. When the coolant flow is large, convective heat transfer is the significant process and will tend to produce temperatures that are uniform in the axial plane

in each refueling region. This is due to (1) the uniformity of heat generation in the refueling regions; (2) the uniform distribution of a large number of small coolant channels, which will have approximately the same coolant mass flow; and (3) the high mass flow, which tends to reduce the thermal gradients in the solid in the axial plane and control the heat transfer in the axial direction.

As the mass flow changes, the relative resistance to flow in each channel changes due to buoyancy and to losses from friction, acceleration, expansion, and contraction. The mass flow through the core may redistribute itself among the coolant channels, since the change in resistance is not uniform throughout the core. Initial temperature, mass flow, and differences in orifice settings contribute to the nonuniformity of change in resistance. If the flow is reduced to a value near zero, the buoyant forces produced in the hot channel could cause stagnation and perhaps upward flow in the channel (reverse flow). Also, when the flow is substantially reduced, heat conduction across the core becomes more important. When the net flow through the core is zero, natural convection loops can develop in the core and, over long periods, will redistribute the heat until the temperatures are uniform throughout the core. During an accident transient, the coolant flow may change from full flow to zero or any intermediate level. During full flow, the conductive heat transfer is very small relative to the convective heat transfer. As the flow decreases, conductive heat transfer becomes more important. Also, reversed (upward) flow becomes possible as the flow decreases. Thus, the relative significance of the physical effects may change with time during the transient. Accurate heat conduction, heat convection, and fluid dynamic models for all possible heat-transfer paths that may occur during a transient are required for any code that is to be utilized for HTGR accident analysis. Such an analysis technique, dealing with (1) the core, (2) the upper, lower, and side reflectors, and (3) the upper and lower plenums, is included in the HEXEREI code.

In the large HTGR design, helium normally flows downward through the fuel element coolant channels, the cracks around the fuel elements, the control rods, and the side reflector elements and into the lower plenum through collector ducts; then it goes up through the core auxiliary cooling

system (CACS) heat exchanger bundle and the CACS circulator and from there into the upper plenum to return to the core.

The analysis for the total loop is included in HEXEREI2. HEXEREI2 contains HEXEREI merged with a package of routines (ACHERONS) that includes the modeling for the CACS portion of the loop. Thus, HEXEREI2 has the capability of analyzing accurately the three-dimensional, heat conduction-convection problem for flows from zero to the maximum values obtained in the closed-loop system. The external boundary conditions for the closed system are the water inlet temperature, flows into the CACS, and circulator performance characteristics.

HEXEREI2 can perform the above analysis for the Fort St. Vrain HTGR as well as all HTGRs that are similar to the 2000- and 3000-MW(t) CAC designs. Since the Fort St. Vrain HTGR does not contain an independent CACS loop, either additional input information must be obtained or minor modifications must be made in the program to represent the FSV loop.

HEXEREI contains a very general mathematical package for the solution of the solid and coolant temperatures. The user has the option of selecting a steady-state technique, an explicit transient technique that is normally too expensive for the general three-dimensional problem, or a very powerful implicit transient technique. The implicit transient technique is capable of considerably reducing the cost of a computation. However, for problems requiring very small time steps, the explicit technique is less expensive than the implicit. The treatment of the boundary condition, material properties, and heat-generation rate is very general in HEXEREI2 and allows the user great flexibility in modeling. However, the hexagonal-mesh generator, the coolant temperature and flow redistribution techniques, the coolant channel description, and the coolant system are somewhat limited to HTGR-type problems. HEXEREI can analyze problems other than heat transfer in an HTGR, but since this development was not within the scope of the project, modifications to the code may be required.

HEXEREI2 will automatically generate the mesh geometry specified by the user within some fairly broad-based constraints. A wide range of possible HTGRs is included within the mesh-generation capabilities of the code. The user can thus modify the model under analysis by changing a few simple geometric input values.

A companion code, HEXPLT, was developed to enable the user to obtain graphic presentation of information pertinent to HTGR analysis.

6.2 Relationship to Other Codes

During the safety studies of HTGR designs of GAC, various ORNL computer codes have been evaluated. One of the first utilized at Oak Ridge for HTGR analysis was HEATING5 (or HEATING4, an earlier version). HEATING5 is a very general, one-, two-, or three-dimensional (either Cartesian or cylindrical coordinate system) heat-conduction analysis technique for steady-state or transient problems. General mathematical techniques, boundary conditions, and material properties make HEATING5 a very versatile code. However, it requires all model boundaries (external and internal) to be parallel to the coordinate axes; thus, an HTGR core would have to be modeled in Cartesian or cylindrical coordinates. It was found from experience with HEATING5 and also from GAC's code development experience that the finest Cartesian or cylindrical grid structure capable of fitting on presently available computers could not perform an adequate analysis due to the lumping of several refueling regions into one cell. Since the thermal properties, age of the fuel, and power are fairly uniform within the refueling regions, it would be desirable to have at least one column of nodes per refueling region. If this criterion is applied to HEATING5 mesh generation, the refueling regions near the core center have many nodes per region. (The cylindrical mesh is discussed here to avoid having to stair-step the boundaries of the model.) This surplus of nodes is more than an annoyance, since the small spacing reduces the maximum time step that can be used in the transient analysis. The surplus also increases the computational work per time step and the core storage requirement beyond current computer capacity. HEATING5 also lacks the ability to model the flow in a multichannel system or to couple the conduction solution to the convection solution so that the temperature of the convecting fluid is a function of the heat transferred into the fluid.

HEXEREI2 was developed from HEATING5 and retained HEATING5's major structure and some of its compatible ancillary routines.

CCCM (Ref. 8) was developed from HEATING5 (prior to the HEXEREI2 development) to provide the capability of analyzing the steady-state and transient conduction in two or three dimensions coupled with convection to a single-channel, axially flowing coolant along the outer boundary of the model. The coolant temperature is modified by the heat input over the channel length. Single-channel analyses of the HTGR are thus possible with the CCCM code and have been performed extensively.⁷ Multichannel coolant flow, a more detailed model of the total core that is adaptable to present computers, and the analysis of the CACS coupled with the core are still not possible with CCCM. The analytical technique of CCCM is similar to the single-average-channel representation in the TAP code¹² of GAC. TAP, however, includes a representation of the primary and secondary coolant systems in its analytical technique.

The single-channel analyses of the HTGR core with TAP or CCCM neglect the effects of interregional heat flow and the distribution coolant flow among the axial channels due to high thermal gradients and low flows. These two effects become very important during accidents involving reactor trip, loss of primary coolant pressure, or loss of primary coolant flow with restart of flow using the CACS.

While HEXEREI2 was in the early stages of development, work was started by Paul on the FLODIS code.⁹ In its earliest version, FLODIS was used in conjunction with CCCM to provide bounding values for analysis of specified transients in the license review for the Fort St. Vrain reactor. Since the fluid dynamic problem was a function of the solid temperatures, FLODIS at first included a one-node representation of each refueling region. This essentially incorporated the heat storage of the graphite into the coolant temperature calculations and enabled FLODIS to perform heat-transfer analyses that overestimated the temperatures developed in the core.

FLODIS provided a convenient starting point for the development of the flow distribution subroutines that were used in HEXEREI2. Significant development was required to provide a set of subroutines that would interchange information with the existing HEXEREI calculations. In this manner, the pressure-loss models in HEXEREI2 profited from FLODIS's prior development. FLODIS also provided a convenient output display, which was utilized in HEXEREI2.

CAC uses a complex system of routines called RECA (Ref. 11) to analyze HTGR accidents. RECA uses a nodal system which, in the axial plane, consists of one node for each refueling region and one or more nodes in the side reflector. Each node is connected (heat is allowed to flow) to each of six neighboring nodes (refueling regions) in the same axial plane and to one node in each of the adjacent axial planes. This nodal system provides a sufficiently detailed analysis within the capability of existing computers. RECA also provides a coupled coolant flow and temperature analysis capability for the HTGR core with a CACS. It is the code whose analytical capability HEXEREI2 was designed to duplicate. However, HEXEREI2 is more independent of computer hardware than is RECA, and the developmental work of HEXEREI2 has attempted to improve the efficiency of the mathematical techniques available in RECA and the accuracy in some areas of RECA's modeling, such as the CACS heat exchanger.

In order to evaluate RECA, Ball developed the ORECA code.¹² The basic model used in ORECA is similar to that used in RECA and HEXEREI2, and the basic equations (fluid dynamic and heat transfer) solved by ORECA are similar to those solved by HEXEREI2. ORECA makes many assumptions in the approximations, solution of equations, and modeling that permit very inexpensive (short running time) analysis. The code requires relatively small amounts of computer memory, and changes can be made fairly easily. However, significant changes in model geometry, material properties, heat generation, or boundary conditions require coding modifications. The computational times reported¹³ for ORECA indicate that assumptions have been made in order to reduce the computational time. Simplifications in a very complex, three-dimensional, transient, nonlinear problem should be made with extreme care. There is no assurance that the assumptions in the code that appear to work well in one set of cases (i.e., in comparison to other codes) will function correctly in another set.

6.3 Documentation of Code

1. *HEXEREI, A Multichannel Heat Conduction Convection Code for Use in Transient Thermal Hydraulic Analysis of High Temperature Gas-cooled Reactors*, G. E. Giles, R. M. DeVault, W. D. Turner, and B. R. Becker, K/CSD/TM-1 (May 1976).

This report contains interim documentation of HEXEREI's performance in some sample heat-transfer problems and includes comparisons with analytical and/or proven computer code results. The test cases were chosen to demonstrate HEXEREI's performance in each portion of an HTGR analysis (i.e., both radial steady-state or transient conduction and convection to a coolant).

2. *Evaluation of Thermal Response in Fort St. Vrain Reactor Primary System to a Design-Basis Depressurization Accident Followed by Cooling with Two Pelton Wheel Drives Operated at 7000 rpm*, J. P. Sanders et al., ORNL/TM-5140 (December 1975).

The above report includes documentation of ACHERONS performance calculations for the Fort St. Vrain steam generator and circulator. ACHERONS was used in a stand-alone mode (without HEXEREI) with GAC-supplied inlet values to evaluate the performance of the steam generator and circulator.

3. Letter dated April 22, 1976, from G. E. Giles to J. P. Sanders. Subject: "Fort St. Vrain Safety Analysis with the Improved HEXEREI Code." (Copy transmitted in a letter dated April 28, 1976, from J. P. Sanders to W. Gilbert, Project Manager, Special Reactors Branch, USNRC.)

The above letter contains the results of a HEXEREI analysis (steam generator and circulator not included in the model) of the behavior of the Fort St. Vrain reactor after reactor trips from both 40 and 105% power. The analysis of the 105% power accident (DBDA2) used the same information that was used in the evaluation reported in item 2, above. A detailed comparison of the HEXEREI solution and the GAC-supplied steady-state solution for the 40% power case was made. A discussion with GAC personnel was prompted by this analysis and yielded some experimental data for the orifice pressure loss, which will be included in HEXEREI. A summary of this information was included in ORNL/NUREG/TM-20, *Monthly Progress Report for April 1976 for the HTGR Safety Studies for the Division of Systems Safety, U.S. Nuclear Regulatory Commission*.

4. *HEXEREI2: Computer Program for the Transient Thermal-Hydraulic Analysis of High-Temperature Gas-Cooled Reactors*, K/CSD/TM-3.

This report is the final report for HEXEREI2 code development and a fairly complete user's manual modeled after the HEATING5 user's manual. Discussions of the mathematical techniques, modeling assumption, program

execution options, and input data are included in the document. Discussions of nine test cases and comparisons with analytical results are presented to illustrate the code's capabilities; some of this information is taken from K/CSD/TM-1. Also included is a series of HTGR and other analyses to aid the user in modeling his particular problem or in utilizing the full power of HEXEREI2.

6.4 Examples of HEXEREI2 Analysis of HTGR

The analyses listed below were performed with the latest HEXEREI2 code. They were done to document the kind of information available from HEXEREI2 and to permit comparison of the results with those from the RECA code of GAC. The RECA results for the DBDA2 analysis were reported in Ref. 14, and those for the 40% power trip with cooldown using four circulators for 15 min were reported in Ref. 15.

6.4.1 DBDA2 analysis

Figure 6.1 presents the comparison of results from HEXEREI2 with those from RECA (curves marked GAC) for the maximum solid and average exit temperatures. This analysis by HEXEREI2 includes the new orifice pressure-drop correlation based on experimental correlations¹⁶ provided by GAC. HEXEREI2 calculates an average coolant exit temperature that is consistently lower than that calculated by GAC throughout the transient. The reason for the exit temperature differences cannot be determined at present. Since both analyses (HEXEREI2 and GAC) use the same coolant inlet conditions, HEXEREI2 will remove less heat from the active core. The high maximum solid temperature of HEXEREI2 may be due to the decreased heat removal. The agreement in the core pressure loss of the two analyses is shown in Fig. 6.2.

6.4.2 Analysis of 40% power reactor trip

Figure 6.3 presents a comparison of results from HEXEREI2 and GAC for the analysis originally requested in Ref. 14. The agreement between the analyses is not as good as that shown for the DBDA2 analyses. The core pressure drop for HEXEREI2 (not shown) was a consistent 0.28 kPa (0.04 psi)

Fig. 6.1. Comparison of HEXERE12 results for DBDA2 with reactor trip from 105% power with results provided by General Atomic Company (GAC).

Fig. 6.2. Core pressure drop for HEXERE12 results for DBDA2 compared with values given by General Atomic Company (GAC).

Fig. 6.3. Comparison of General Atomic Company (GAC) and HEXERE12 results for Fort St. Vrain reactor trip from 40% power with four circulators operating for 15 min (includes most recent orifice correlation coefficients from GAC).

lower than that for GAC. In the discussions with GAC in May, it was revealed that several of the coolant orifices were completely closed and all the flow was through cracks in the upper reflector elements. Since HEXERE12 forces all coolant exit temperatures to be equal for the steady-state analysis and was able to achieve this status without closing down the orifices, some other criterion for orifice settings must have been used by GAC. Once

the steady-state orifice-control functions used by GAC are simulated in HEXEREI2, the resulting transient analysis should agree with the GAC analysis just as well as did the analyses for the DBDA2.

7. RECOMMENDATIONS FOR FUTURE WORK

J. P. Sanders

The need for future work in the licensing of HTGR plants falls into two categories. First, and most immediate, are the analytical needs relative to the review of the Fort St. Vrain (FSV) operation during the rise-to-power sequence and the need to review or modify the technical specifications for this reactor design. Second, there is the need to review the design for large commercial plants - in particular, the design described in the CASSAR-6 document³ and the related licensing topical reports.

7.1 Review of Fort St. Vrain Operation

It should be remembered that, at present, a single computational procedure to analyze the total detailed transient response of the core and the total primary loop for the FSV reactor does not exist, neither within the methods developed at ORNL under this task nor within the methods disclosed by GAC. As an obvious consequence, there is no representation of the balance of plant (BOP) that is coupled with the detailed primary loop analysis. All such analyses must be done in a piecemeal manner, either by GAC or by use of the computational methods developed under this task.

GAC has a version of the TAP code¹² that represents the BOP plus the primary circuit of the FSV reactor; however, the core representation in this model is that of a single, nominal channel. At zero or low flows, this single-channel model is not capable of representing redistribution of flow or flow reversals, both of which have important ramifications concerning the safety evaluation of abnormal transients. To obtain the required detail, GAC must use output from the TAP calculation as input to a RECA calculation.¹¹ An obvious implication of the procedure is that there can be no feedback from the detailed RECA analysis to the TAP calculation. Since the RECA calculation represents more of the detailed flow paths in the primary circuit, the match of TAP output and RECA input is not exact. Thus, there is a need for a single computation

that includes both the detail of the primary loop and sufficient detail of the BOP to perform a single analysis of the transient.

In the work performed for the Office of Nuclear Reactor Regulation at ORNL, our efforts have been directed primarily toward "abnormal transients" in large HTGRs, and only in the last 12 months has the primary focus of our work changed to the analysis of FSV transients. For the large reactor designs, the abnormal transients cited were initiated after the loss of the main loop cooling. Thus, our work was directed toward the analysis of the core in conjunction with the core auxiliary cooling system (CACS). The present HEXEREI2 code¹⁷ represents the ultimate development in analytical methods for these transients. It is adaptable to cores of various power levels with the same general primary circuit configuration as shown in the GASSAR-6 document³ for the analysis of any accident involving loss of cooling in the main loop.

HEXEREI2 can also be used to evaluate the detailed performance of the FSV core and primary loop during abnormal transients; however, additional time-dependent information such as the fraction of the steam generator bundle that is flooded plus information on circulator operation must be provided. At present, it will not perform these analyses as a stand-alone computation.

The FLODIS-CCCM combination,⁷⁻⁹ and more recently the expanded FLODIS calculation,¹⁰ can perform detailed thermal-hydraulic analyses for the FSV core plus the surroundings. Input detailing the total net flow into this cavity and the inlet temperature as a function of time during the transient is required for these calculations. Certain assumptions that are used to determine the initial temperature distribution in the cavity and core are built into the calculations; these temperatures constitute the initial conditions on which the transient temperatures are based. Thus, FLODIS performs the calculation for the FSV core in somewhat the same manner that GAC uses in its RECA code. However, the FLODIS representation of the core is more detailed and realistic and should produce values for fuel and outlet gas temperatures that are more useful for licensing analysis. Although RECA is a conservative model for some transients, it appears that it will not be conservative for other postulated accident sequences.

In June 1976, correlations were provided by GAC based on experimental operation of the test for FSV orifice assembly. Prior to that time, general correlations for circular orifice plates, which obviously had limited application, had been used. The new correlations have been substituted in the flow calculations for both the HEXEREI2 code¹⁷ and the FLODIS code.¹⁰ When comparisons of results from HEXEREI2 and FLODIS with those from the RECA calculation are possible, good agreement is found in the evaluation of overall pressure loss across the core for significant flows, but the distribution of flow produced when "nearly closed" orifices are required does not agree.

It should be noted that mechanical movement of the orifice device can completely cover the rectangular orifice openings. A minimum flow area is provided by the clearances between the various parts of the assembly and between the upper plenum elements.

Current calculations differ for the partial refueling regions at the core-reflector boundary. These partial regions, which have less than seven fuel columns, have the same orifice mechanism as the complete region. It is obvious that these partial regions will always be operated with the orifice mechanism in a more closed position as compared with the complete region. In fact, in many cases the orifice mechanism will be in the fully closed position and still will not restrict the flow sufficiently to produce an outlet temperature from the region equal to the mean exit coolant temperature.

Again, the HEXEREI2 and FLODIS calculations do not correspond to the RECA results. It is possible that in the fully closed position, a different representation of the "orifice loss" is required.

7.2 Review of Large HTGR Performance

HEXEREI2 can be used to evaluate all abnormal transients for large HTGR designs in which there is immediate loss of main-loop cooling (LOMLC). The core representation is "built up" by the program from overall parameters such as total core power, fuel element size, and total core dimensions. It is assumed that the reactor is brought to sub-criticality simultaneously with the LOMLC; however, the initiation of

CACS operation can be delayed for a specified period. Certain information, such as the inlet temperature and flow of coolant water to the CACS heat exchanger, must be specified.

The performance limitations of the electric motor drive for the CACS circulator are built into the code and must be changed internally if these features are altered. Control features that will regulate the circulator speed to maintain an outlet water temperature from the CACS heat exchanger of 478 K (400°F) or less are built into the code. The required circulator motor speed, torque, and power are indicated.

A feature that is not included in the HEXEREI2 code¹⁷ at present is the internal ability to calculate the rate of air ingress following a depressurization accident. This rate of air ingress is calculated by a separate computation called INGRES.¹⁸ The calculation could be run separately, and provisions are made in HEXEREI2 to specify the effective molecular weight of the coolant as a function of time during a depressurization accident. The INGRES calculation could be added as a subroutine in the HEXEREI2 computation with relatively little additional effort.

Since most of the ultimate licensing questions involve the evaluation of protection of the public, it is the performance of the CACS system that must be investigated to resolve most questions of this type. Operation of the main-loop cooling systems for any period of time following reactor trip will result in lower initial temperatures and smaller potential heat sources than will be found in those analyses depending only upon heat removal by the CACS. It can be assumed in those cases that thermal transients will be less severe.

The HEATUP code⁵ determines the thermal conditions for the ultimate accident when all forced convective cooling is lost during the heat-up period. The HEATUP calculation implies that depressurization has occurred, since the transport of heat by natural convection currents within the core cavity is neglected. As noted in this report, the HEATUP code can associate the calculated temperature of the fuel with the age of the fuel (i.e., its exposure to irradiation). Generally, the older fuel will be operating at lower power densities and, therefore, will heat up more slowly than the newer fuel. The irradiated fuel does fail at lower temperatures. Heat removal from the perimeter of the core is neglected in

this calculation, since for large HTGR cores, sublimation temperatures for graphite are reached before significant amounts of heat are removed from the perimeter. The specified 20 to 10% uncertainty in afterheat generation is not incorporated in this code and should be added.

7.3 Computational Techniques Available from Other Codes

During the same period in which the above work was being done, parallel work concerning the safety of HTGR designs was initiated at ORNL under sponsorship of the Division of Reactor Safety Research of the U.S. Nuclear Regulatory Commission. One of the initial objectives of that program was to evaluate computational methods for safety analysis developed by GAC. During the interval between the assignment of the task and the transmittal by GAC of detailed listings and information about the computer programs, members of the staff of ORNL undertook to develop similar computational methods from the general descriptions of the codes that existed in the open literature. The purpose of this effort was to add to our overall understanding of the problems involved in the development of these computational methods and to have known methods available for use in the execution of comparison cases.

In February 1976, representatives of the Division of Systems Safety made a direct request to the Chief of the Experimental Gas-Cooled Reactor Safety Research Branch of the Division of Reactor Safety Research to have this effort at ORNL directed toward the development of a specific total-system performance code for the Fort St. Vrain reactor. In response to this request, a code called ORTAP, which was written to perform the same type of computations as the TAP code of GAC, was modified to represent the primary circuit and BOP for the FSV plant. This code includes a single-channel model for the core [CORTAP (Ref. 19)] that is coupled with point neutronic calculations; it also includes detailed representations of the steam generator [BLAST (Ref. 20)], the main helium circulator, and the BOP.

At the same time, the ORECA code¹³ was developed specifically to model the core of the FSV reactor in a manner similar to characteristics described for RECA. The task of coupling ORECA with ORTAP to provide a

complete calculation for all transients is yet to be completed. It should be remembered that FSV, unlike the large HTGRs, does not have an auxiliary cooling loop. FSV depends upon the use of an auxiliary drive for the main circulators plus the flooding of the steam generators to remove the afterheat in an emergency situation.

Another important point is that ORTAP and ORECA were developed as best-estimate calculations with the original purpose of providing comparison calculations for the corresponding GAC codes. This method of development differs from the procedures used to develop computational methods for the licensing division. The latter procedures were developed in such a manner that the results would always be conservative, and they were also compared, when possible, with other computational techniques at each stage of development.

Additional work is required for the ORECA code to include the modified orifice correlations that were provided by GAC in June 1976.

A comparison of calculated primary loop flows for the DBDA in the FSV with reactor trip from 40% power identified two bypass streams that are included in the TAP calculation but not in any ORNL calculations. These bypass flows occur at the bottom of the core barrel and at the lower support floor. The first flow bypasses the core, and the second bypasses both the core and the steam generator. Drawings giving physical dimensions of these structures were provided by GAC in June 1976; these dimensions will be used to estimate the amount of bypass flow, and the information will be added to ORTAP.

The GAC-FSV version of TAP also represents the heat lost to the PCRV liner cooling system as the gas returns to the core inlet from the circulator discharge. This effect is not represented in the current version of ORTAP.

The current plan is to obtain operational data from the FSV plant for the verification of both ORTAP and ORECA, in total and for individual parts, for all transients that can reasonably be executed. In the interim, comparison cases can be executed using ORECA, HEXEREI2, and the expanded version of FLODIS.

7.4 Conclusions

1. For current detailed analysis of thermal conditions in the FSV core during low-flow conditions, either FLODIS or HEXEREI2 can be used as the basis for licensing decisions.

2. HEXEREI2 can be used to evaluate FSV primary loop performance only if additional transient information is available.

3. HEXEREI2 can be used for LOMLC and coincident reactor trip for large HTGR designs, except for depressurization accidents. For these accidents, INGRES must be executed separately and the results entered into HEXEREI2. These two calculations should be merged into one program.

4. Additional modifications of ORECA are required to incorporate the experimental orifice coefficient relationships provided by GAC.

5. Modifications of the present version of ORTAP are required to represent the two bypass streams and the heat loss from the circulating gas to the PCRV-liner cooling system.

6. ORECA and ORTAP can be executed for FSV transient analysis on a best-estimate basis. These two calculations are being merged. The combined code will be verified with experimental data from FSV.

7. Temporary modifications were previously made in ORECA to determine temperatures in the PCRV during a temporary loss of PCRV-liner cooling capability. General limits must be defined for this accident, and permanent modifications must be made to accommodate a range of problems of this type.

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