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High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Accident Evaluation Quarterly Progress Report, April 1–June 30, 1984

S. J. Ball
J. C. Cleveland
R. M. Harrington
C. F. Weber
J. H. Wilson

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HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION QUARTERLY PROGRESS REPORT, APRIL 1—JUNE 30, 1984

S. J. Ball, Manager
J. C. Cleveland
R. M. Harrington
C. F. Weber
J. H. Wilson

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Oak Ridge, Tennessee 37831
operated by
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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).

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J. C. Conklin, *ORTURB: A Digital Computer Code to Determine the Dynamic Response of the Fort St. Vrain Reactor Steam Turbines*, ORNL/NUREG/TM-399 (March 1981).

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FOREWORD

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

This report covers work performed from April 1–June 30, 1984. Previous quarterly reports and topical reports published to date are listed on pages v and vi. Copies of the reports are available from the Technical Information Center, U.S. Department of Energy, Oak Ridge, TN 37831.

HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION QUARTERLY PROGRESS REPORT, APRIL 1—JUNE 30, 1984

S. J. Ball, Manager

J. C. Cleveland C. F. Weber
R. M. Harrington J. H. Wilson

ABSTRACT

Modeling, code development, and accident analysis work on the modular High-Temperature Gas-Cooled Reactor (HTGR) systems concentrated on predictions of core and other system temperature histories for postulated long-term loss of forced circulation accidents both with and without system depressurization. Fission-product (FP) release experiments to investigate vapor pressure and diffusion rates through graphite were continued. Experiments with additional elements were conducted.

1. HTGR SYSTEMS AND SAFETY ANALYSIS

Work for the Division of Accident Evaluation (formerly Reactor Safety Research) under the High-Temperature Gas-Cooled Reactor (HTGR) Systems and Safety Analysis Program began in July 1974, and progress is reported quarterly. Work during this quarter included continuation of model and code development for the modular HTGRs and for fission-product (FP) redistribution during severe accidents in large HTGRs. FP release and transport experiments and technical assistance work on a Fort St. Vrain (FSV) technical specification (tech spec) review were continued.

1.1 Modular HTGR Simulation Development

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

1.1.1 Introduction

Initial dynamic simulator development for the modular HTGR concentrated on the task of predicting the temperature and flow transients following postulated loss of forced circulation (LOFC) and design basis depressurization accidents (DBDAs). The first phase of this work consisted of developing simplified models for the core and primary coolant loop thermal hydraulics, using IBM's CSMP simulation language. The objectives of the task were to determine both the characteristics of the system response to various postulated accident scenarios and the sensitivity of the results to various assumptions of system design features, operational maneuvers, and modeling uncertainties. Of particular interest are the temperature histories of the hottest fuel and the metallic structural material exposed to high-temperature gas.

1.1.2 Core model

The reference model used for the pebble bed core and graphite block side reflector is a two-dimensional (R-Z) representation that includes both radial and axial conduction. Convection cooling by the primary system helium is assumed to occur in the pebble bed core but not in the reflector. In the nodal structure each axial segment has three radial nodes for the pebble bed core, two for the side reflector, and one for the core barrel wall (Fig. 1). In a more detailed core model used for

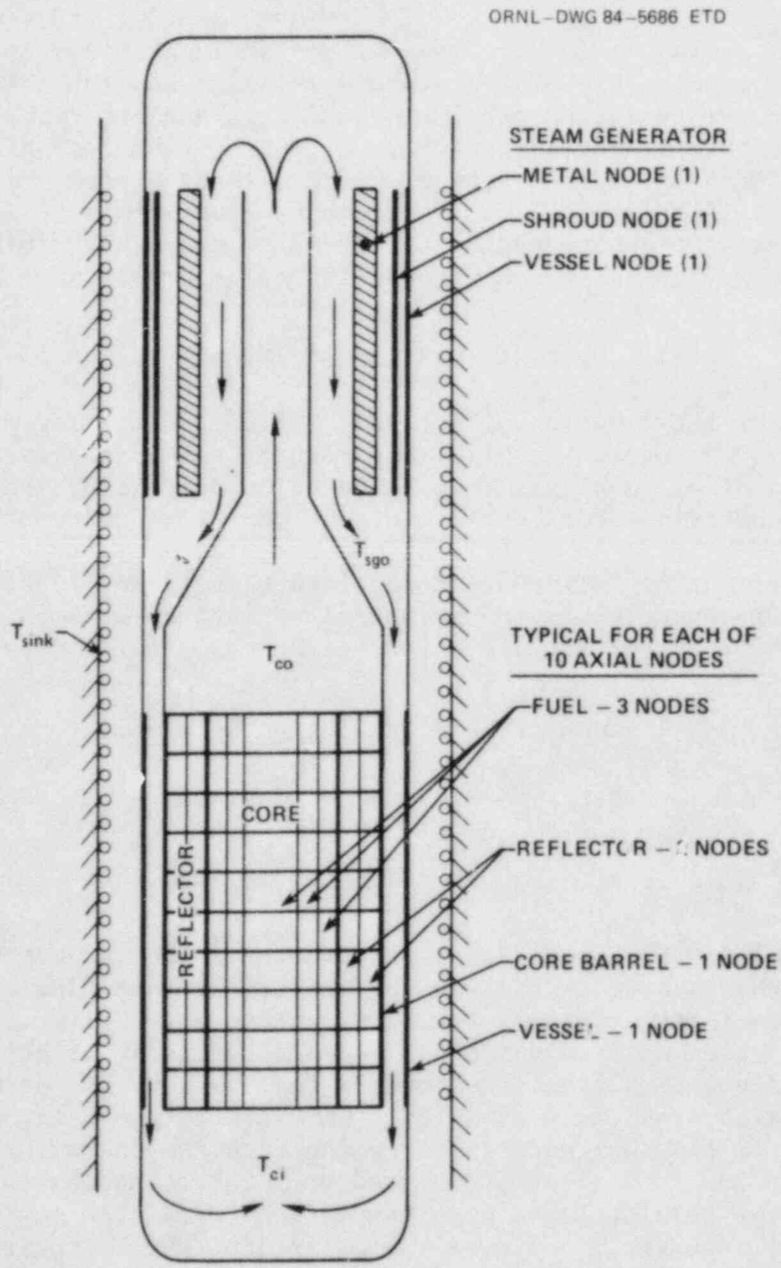

Fig. 1. Nodal structure of modular HTGR thermal model.

sensitivity studies, six radial nodes were used for the pebble bed core. There are ten axial segments. At this point, the radial core flow distribution is assumed to be uniform, and the total flow (if nonzero) is always assumed to be in the normal (upward) direction. A capability for modeling reverse (downward) flows would be useful only for simulating cases where slow leaks occurred near the bottom of the pressure vessel. The convection cooling model uses an exponential approach algorithm for computing coolant gas temperature, which permits representation of very low flows. No models are currently included for the top and bottom reflectors.

The core (and reactor) design features were assumed to be those of the latest General Atomic Technologies (GAT) plant as of January 1984,[1] with major characteristics as shown in Table 1. Physical property data and correlations were taken, where possible, from AVR information and

Table 1. 250-MW(t) modular HTGR performance parameters

Primary system data	
Reactor power, MW(t)	250
Power density, W/cm^3	4.1
Heat losses from NSS, MW(t)	3
Thermal power to NSS from circulators, MW(t)	4
NSS thermal power, MW(t)	251
Primary helium pressure, MPa (psia)	7.2 (1050)
Reactor inlet temperature, °C (°F)	283 (541)
Reactor outlet temperature, °C (°F)	688 (1270)
Steam generator gas inlet temperature, °C (°F)	686 (1266)
Steam generator gas outlet temperature, °C (°F)	279 (534)
Number of helium circulators	4[a]
Helium circulator, ΔP, psi	19.0
Gas flow rate, kg/s (lb/h)	119 (944,000)
Secondary system data	
Number of steam generators	4[b]
Steam generator power, MW(t)	251
Steam flow rate, kg/s (lb/h)	103 (816,000)
Steam outlet temperature, °C (°F)	541 (1005)
Steam outlet pressure, MPa (psia)	17.3 (2515)
Feedwater temperature, °C (°F)	221 (430)
Feedwater pressure, MPa (psia)	21.0 (3050)
Overall system	
Generator rated output, MW(e)	102
Net electrical output, MW(e)	95
Net plant efficiency, %	38.0

[a]Horizontal, single stage, axial compressor, external drive.

[b]Helical bundle, upflow boiling, nonreheat.

other current sources applicable to pebble bed technology. Helium convection heat transfer uses the Jeschar correlation;[2] pebble bed core effective conductivity, which accounts for radiant heat transfer, is derived from Breitbach and Barthels;[3] core specific heat uses a correlation by Petersen;[4] and the afterheat curve, which accounts for continuous fuel recycling, is from AVR sources.[5] Other physical property data, such as those for helium and reflector graphite, were taken from FSV reactor sources. A fuel failure model is included that calculates failure fractions that are dependent on the time that the fuel spends at a given temperature. Based on a model by D. T. Goodin of GAT, it was derived primarily for the large, prismatic-fuel cores and higher temperatures.[6] More sophisticated models that are better suited to the modular reactor (lower accident temperature) studies will be used later.

1.1.3 Steam generator models

Heated helium flows upward from the core through a central duct, turns as it reaches the top of the reactor vessel, and flows down through the steam generators. Several steam generator concepts have been proposed for the small modular HTGR. One involves the use of 4 FSV steam generator modules in parallel; 12 of these modules are employed at FSV to produce about 3.5 times the power proposed for the modular HTGR. The FSV-type modules would not be equipped with reheat sections. The other proposed steam generator concept is an annular design for which design details are not yet available. Complete information is available for the FSV modules, so FSV parameters were used to calculate coefficients for the model discussed below.

The present steam generator model is very rudimentary but is equipped with two modes to allow the simulation of either continued feedwater flow or the loss of all feedwater flow. The mode that simulates continued feedwater flow is essentially a steady state model. The helium is assumed to exchange heat with metal tubes that are at a single uniform temperature. The metal temperature is an input parameter and is assumed to be maintained constant by the continued flow of feedwater. The exponential-approach formulation is used to calculate the temperature of the helium as it flows downward through the steam generators.

The model that simulates the loss of feedwater treats the steam generator tube metal as a single heat capacity that exchanges heat with the incoming helium. Immediately after the loss of feedwater, there is a brief period during which there is a heat removal term that simulates the boiloff of the initial water inventory. There is also a heat removal term for the radiant heat transfer from the tubes to the steam generator shroud, which, in turn, radiates heat to the reactor vessel, and the reactor vessel radiates and convects heat to the reactor cavity. The surface temperature of the cavity is assumed to be maintained at a constant temperature by the cavity cooling system, which is normally active but can operate in a passive mode if necessary. The radiant heat transfer is not significant until after many hours into an LOFC accident when the steam generators have been heated to well above their normal temperature range.

The convective heat transfer coefficient between the helium and the steam generator tubes is calculated by means of the Grimison correlation with coefficients given by Carosella.[7] This correlation accurately describes the turbulent film coefficient and in the current model is extended to the laminar region as well.

1.1.4 Reactor vessel (core region) model

In an extended LOFC accident there is a significant transfer of heat from the reactor core to the reactor vessel and, hence, to the surrounding reactor vessel cavity. It is this heat loss that ultimately turns around the heatup of the core in an unmitigated, depressurized LOFC accident. Therefore, the core barrel and reactor vessel are modeled in some detail in the region surrounding the core. The reactor vessel model interfaces directly with the core model by a heat flux boundary condition between the core barrel and the core side reflector.

The temperature of the core barrel and reactor vessel is calculated for each of 10 axial regions that correspond to the 10 axial regions utilized by the core model. The 2.54-cm-thick (1-in.) core barrel is in contact with, and receives heat by, conduction from the outer reflector of the core. The core barrel is cooled by radiant heat transfer to the reactor vessel and by convective transfer to the coolant. The reactor vessel is heated by radiant heat transfer from the core barrel and convective heat transfer from the coolant and is cooled by radiation and convection to the reactor vessel cavity. The surface of the cavity is assumed to be maintained at 150°C (300°F) by the cavity cooling system acting in the passive mode.

The annular space between the core barrel and the reactor vessel is in the main helium flow path between the steam generator exit and the core inlet. Using the exponential approach method, the temperature of the helium is calculated for each of the 10 nodes, modified so that the helium approaches the surface area-weighted average temperature of the core barrel and reactor vessel. The coefficient for convective heat transfer between the core barrel or reactor vessel surface and the helium is calculated by means of the Dittus-Boelter correlation if flow is turbulent and by means of the Sieder-Tate correlation if the flow is laminar. If the Reynolds number is in the 2100 to 4000 transition region, the code linearly combines the turbulent and laminar formulas.

The radiant heat transfer between core barrel and reactor vessel in each axial segment is modeled as gray-body radiation with an emissivity of 0.8 for each surface. It is assumed that the annular gap is small, relative to node height, so that there is negligible radiation to nodes above and below. Radiant heat transfer from the outer surface of the reactor vessel is modeled as a gray body with an emissivity of 0.8 radiating to the surrounding enclosure.

1.1.5 Model for calculation of helium pressure and natural circulation

After the loop temperatures are calculated, the reactor vessel pressure can be calculated if the total mass is known. For transients not involving depressurization, the total mass of helium in the primary system remains constant at its initial value throughout the transient. For transients involving depressurization, the current model bypasses the pressure calculation and accepts an input pressure vs time profile.

It may be desirable in the future to install a calculation of coolant leakage and total mass if the rate of depressurization of the reactor vessel must be predicted.

The pressure calculation uses four coolant temperatures: (1) the average hot leg temperature, assumed to be equal to the core outlet temperature, (2) the average steam generator temperature, available as an analytical average of the exponential variation of helium temperature between inlet and outlet, (3) the average cold leg temperature, assumed to be equal to the steam generator outlet temperature, and (4) the average core coolant temperature, an arithmetic average of the 30 in-core nodal temperatures calculated by the core model. An approximate volume for each body of coolant was assumed since design details are not yet available. There appears to be more helium in the hot leg than in any of the other three volumes.

The natural circulation flow rate of helium during an LOFC accident depends on the driving head caused by the helium density differences around the loop and on the total pressure drop due to friction and form losses. Since all the coolant within the reactor vessel is at essentially the same pressure, the density differences are due to the temperature differences in the primary coolant flow circuit.

A total unrecoverable pressure drop of 20 psi at full power was used for the helium circuit. To relate this known total pressure drop to the unknown total pressure drop at reduced flows, the "smooth pipe" assumption was employed: the friction factor is proportional to the −0.2 power of Reynolds number (or mass flow). For a 100-fold decrease in flow, the assumed friction factor would increase by a factor of 2.5. The total pressure drop is, of course, a composite of individual component losses, some of whose friction factors may not begin to increase until the laminar range is approached closely. Therefore, use of the "smooth pipe" friction factor is, in this case, probably conservative for flow reduction into the laminar range. Future investigation is planned to examine component pressure drops individually to determine the actual variation of the composite friction factor.

1.1.6 Results

Several variations of the worst-case loss-of-cooling accident (LOCA) were run. The most extreme case is a simultaneous loss of primary system pressure and LOFC, along with a loss of feedwater cooling to the steam generators (Fig. 2). In this case the maximum fuel temperature reached 1728°C (3143°F) at 29 h from the start of the transient. Maximum pressure vessel temperatures were <400°C (750°F). Average steam generator tube metal temperature peaked at ∼595°C (1100°F) within the first hour. The steam generator tubes at the top (primary coolant inlet) end would reach higher temperatures because they are the first to be exposed to the hot helium drafting upward from the core [peak coolant temperature at core outlet = 1460°C (2260°F) at 38 h]. The single-node steam generator model calculates the average tube temperature but provides no estimate of the temperature of the hottest tubes.

The small amount of primary system natural circulation flow (∼0.5 kg/s or 0.05%) was marginally effective in reducing the maximum fuel temperatures, as evidenced by the fact that in a run in which the

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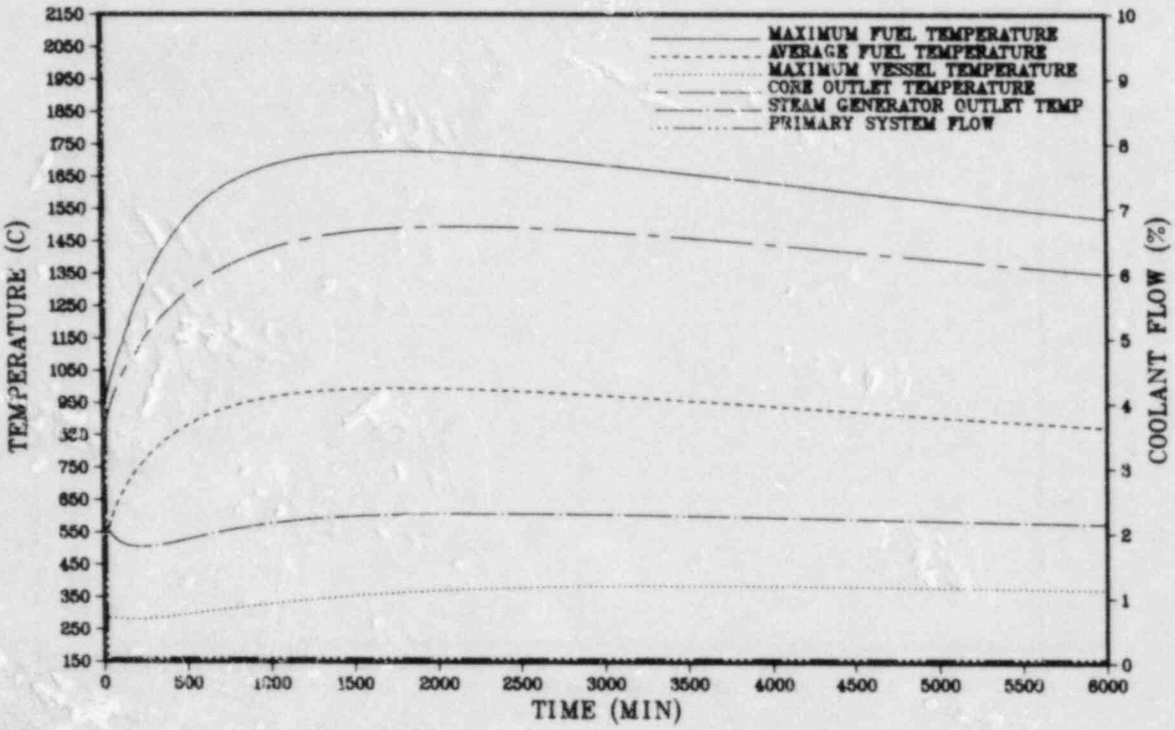

Fig. 2. LOFC with depressurization and loss of feedwater.

flow was forced to zero, the maximum fuel temperature reached 1783°C (3472°F) at 36 h. The time spent at the higher temperatures was also longer for the no-flow case. The total fuel failure as predicted by the Goodin model[6] was 0.3%.

In a second variation, it was assumed that forced circulation was lost and the system depressurized, but the feedwater to the steam generators was maintained (Fig. 3). This maintenance led to a slight reduction in the maximum fuel temperature by virtue of the slightly increased natural circulation (0.07 kg/s). Here, the maximum fuel temperature of 1679°C (3054°F) occurred at 25 h. Naturally, the steam generator and steam generator cavity temperatures were reduced considerably, remaining below the normal operating values after the start of the transient. Maximum pressure vessel temperatures in the core region were also lower [<345°C (650°F)].

In a third, more realistic case, it was assumed that the system remained pressurized, with LOFC and loss of feedwater flow to the steam generators occurring at time zero (Fig. 4). The natural circulation flows are much larger here (0.7 to 1.0 kg/s), and the core cools off relatively rapidly. The maximum fuel temperatures are no greater than the normal operating values. The pressure vessel temperature does get higher than in the other cases, however, due to the higher flow rates during the cooldown and reaches 470°C (880°F) 4 h into the run. The average steam generator tube temperature (approximately equal to the steam generator outlet temperature on Fig. 4) reaches a maximum of 697°C (1286°F) 1 h after accident initiation. Tubes at the steam generator

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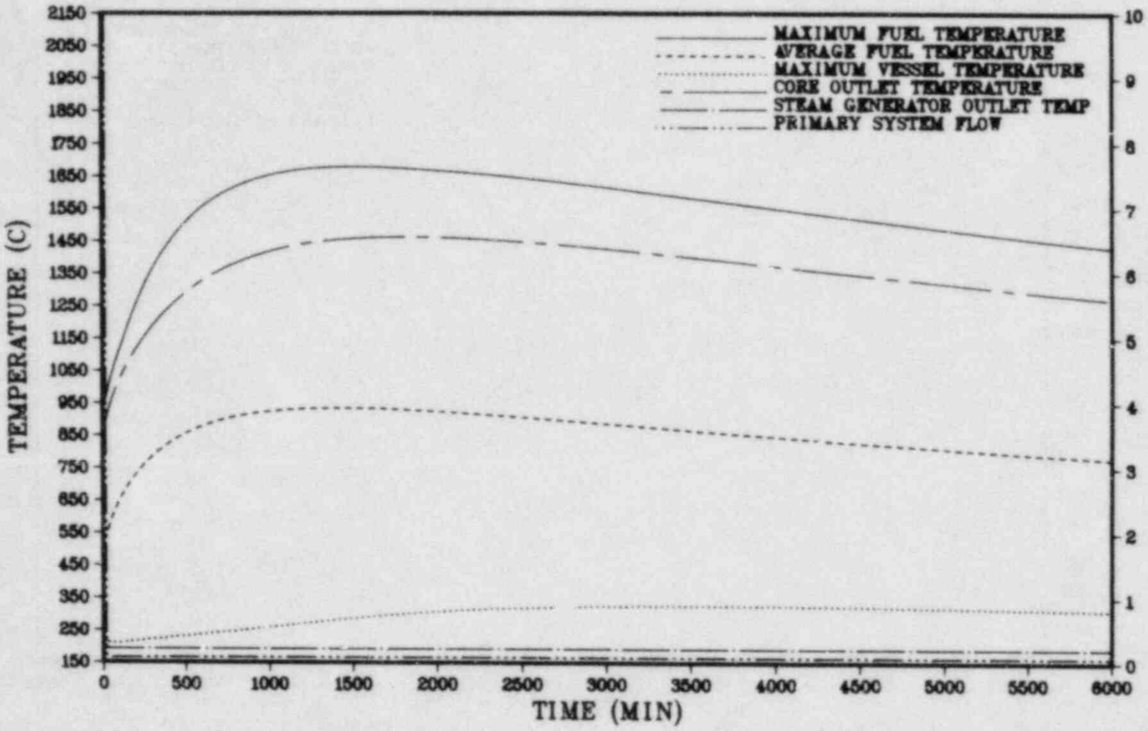

Fig. 3. LOFC with depressurization and continued feedwater flow.

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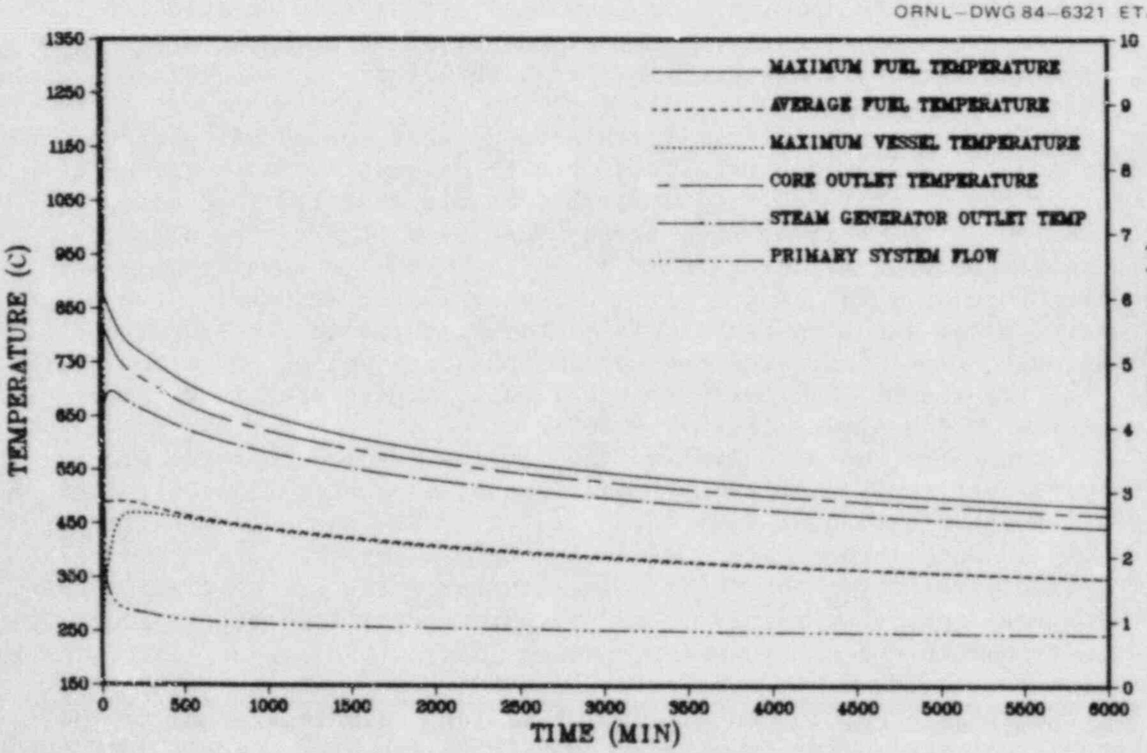

Fig. 4. LOFC with loss of feedwater but without depressurization.

inlet would more closely approach the core outlet temperature, which peaks at 848°C (1558°F) after 1 h.

The sensitivity of the results to variations in model, parameter, and operational input assumptions is also interesting, especially because other studies have claimed that the maximum fuel temperatures are somewhat lower than predicted here and, also, that the peak temperature for the depressurized case is not sensitive to natural-convection flow. A variation on the core nodal structure was run in which the number of radial fuel nodes was doubled (from 3 to 6), making a total of 60 fuel nodes. The results of comparison cases in which the convection flow was small (0.09 kg/s) showed that with the finer node structure, the maximum fuel temperature was higher by only 27°C. Because of this small difference in the most critical results, it was decided to use the simpler model as the reference case for subsequent runs. Other sensitivity studies planned include use of longer, more realistic flow coastdown and depressurization times (current calculations assume 5 min) and parameter refinements on the steam generator and natural circulation flow models.

1.1.7 Conclusions

Initial model development is complete, and a range of LOFC accidents has been investigated. The calculated results seem reasonable, but there is a significant difference between the results discussed above and previously published results. The peak fuel temperature specified above for the depressurized LOFC is 1728°C — about 150°C higher than the peak temperature reported from both GAT and Interatom studies,[8] presumably for the analogous accident. These differences can be due to a combination of various modeling and design assumptions, physical property data, and operational actions taken during the system depressurization and LOFC.

The safety and investment claims for the modular HTGR for loss of main heat sink accidents have been summarized by J. C. Cleveland.[9] Comments on each of the claims are listed below:

CLAIM: In the event of a loss of main heat sink accident, decay heat can be removed through the vessel wall to the vessel cooling system without excessive heatup of the fuel. This claim is made for both a pressurized primary system and a depressurized primary system.

COMMENT: The answer hinges on the definition of "excessive." If the higher fuel temperatures predicted here are valid, then some fuel damage will occur. The worst-case calculation of fuel damage using Goodin's model for large HTGRs gave a maximum cumulative particle coating failure fraction of 0.3%. In any case, it is expected that only a small fraction of the core would be damaged, so the problem, if any, would be one of investment protection (cleanup of in-plant contamination) rather than one of safety protection.

CLAIM: The vessel cooling system is normally in the active mode but can be converted to the passive mode and in the passive mode can provide cooling for 8 to 16 d without makeup of water.

COMMENT: Not much can be said here since the system is not yet designed. A volume of up to 490,000 liters (130,000 gal) of water would be required to last for 16 d.

CLAIM: Further, if the vessel cooling system is lost for whatever reason after 16 d, no overheating of the fuel sufficient to cause any significant release of fission products results.

COMMENT: Work on heat transfer to the surrounding earth in the cases of such extended outages is planned.

CLAIM: Relative to investment protection, with loss of main heat sink but with a functioning vessel cooling system, decay heat can be removed to the vessel cooling system (the primary system either pressurized or depressurized) without damage to the metallic heat exchanger components, to the reactor vessel, or to the vessel internals.

COMMENT: The acceptability of stress levels accompanying any given pressure/temperature transient depends heavily on design parameters, such as materials and dimensions, that have not yet been set for the modular HTGR; therefore, a firm conclusion regarding investment protection is much harder to make.

The worst transient for the reactor vessel is an LOFC accident in which the reactor vessel remains pressurized. The natural circulation realized in the pressurized cases provides enhanced cooling of the core and more efficient transport of heat to the reactor vessel. The rough calculation of peak temperature for the reactor vessel is 470°C (880°F). The strength of the reactor vessel steel would be reduced to about 80% of normal [i.e., 218°C (425°F)] at this temperature, but the vessel would not necessarily be overstressed.

Peak temperatures for steam generator tubes occur at the primary coolant inlet end and can approach the peak core outlet helium temperature in a pressurized LOFC (850°C). These temperatures would probably be unacceptable for a tube material such as 2-1/4 Cr-1 Mo steel (which is used for the main portion of the steam generators at FSV) but would be acceptable for a material such as Incoloy 800. For the depressurized case the average tube temperature is lower, but the coolant flowing out of the core reaches a temperature high enough (1460°C) to damage most high temperature alloys (e.g., Incoloy 800). Assessment of steam generator tube damage for the depressurized case depends on the completion of more detailed steam generator and natural circulation modeling.

1.1.8 Recommendations for future work

As noted, several modeling improvements are warranted. In the core, account should be taken of the top and bottom reflector heat capacity, nonuniform core radial flow distributions, and possible radial and recirculating flows in the core during periods of natural circulation. The fuel failure model used[6] was derived for large HTGRs, and GAT has recommended the use of other methods for modular reactors.[10,11] Apparently, the current model may significantly underestimate the release of metallic fission products that diffuse through intact Triso fuel particles at temperatures as low as 1100 to 1200°C.

Refinements are planned for the natural circulation flow calculations, including use of more detailed correlations for the full range of turbulent-to-laminar flow. A finer structure model for the steam generator and steam generator cavity is also planned as more design data become available. Use of more steam generator nodes is expected to result in higher predicted natural circulation flows, which could reduce the predicted peak fuel temperatures, and in higher steam generator tube temperatures, which could increase the possibility of tube damage. More design details on the cavity cooling system would be required for refining the studies of various active and passive cavity cooling system availability scenarios.

The initial cooldown trajectory assumptions could also have a significant effect on peak temperatures: for example, longer depressurization and helium circulator coastdown times and extended steam generator feedwater flow availability.

Models will also be developed for predicting long-term LOFC accidents in which the cavity cooling system fails and heat transfer to the surrounding earth is significant.

Other model changes for different reactor designs — such as the side-by-side configuration (rather than the steam generator located above the core) and annular (rather than cylindrical) cores — would, of course, necessitate major code changes.

1.2 Fission-Product Release from HTGRs

J. H. Wilson R. L. Towns

The objective of this task is to generate experimental data required for the analysis of FP release in HTGR severe accidents. Initial efforts involve the determination of FP vapor pressures and diffusion rates through graphite. The experimental procedure consists of measuring the rate of loss at high temperature from a mixture of powdered graphite and simulated FPs that has been placed in a 6.4-mm-diam graphite tube. As the products diffuse through the tube wall, they are transported through a cold collection tube by argon carrier gas.

As discussed in the last quarterly report, the experimental data have been analyzed assuming gas phase diffusion through the porous graphite. To demonstrate that the mode of transport indeed is primarily by gas-phase diffusion rather than by surface diffusion, experiments with silver and palladium were conducted under vacuum ($\sim 1 \times 10^{-3}$ atm). If surface diffusion were the dominant mode of transport, essentially no effect of the low pressure would be expected. The experimental diffusion rates at the low pressure were found to be about a factor of 10 lower than those at atmospheric pressure. Thus, these results are believed to support the assumption of gas phase diffusion, even though an increase in diffusion rate at the low pressure may have been expected. The lower rates are believed to be a result of the higher volumetric rates through the porous graphite (by a factor of about 1000 at the same mass rate). At these high rates or velocities, frictional losses become limiting.

Previous experiments with palladium, rhodium, and copper produced vapor pressure vs temperature curves that differed significantly from those presented in the literature.[12] To verify the experimental vapor pressure data, experiments were performed in which the metal was placed in an impervious zirconia tube (which had one open end and one closed end) and the rate of loss from the tube determined as a function of temperature under an argon atmosphere. In this system the rate of loss is dependent upon the vapor pressure and the gas-phase diffusion coefficient. This dependence is also assumed to be the case in the experiments involving transport through the wall of a porous graphite tube. Here, however, the transport occurs by diffusion through the gas space above the metal sample and not through the tube wall. Two data points were obtained with silver before the zirconia tube cracked. When new tubes have been received, the silver test will be completed, and data with palladium will be obtained. Since the gas phase diffusion coefficients for palladium and silver should be essentially equal, the vapor pressure of palladium may be determined by comparison with the data for silver. Silver is used as a standard for comparison since the heat of vaporization of silver determined in the graphite diffusion tests was within 3% of the reported values.[12]

Plans are to complete diffusion measurements with the noble metals and then begin experiments with the rare earth metals.

1.3 Model and Code Development for Fission-Product Redistribution During Severe Accidents

C. F. Weber

An extensive literature survey has been undertaken to study the various facets of FP transport (including release from fuel particles, passage through fuel rod matrix and structural graphite, transport in coolant stream, condensation, and chemical reactions). Apparently, most available data are at temperatures too low to be of use in the high-temperature accidents under consideration. Furthermore, there are a lack of data for some FP groups altogether.

It is important from a computational viewpoint to calculate movement of all FP groups by similar models, even though the actual transport mechanisms may vary considerably. To ensure this similarity, the following model has been tentatively adopted for the diffusive transport through structural graphite:

$$\frac{\partial C}{\partial t} = D \nabla^2 C + \mu m - bC,$$

$$\frac{\partial m}{\partial t} = bC - \mu m,$$

where C and m = concentrations diffusing and stationary, respectively, and D, μ, and b are constants generally representing coefficients of diffusion, adsorption, and revaporization, respectively. This model would be used for all FP groups; for nonadsorbing species, $\mu = b = 0$. Boundary conditions are determined by the release from fuel and the coolant concentration. The former is determined in the same manner as in the SORS computer code,[13] while the latter is determined by a simple convection model upward through the core. Current intentions are to pursue the following work plan:

1. Set up a calculation to solve the equations in the effective radial geometry used by ORECA.

2. Insert simple models for release from particles in core coolant transport of FPs.

3. Modify the above routines to make them subroutines of ORECA.

4. Add more rigorous treatment of coolant inventories: deposition on cooler surfaces and transport to the entire primary system.

1.4 Review of FSV Reactor Technical Specification on Limiting Maximum Core Temperatures

S. J. Ball

Further work was done reviewing FSV technical specification (tech spec) limiting condition for operation (LCO) 4.1.9. This project is supported in part by the Nuclear Regulatory Commission (NRC) Region IV in Arlington, Texas. LCO 4.1.9 is used in conjunction with several other tech specs to ensure that core temperatures do not exceed established limits.

FSV startup and shutdown transients were investigated in detail, using specially modified versions of the PSC HISTORY code and the Oak Ridge National Laboratory (ORNL) ORECA code. Work was done on developing a method for estimating primary helium flows in the 0 to 15% range, using circulator performance map information in conjunction with measured circulator speeds, temperatures, and pressures. The currently used methods have problems with accuracy and dependability at very low flows, making it difficult to assess compliance to LCO 4.1.9. The new map method appears to give generally satisfactory results (i.e., they result in good heat balances). Analyses of the FSV transient data were also done to infer core bypass flow fractions and to compare predicted and measured core and loop flow resistances.

Further work investigating the tendencies for refueling region flow redistribution and stagnation was done using ORECA. Data supplied by PSC for a November 1983 reactor startup were used as a reference, and postulated variations in operating conditions about the reference were tested to determine how close they were to stagnation conditions. The tests showed that for wide variations around "normal" operating paths, very little flow redistribution between regions is predicted, and flow stagnation is much further removed and difficult to achieve. Inter-region flow redistributions, which are precursors to stagnation, are readily observable by measuring changes in outlet temperature dispersions due to changes in total primary loop flow.

1.5 Cooperative Programs with the Federal Republic of Germany (FRG)

J. C. Cleveland

No work was performed on this task during this quarter because the agreement for the cooperative program between the West German Ministry of the Interior and NRC has not yet been formally approved.

2. TRIPS MADE UNDER PROGRAM SPONSORSHIP

2.1 Midyear Program Review Meeting at NRC, Bethesda, Maryland

S. J. Ball R. M. Harrington
J. H. Wilson

The annual program review was held May 2–3, 1984, in conjunction with corresponding reviews of the Brookhaven National Laboratory, Idaho National Engineering Laboratory and Los Alamos National Laboratory programs. The presentations covered the past two years' work (rather than one) and included more than the usual amount of background material for the new Nuclear Reactor Regulation (NRR) group assigned to HTGRs. The ORNL work reviewed included code development and verification, the 2240-MW(t) cogeneration plant source-term study, the FSV severe accident analyses, initial modeling and analyses of small modular HTGR dynamics and safety features, and the theoretical and experimental work on FP transport.

Plans for FY 1985 programs were also discussed at length. Some of the research direction depends heavily on DOE decisions expected next spring on lead plant type.

2.2 Lecture on Reactor Dynamics at Tennessee Technological University, Cookeville, Tennessee

S. J. Ball

In response to an invitation, a lecture was given to a graduate electrical engineering class at Tennessee Technological University on April 25, 1984. The talk covered HTGR and other power reactor modeling and dynamic analysis techniques.

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Modeling, code development, and accident analysis work on the modular High-Temperature Gas-Cooled Reactor (HTGR) systems concentrated on predictions of core and other system temperature histories for postulated long-term loss of forced circulation accidents both with and without system depressurization. Fission-product (FP) release experiments to investigate vapor pressure and diffusion rates through graphite were continued. Experiments with additional elements were conducted.

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