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NUREG/CR-3492
Volume 4
ORNL/TM-8921/V4

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High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Accident Evaluation Quarterly Progress Report, October 1–December 31, 1983

S. J. Ball, Manager
J. C. Cleveland
R. M. Harrington
I. Siman-Tov
J. H. Wilson

Prepared for the U.S. Nuclear Regulatory Commission
Division of Engineering Technology
Under Interagency Agreements DOE 40-551-75 and 40-552-75

OPERATED BY
MARTIN MARIETTA ENERGY SYSTEMS, INC.
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161

Available from
GPO Sales Program
Division of Technical Information and Document Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

NUREG/CR-3492
Volume 4
ORNL/TM-8921/V4
Dist. Category R8

HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF ACCIDENT EVALUATION QUARTERLY PROGRESS REPORT, OCTOBER 1—DECEMBER 31, 1983

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Manuscript Completed — May 25, 1984
Date Published — July 1984

NOTICE: This document contains information of a preliminary nature. It is subject to revision or correction and therefore does not represent a final report.

Prepared for the
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NRC FIN No. B0122

Prepared by the
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37831
operated by
MARTIN MARIETTA ENERGY SYSTEMS, INC.
for the
U. S. DEPARTMENT OF ENERGY
under Contract No. DE-AC05-84OR21400

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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 October 1—December 31, 1983. 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, OCTOBER 1—DECEMBER 31, 1983

S. J. Ball, Manager

J. C. Cleveland   I. Siman-Tov
R. M. Harrington   J. H. Wilson

ABSTRACT

Development work continued on models and codes for predicting source terms in both the Fort St. Vrain (FSV) and 2240-MW(t) lead plant reactors. Experimental work on fission-product vapor pressures and diffusion rates through graphite continued at temperatures up to 2775 K, and a mathematical model of the experimental system was developed to aid analysis of the results and to guide improvements in the system and experiment design. Benchmarking of the BLAST steam generator code continued using FSV data, and more support work was done for proposed FSV core bypass flow model verification. Progress was made in setting up cooperative high-temperature gas-cooled reactor (HTGR) safety research with the Federal Republic of Germany. A review of a FSV technical specification on limiting maximum core temperature was begun.

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1. HTGR SYSTEMS AND SAFETY ANALYSIS

S. J. Ball

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 continuing development and benchmarking the Oak Ridge National Laboratory (ORNL) HTGR codes and their applications to safety and severe accident analyses of both the Fort St. Vrain (FSV) and lead plant HTGRs, small-scale experiments for studying fission-product (FP) release and transport behavior, and plans for large-scale thermal-hydraulic tests.

1.1 Development of ORECA Code for Simulating FSV Severe Accident Transients

R. M. Harrington

The effort to modify the FSV version of the ORECA code[1] (ORECA-FSV) to calculate thermal response of reactor core and prestressed concrete reactor vessel (PCRV) during unmitigated core heatup accidents was initiated in March 1982. This effort has consisted primarily of adding new capabilities, with very little modification to already existing coding. As features have been added to ORECA-FSV, they have been discussed, and typical results of their use have been presented in previous quarterly reports.

In October 1983, nuclear engineering department personnel at the FSV parent utility, Public Service Company of Colorado (PSC), expressed an interest in becoming familiar with the programming of the severe accident version of ORECA-FSV. To maximize the exchange of information, an effort was made to improve the documentation of the code before releasing the listing. Comment statements were added to highlight and label new segments of code, and a separate document was written that contained a brief explanation of the organization of the modifications. This document is included in the following subsections.

1.1.1 Plenum elements

A plenum element (PE) sits atop each fuel column. The purpose of the PEs is twofold; the bottom part of each PE is filled with crushed boronated graphite that reduces neutron streaming from the core, while the top part is open on the inside and provides a plenum that distributes coolant from the single refueling region flow control orifice to each of the seven (or five) fuel columns in the region.

PEs were neglected in previous versions of ORECA because their effect on most normal operational or postscram transients is not significant. However, in prolonged loss-of-forced-convection (LOFC) accidents, radiant heat transfer from the top of the core to the PCRV above becomes a prime consideration. PEs comprise a multiple barrier to radiant heat transfer and have therefore been included in the modifications. Two nodal temperatures are calculated for each of the 37 refueling region PEs: a temperature for the bottom (boronated graphite) part and one for the top (metal structure) part.

Various parts of the PE calculation are computed in ORECA-FSV-SA subroutines.

1. dynamic nodal heat balances: subroutine INPL;
2. convective heat transfer from PE nodes to reactor coolant flowing inside them, coolant temperature at exit of nodes: subroutine CONVEC; and
3. all other heat exchange with surroundings: subroutine INPL.

During initial planning of the modifications, it was determined that the calculation of all variables needed for or related to the severe accident version should be entirely separate from the existing ORECA-FSV routines. Therefore, the bulk of the calculations of PCRV cover plate temperatures, heat transfer rates, etc. is done in the new subroutine INPL. It was not deemed advisable to separate the calculation of internal PE convective heat transfer and coolant temperatures because the PEs are an extension of the refueling region coolant channel, and the normal operation of ORECA-FSV typically requires channel coolant temperature to be calculated more than once per time step. Therefore, part of the PE calculations are done in CONVEC and the rest in INPL.

The user may elect not to calculate PE temperatures by setting a control flag KPEC = 0.

1.1.2 Core inlet plenum

For the core inlet plenum, calculations are made for a single bulk coolant temperature, a temperature of the PCRV insulation cover plates (or of the liner if cover plates have failed) directly above each of the 37 refueling regions, and a single average PCRV insulation cover plate temperature for the PCRV inlet plenum region not directly above active fuel.

An average liner temperature is calculated for the PCRV liner directly above active fuel and for the inlet plenum region not above active fuel. Heat transfer to this liner region must be calculated from 37 different cover plate temperatures. The average liner cooling system (LCS) water exit temperature and the temperature of the PCRV concrete (at different depths into the concrete) are also calculated for each of the two inlet plenum regions.

The inlet plenum calculations are done primarily in the INPL routine; the bulk coolant temperature calculation is in MAIN.

The INPL routine calls other subroutines as required to complete details of the above calculations; for example,

1. DETLCS calculates LCS heat removal and water temperatures;
2. EMATS and MSOLVE compute PCRV concrete temperature profiles;
3. RADFAC provides radiant heat transfer view factors in core inlet plenum; and
4. HPLUME calculates the heat transfer coefficient between the plume from a refueling region in upward (reverse) flow and the cover plate directly above.

1.1.3 Sidewall region

The sidewall region is divided into ten axial regions that correspond to the ten axial regions of the ORECA core model. For each of the ten regions, core barrel temperature, PCRV insulation cover plate temperature, and coolant outlet temperature are calculated. A single average PCRV liner temperature and an average LCS heat removal and cooling water exit temperature are also calculated. Primary coolant temperatures are calculated in CONVEC, and all other sidewall calculations are performed in INPL.

1.1.4 Circulator discharge plenum

For the circulator discharge plenum, bulk coolant temperature, PCRV insulation cover plate temperature, and temperature of the metal floor that separates the circulator inlet plenum from the circulator outlet plenum are calculated. These calculations are conducted in ORECA-FSV-SA routines

1. bulk coolant temperature in MAIN; and
2. cover plate and metal floor temperatures in INPL.

There are no liner, LCS, or PCRV concrete temperature calculations for the circulator inlet or discharge plenums in the present code. Because these regions of the PCRV are separated from the core, the assumption is made that the average liner temperature remains at 100°F.

1.1.5 Steam generator coolant temperature

If the Kloop = 1 option is selected, temperature in subroutine TIN is interpreted as steam generator tube temperature instead of steam generator primary coolant exit temperature. A very rudimentary calculation in MAIN calculates the steam generator primary coolant outlet temperature from the input tube temperature.

1.1.6 Primary coolant pressure

By selecting the KPREC = 1 option, the code will ignore the input pressure in the PRESS subroutine and calculate the primary coolant pressure throughout the transient. The pressure calculation is done in the INPL subroutine, using the initial helium total mass calculated in MAIN. The PRESS subroutine must still be included to specify initial pressure.

1.1.7 Fuel failure

The ORNL adaptation of the Goodin fuel failure model[2] has been included in the ORECA-FSV code (as the GOODVT subroutine). For each of the 222 active fuel nodes, GOODVT calculates the cumulative failure fraction as a function of time. The fuel failure is defined in this context as the inability of the fuel particles to contain noble gas FPs.

1.2 Fission-Product Release from HTGRs

J. H. Wilson A. L. Weinberger

The objective of this task is to generate experimental data required for the analysis of fission product (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 temperatures (maintained by a graphite-resistance furnace) from a mixture of powdered graphite and simulated FPs (using nonradioactive species) 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.

Chromel-Alumel thermocouples were inserted at various locations in the cold collection tube, and the furnace temperature was raised until the temperature limit of the thermocouples was approached. By extrapolating the measured temperature gradients in the collection tube, estimates can be made of the temperatures at which deposition of the simulated FPs occurred in the previous experiments. At a furnace temperature of 1900 K, a temperature drop of ~400 K occurred over a distance of 35 mm from the sample source. At this location, deposits were observed in the experimental runs. If the temperature gradient increases with increasing furnace temperature as expected, direct measurement of the temperature in the deposition region of the collection tube when operating at a furnace temperature of 2775 K may be possible. For this purpose, the Chromel-Alumel thermocouples have been replaced by platinum vs platinum-rhodium thermocouples.

Neutron activation analysis (NAA) results from the second run (2775 K for 25 h) were received. The sample originally contained carbides of Si, La, Nd, Ce, Sr, and Zr dispersed in graphite. Whereas only 20.5% of the zirconium was lost from the sample after heating, essentially 100% of each of the other elements was lost (silicon was not detected by NAA and will be analyzed by another method). As in the first run, very rapid diffusion through the wall of the graphite sample tube was indicated. For both the first and second runs, the amounts of the elements that were found to have deposited downstream on the cold collection tube were 1% for zirconium and from 7 to 27% for the other elements (excluding silicon), as based on the original sample weights. Analysis of a composite sample of the carbon felt insulation that surrounds the heating element indicated that diffusion of the elements through the outer graphite tube, which is concentric with the sample tube, could account for the low recovery on the cold collection tube, with subsequent deposition on the felt insulation.

A mathematical model of the experimental system was developed to aid in the analysis of the experimental results as well as to provide insight for modifications to the system that potentially could improve the quality of the data. Calculations showed that with the present system, the low recovery on the cold collection tube was in fact due to loss by diffusion through the outer graphite tube. To reduce the loss to the felt insulation and to allow the experimental determination of both diffusion coefficients and vapor pressures, the model indicated that either the wall thickness of the tube surrounding the sample tube must be increased or this outer tube must be fabricated from a more impervious material.

A fourth run was made at 2675 K for 12 h. The sample contained the same components (Si, Ce, Nd, Ba, Zr, Nb, Ru, Rh, Pd, and Mo as metals and metal carbides) as those in the third run at 2675 K for 26 h. The argon flow rate was increased so that a larger fraction of material would deposit on the cold collection tube. The run length was decreased because a preliminary analysis showed that in the third run the losses of the elements from the sample were high (i.e., essentially 100% loss). Lower losses are necessary to accurately determine the rate of loss that is needed for calculation of the diffusion coefficient.

Preliminary results from the fourth run also indicated high losses of the elements from the sample tube. Consequently, a fifth run was made at 2675 K for 3 h, the shorter time again for the purpose of obtaining a better measure of the rate of loss. In the fourth and fifth runs, additional carbon felt insulation was added to maintain a less steep temperature gradient along the cold collection tube. From an x-ray of the cold collection tube, a better separation of the elements that deposited from the argon carrier gas was achieved.

Because of the long delay time involved for the NAA, future runs will be made where only one element at a time will be charged to the sample tube and where the rate of loss will be calculated by weight loss. This will allow data for the individual components to be generated very quickly. After building this data base, mixtures will be run to study possible interactions, with neutron activation again being used as the analysis method.

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

S. J. Ball

At the request of the Nuclear Regulatory Commission (NRC) Region IV, ORNL is providing technical support in their review of the FSV limiting condition on operation (LCO) 4.1.9. The intent of this technical specification (tech spec) is to ensure that during low-power and low-flow operation (0 to 15%), core region temperatures will be limited to acceptable maximum values. The major basis for this concern is that at low core flows (and hence low core pressure drops), the effects of higher buoyancy forces of the pressurized helium coolant in the cooler channels, coupled with the higher flow resistances in the hotter channels, may lead to flow stagnation and reversals in some channels. The uncertainties of the region heat removal processes under these circumstances make it desirable to ensure that region flow stagnation and reversals do not occur at all. The basis of the current LCO 4.1.9 is to specify a set of conservative operating limits for both startup and shutdown, with the reactor either hot or cold, or pressurized or not pressurized. NRC, PSC, and General Atomic Technologies (GAT) have all identified problems with consistency, accuracy, and conservatism of the current and interim tech specs, and NRC has indicated that a thorough independent review and analysis is warranted.

The initial work on the task included a review of the basis for the LCO and identification of the problems in the analyses, measurements, and operation.

To conduct detailed analyses of the reactor operating conditions, the model verification version of the ORECA-FSV code was modified to facilitate startup and shutdown runs, including typical refueling region orifice maneuvers. For startups, the calculation begins with a zero-power uniform-temperature core and follows user-input trajectories of total circulator flow, thermal power, primary system pressure, and core inlet temperature. Guidelines for typical startup scenarios have been obtained from the FSV DC-5-2 (Issue C) manual both for startup from refueling conditions and for startup with full inventory. For shutdowns, the power and flow rundown conditions are arbitrary user inputs. In both cases, orifice manipulation routines are executed to go from approximately equal-flow to equal-temperature rise settings, or just the opposite, at specified times. Other user inputs include the refueling region peaking factors (RPF) and orifice positions and the various core and refueling region bypass flow fractions. A watchdog routine was also added to ORECA that detects violations of the existing LCO (for both LCO 4.1.9 Figs. 1 and 2 conditions), noting the beginning and ending times for the violations and, for the Fig. 2 case, the value of the maximum region temperature rise. The code includes a model of the dynamic response of the region outlet thermocouples, which have fairly long response times, especially at the low flows associated with startup and shutdown. Calculations for the LCO of core thermal power and region temperature rises are made based on these simulated thermocouple measurements rather than "actual" region outlet temperatures, since the measurements are used by the operators to determine the coordinates on the two figures.

Some basic problems and limitations were noted with the restrictions and the bases for both the existing and temporarily imposed LCOs. The idea of the tech spec is to limit the maximum core temperature by ensuring that region flow stagnation will not occur. Because there is no direct way of measuring stagnation, the calculation needs to be fairly conservative and, for the sake of simplicity, cover a wide range of possible values for operating parameters such as RPFs and orifice positions. Simply ensuring that no stagnation will occur does not necessarily mean that the core will not overheat. Another simplification that increases the conservatism is that the LCO is made to apply to both startup and shutdown conditions.

Several specific problems were noted: (1) There may be problems in the measurement accuracy for low reactor powers, because of the inherent limitations of low-flow measurements. (2) There are also large uncertainties in the fraction of the total circulator flow that bypasses the core entirely and the bypass fraction for the refueling regions within the core. These fractions could change significantly with changes in operating parameters such as total flow and refueling region orifice positions. (3) The GAT analyses established the operating limits by using only steady state codes, while in fact, in typical startup and shutdown scenarios, the reactor is not necessarily in steady state. It is important to know if the consideration of the dynamic effects would impose more restrictive conditions. Dynamic effects should also be considered in determining how much time should be given for corrective action in case limits are exceeded. (4) The current LCOs indicate an "equal-flow orifice positions" mode but do not specify what these orifice positions are. The tendency for flow redistribution is very sensitive to the absolute, rather than just the relative orifice positions. For nearly closed orifices, it is very difficult to slip into reverse flow. At the same time, however, higher region flow resistances lead to higher region bypass flow fractions.

Based on our initial review, two preliminary observations and recommendations were made for discussion purposes: (1) An LCO that would nearly satisfy the intent of the tech spec (limit core temperatures) would be used to have the region orifice positions set according to calculations of the RPFs, with verification of the success of the maneuvers made via measurement of region temperature rises. In cases where the total circulator flows are low, stipulations could be made about the absolute orifice settings such that significant flow redistributions due to heatup effects would be avoided. For such an LCO, a single upper limit on power-to-flow ratio would probably be sufficient. As long as the temperatures did not get too far out of line, there could be reasonably long compliance times for correcting off-normal flows. (2) Verification should be considered an important part of the proposed LCO revision, because the success of the orifice manipulations would depend on the accuracies of the calculation of RPFs and the estimate of flows resulting from orifice position changes. Most of this verification work could probably be based on existing data system records of several startups and shutdowns; however, some tests to verify the models for calculating region flow redistributions or reverse flows may be justified.

1.4 Development of the BLAST Steam Generator Dynamics Code

J. C. Cleveland

As a part of our code benchmarking activities, BLAST[3] results were compared with measured FSV steam generator data for a loop shutdown transient, which occurred on November 9, 1981. The computed results for a steam generator module in the shutdown loop predict that steam generator flood out should occur ~4 min into the transient. However, measured data imply that flood out had not occurred even 25 min into the transient. One possible explanation is that the feedwater flow measurement, which can be quite inaccurate at low flows, was indicating a higher-than-actual flow. The differences in calculated and measured conditions will be investigated further and discussed with PSC personnel.

The most recent version of the BLAST steam generator dynamics code, the code documentation, and a sample problem were sent on request to PSC. PSC plans to use the BLAST code to investigate the response of a steam generator module to postulated leaks of cold helium through the buffer seals.

The BLAST HTGR steam generator simulation code, the documentation, and a sample problem were also provided to Rheinisch Westfälischer Technischer Überwachungs Verein e.v. (RWTÜV) of Essen, Federal Republic of Germany (FRG) at their request. This version contains all modeling improvements and refinements made by ORNL to the RWTÜV/KFA version of BLAST, which was provided to ORNL in December 1981.

1.5 Large-Scale Thermal-Hydraulic Experiment Planning - Core Bypass Flow Tests

S. J. Ball

An updated proposal for verification work on the core bypass flow models was submitted to NRC and PSC for comment. The additional material is based on input from both PSC and GAT and on further analysis of FSV plant data.

Detailed scoping calculations were done for the FSV shutdown transient from 100% power on November 9, 1981. Extensive data (from the plant data logger) were made available by PSC, and numerous comparisons were made of the data with ORECA code predictions and with other calculations. This was an excellent transient to use for bypass flow modeling investigations, because it involved a transition from full-power steady state conditions to part-load operation on one loop, and finally to the resetting of the refueling region orifices (to ~17% open) after 3 h.

Using a model for the FSV circulator performance developed for use in the ORTAP code,[4] predictions could be made of expected flow and loop (or circulator) ΔP and circulator temperature rise, given the circulator speed, inlet temperature, and pressure as inputs. Absolute values of and changes in core ΔP were also calculated by ORECA for various sets of assumptions about core bypass flows. While some of the data logger readings obviously had bias or other errors, most appeared to be quite reasonable and checked out well with the predictions.

The total primary flow resistance R_L is assumed to be related to the loop pressure drop ΔP_L (or circulator pressure rise), the ratio of absolute temperature to pressure T/P, and total loop flow W_L by

$$\Delta P_L = R_L \ (T/P) \ W_L^{\ 2} .$$

Making use of the ciruclator model, calculations show that the data indicate a decrease in effective loop resistance of ~40% due to the loop trip and accompanying reduction in flow and pressure drop. Only an insignificant part of that (<2%) is due to a change in core flow resistance. The resetting of the refueling region orifices increased the core resistance 61% while increasing the total loop resistance 23%. Using a variety of "reasonable" bypass flow assumptions, preliminary ORECA calculations predicted increases in core resistance due to the orifice changes of between 54% and 59% (vs 61%); however, the absolute values of calculated core ΔP were 18% to 43% higher than measured. Part of this discrepancy may be due to the fact that some of the core flow bypasses the orifices, an effect that is not modeled in ORECA. One PSC estimate for this bypass flow is ~16% which could account for a significant part of the ΔP error.

Refinements in the ORECA model are planned, including an option to incorporate circulator performance, so that core inlet temperature and flow can be derived from circulator speed and pressure drop data. An option to specify an orifice bypass flow fraction will also be added.

Detailed bypass flow experiments planning would include simulating proposed scenarios using the updated ORECA code, accounting for uncertainties in models and parameters by sensitivity studies, and taking advantage of enhanced plant data logger capabilities.

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

I. Siman-Tov

A list of fission products, heavy metals, and activation products present in an HTGR, composed of the nuclides described by KFA[5] and GAT,[6,7] are presented in Table 1. The table also includes the mobility groupings and boiling points given in Refs. 5-7. Since melting points may be of greater significance in determining the mobility of the diffusing and stationary fission products, they are also included with boiling points as given by two sources in the *Chemical Engineering Handbook* (*CEH*)[8] (Table 1). For some nuclides, there are major unexplained differences between the boiling points from the two *CEH* sources.

For purposes of present code development, five FPs were chosen: one stationary, one volatile, and three mobiles with different rates of mass diffusion. These can later be replaced by groups with effective mass transport, nuclear decay, and production properties. A final decision on the particular five nuclides will be left open until the relevant parts of the code are ready for testing.

A block structure established for the code that is presented in Fig. 1 forms the basis for the stepwise refinement process and presents the main flow of the program. Stepwise refinement is in progress and is a continuous process during the entire code development stage. A card image file is being created that includes the documentation of the code as it is being developed. At present the file consists of the problem description with its basic assumptions and limitations and the main block description.

1.7 Cooperative Programs with FRG

J. C. Cleveland

A meeting was held at ORNL on October 21, 1983, with K. D. Paul of RWTÜV to discuss possible cooperative programs for HTGR accident analysis and code verification. RWTÜV is using our BLAST steam generator dynamics code for THTR and SNR (liquid-metal-cooled fast breeder) reactor analyses and has compared BLAST predictions with steady state and transient data taken at the AVR reactor. RWTÜV is obligated to have all parts of their codes "verified" by comparison with experimental data (preferred) or with independent codes; hence, our work on BLAST comparisons with FSV reactor data is of interest to them. A major effort at RWTÜV is to complete an extensive series of THTR postulated accident sequence analyses by May 1984, as a part of the assessment required for THTR's 30% power operating license. The plant vendor, HRB, has already done

Table 1. Nuclides for FP distribution model[a]

	KFA[b]		GAT			CEH (Table 3-1)		CEH (Periodic Table)	
Nuclide symbol	Group No.	b.p. (°C)	Group[c] No.	Group[d] No.	b.p. (°C)	m.p. (°C)	b.p. (°C)	m.p. (°C)	b.p. (°C)
Ag	3	2212		2	2177	961	1950	961	2210
Am	5	2607							
As				1	593		5615		5615
Ba	3	1640	3	2	1637	850	1140	714	1640
Br	1	57	5	1	58	-7	59	-7	58
Cd				1	765	321	767	321	765
Ce	4	3257	2	3	2927	645	1400	795	3468
Co	6	2870				1480	2900	1495	2900
Cr	6	2482				1615	2200	1875	2665
Cs	3	690	5	1	685	29	670	29	6900
Dy				3	2327			1407	2600
Eu	4	1597	4	2	1427	826	1439	826	1439
Fe	6	2750				1535	3000	1536	3000
Gd				2	2727			1312	3000
Ge				2	2827			937	2830
I	1	184	5	1	183	114	184	114	183
In				2	2047			156	2000
Kr		-152	5	1	-152	-169	-152	-157	-152
La	4	3454	2	3	3367	826	1800	920	3470
Mo	4	5560	1	3	4827	2620	3700	2610	5560
Nb	4	4927	1	3	4927			2568	3300
Nd	4	3127	2	3	3087	840		1024	3027
Np	5	3902						637	
Pa			1					(1230)	
Pd	4	3140	3	3	3127	1555	2200	1552	3980
Pm	4	2700	3	3	(2727)			(1027)	
Pr	4	3212	2	3	3017	940		935	3127
Pu	5	3327						640	.235
Rb	3	688	5	1	701	39	700	39	688
Rh	4	3730	2	3	(3727)	1995	>2500	1966	.500
Ru	4	3900	2	3	(3727)	2450	>2700	2500	4900
Sb	4	1750	5	2	1637	631	1380	631	1380
Sc	6	2832				1200	2400	1539	2730
Se	2	685	5	1	685	217	688	217	685
Sm	4	1778	3	2	(1587)	>1300		1072	1900
Sn					2687	232	2260	232	2270
Sr	3	1384	4	2	1367	800	1150	768	1380
Tb			1	3	(2527)				
Tc	4	4927	2	3	4627			2140	
Te	2	890	5	1	987	452	1390	450	990
Th	5	3800				1845	>3000	1750	3850
U	5	3818				1133	3500	1132	3818
Xe	1	-107	5	1	-108	-140	-109	-112	-108
Y	4	3337	2	3	(3227)	1490	2500	1509	2927
Zn	6	907				419	907	420	906
Zr	4	4377	1	3	4377	1700	>2900	1852	3580
Tritium	7	-253							

[a] b.p. = boiling point
m.p. = melting point
v.p. = vapor pressure
D = diffusivity
S = sublimation

[b] KFA groupings:
1. gaseous FPs
2. volatile nonmetals
3. metals adsorbing on graphite
4. nonvolatile metals
5. heavy metals
6. activation products
7. tritium

[c] GAT grouping for CORCON each group characterized by:
1. v.p. and D of PA
2. v.p. of La, D of Ce
3. v.p. of Ba, D of Ba
4. v.p. of Sr, D of Sr
5. volatile

[d] GAT grouping in FSV-FSAR
1. b.p. < 1227°C
2. intermediate between 1 and 3
3. v.p. of element or carbide >10⁻ atmosphere at 1227°C

ORNL-DWG 84-5598 ETD (part A)

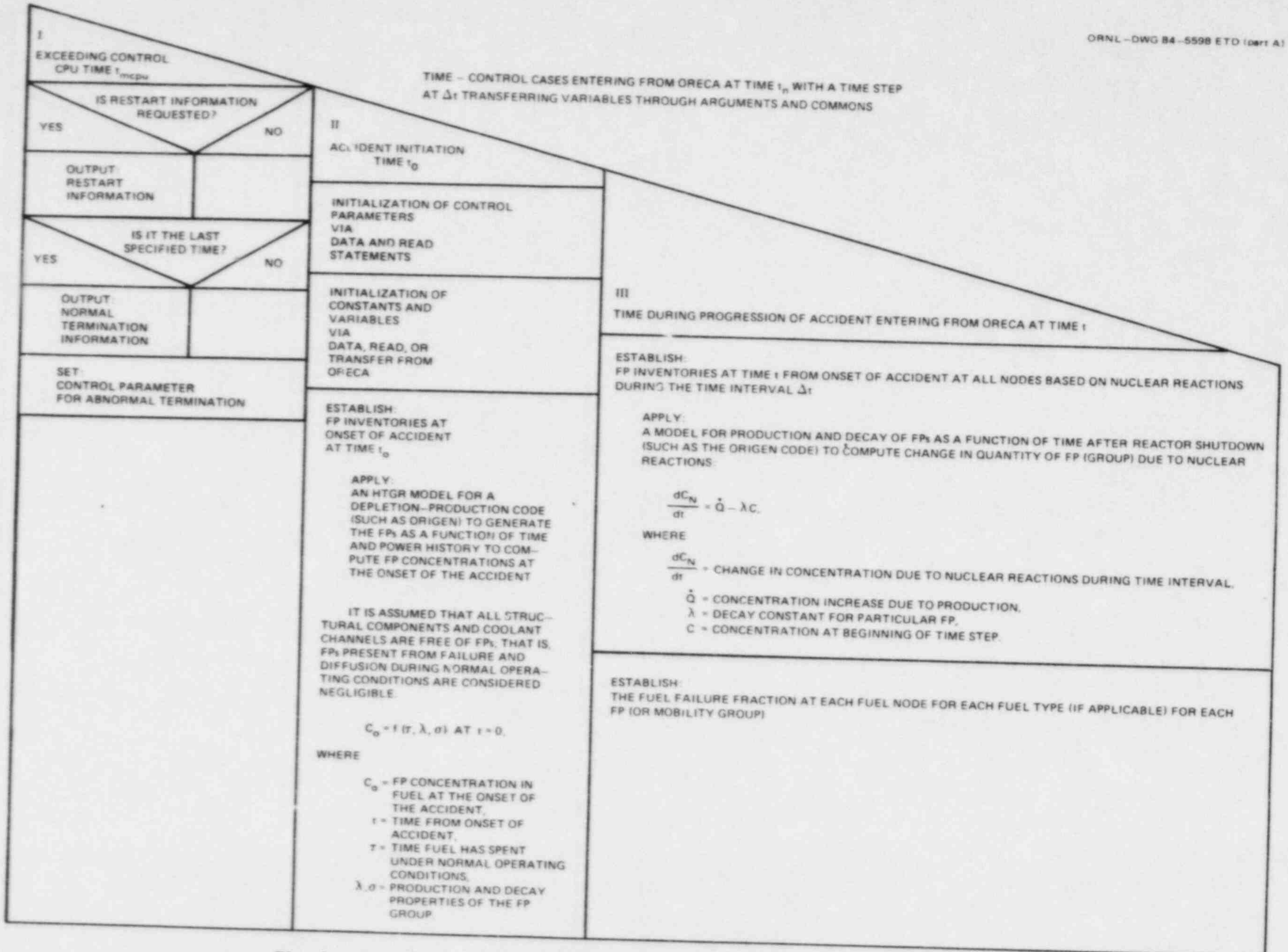

Fig. 1. Structure diagram for FP and heat source redistribution for ORECA code.

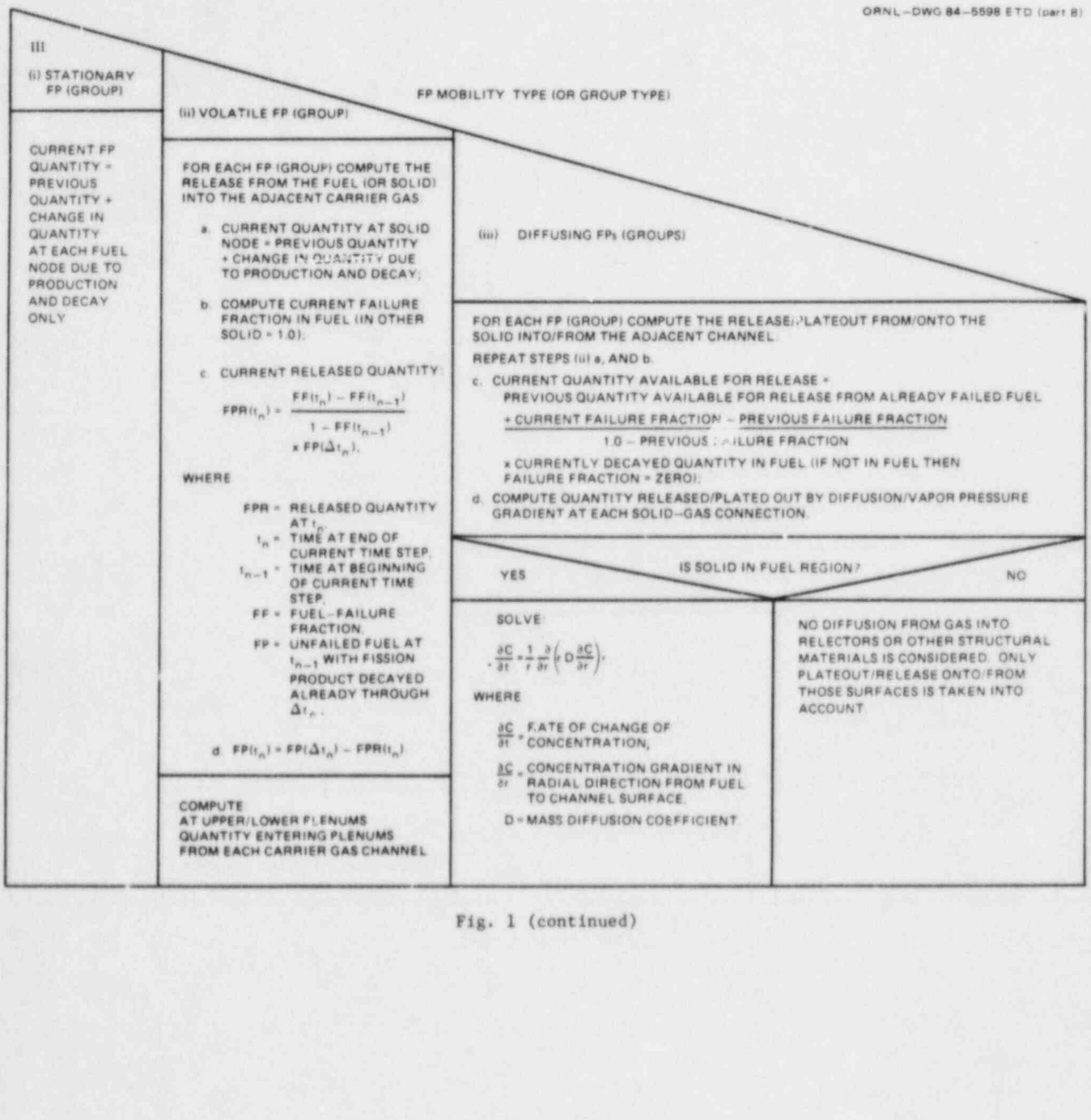

Fig. 1 (continued)

ORNL–DWG 84–5598 ETD (part C)

III (continued)

(i)

(ii)

IN EACH CHANNEL:

$$\dot{M}_{out} = \dot{M}_{in} + \sum_k \dot{M}_{sc} ,$$

WHERE

$\dot{M}_{out}$ = FLOW RATE INTO PLENUM,
$\dot{M}_{in}$ = FLOW RATE AT INLET OF CHANNEL,
$\dot{M}_{sc}$ = MASS TRANSFER RATE FROM SOLID TO CHANNEL,
k = ALL AXIAL POSITIONS ALONG CHANNEL.

COMPUTE:
AT UPPER/LOWER PLENUMS
TOTAL QUANTITY OF FP IN PLENUM (WELL–MIXED CONDITION)

$$\dot{M}_{ip} = \sum_j \dot{M}_{out} ,$$

WHERE

$\dot{M}_{ip}$ = TOTAL FLOW RATE INTO PLENUM,
$\dot{M}_{out}$ = FLOW RATE FROM EACH CHANNEL,
j = ALL CHANNELS WITH FLOW INTO PLENUM (AT PARTICULAR TIME).

(iii)

THE BOUNDARY CONDITION AT THE SOLID–GAS INTERFACE ($r = R$) IS GIVEN BY A MASS–TRANSFER CORRELATION WHICH MAY BE A NONLINEAR RELATION OF VAPOR PRESSURES:

$$\dot{M}_{sc} = hA\,(VP_s - VP_c) \text{ @ } r = R,$$

WHERE

$\dot{M}_{sc}$ = MASS TRANSFER RATE FROM GRAPHITE SURFACE TO CHANNEL,
h = MASS TRANSFER COEFFICIENT,
A = CROSS–SECTIONAL AREA NORMAL TO MASS TRANSFER,
VP_s = VAPOR PRESSURE @ GRAPHITE SURFACE,
VP_c = VAPOR PRESSURE IN CHANNEL.

@ $r = 0$ $\frac{\partial C}{\partial r} = 0$ BECAUSE OF SYMMETRY.

COMPUTE:
AT UPPER/LOWER PLENUMS
TOTAL QUANTITY OF EACH FP (GROUP) ENTERING PLENUM FROM EACH GAS CHANNEL.

ASSUMING:
A WELL–MIXED CONDITION @ STEADY STATE AXIAL FLOW IN THE CHANNELS, THEN AT EACH AXIAL POSITION IN THE CHANNEL

$$\dot{M}_{out} = \dot{M}_{in} + \dot{M}_{sc} ,$$

WHERE

$\dot{M}_{out}$ = AXIAL DOWNSTREAM FLOW RATE OUT OF CHANNEL,
$\dot{M}_{in}$ = AXIAL INLET FLOW RATE FROM UPSTREAM,
$\dot{M}_{sc}$ = RADIAL FLOW RATE FROM ADJACENT SOLID.

AT THE LAST AXIAL POSITION DOWNSTREAM $\dot{M}_{out}$ IS THE FLOW RATE INTO THE PLENUM.

Fig. 1 (continued)

ORNL-DWG 84-5596 ETD (part D)

III (continued)

(i)

(ii)

(iii)

COMPUTE:
AT UPPER/LOWER PLENUMS
TOTAL QUANTITY OF FP IN PLENUM (WELL-MIXED CONDITION)

$$\dot{M}_{ip} = \sum_j \dot{M}_{out} ,$$

WHERE

$\dot{M}_{ip}$ = TOTAL FLOW RATE INTO PLENUM,
$\dot{M}_{out}$ = FLOW RATE FROM EACH CHANNEL,
j = ALL CHANNELS WITH FLOW INTO PLENUM (AT PARTICULAR TIME).

COMPUTE:
QUANTITY RELEASED/PLATED OUT FROM/ONTO PLENUM SURFACES USING A VAPOR PRESSURE CORRELATION (NOT NECESSARILY THE FOLLOWING):

$$\dot{M}_{ps} = hA\,(VP_{ps} - VP_g),$$

WHERE

$\dot{M}_{ps}$ = MASS TRANSFER RATE FROM PLENUM SURFACES TO GAS,
h = MASS TRANSFER COEFFICIENT,
A = EFFECTIVE GEOMETRIC FACTOR FOR CROSS-SECTIONAL AREA,
VP_{ps} = VAPOR PRESSURE ON SURFACE,
VP_g = VAPOR PRESSURE IN GAS.

IS THERE LEAKAGE OUT OF PCRV?

YES

NO

Fig. 1 (continued)

III (continued)

(i)

(ii), (iii)

COMPUTE:

$$M_1 = F_g(M_{ip} - M_p),$$

WHERE

M_1 = QUANTITY OF FP LEAKED OUT OF PCRV.
F_g = FRACTION OF PLENUM GAS LEAKED OUT OF PCRV
M_{ip} = TOTAL QUANTITY OF FP ENTERED FROM ALL GAS CHANNELS.
M_p = TOTAL QUANTITY OF FP THAT PLATED OUT/RELEASED ONTO/FROM PLENUM SURFACES.

COMPUTE:
QUANTITY OF FP (GROUP) REMAINING IN UPPER/LOWER PLENUM

$$\dot{M}_{net} = \dot{M}_{ip} - \dot{M}_p - \dot{M}_1 ,$$

WHERE

$\dot{M}_{net}$ = NET FLOW RATE INTO PLENUM.
$\dot{M}_{ip}$ = TOTAL FLOW RATE ENTERING PLENUM FROM ALL CHANNELS WITH INFLOW.
$\dot{M}_p$ = TOTAL PLATEOUT/RELEASE RATE FROM PLENUM SURFACES (FOR VOLATILES ALWAYS = 0).
$\dot{M}_1$ = LEAKAGE RATE OUT OF PCRV.

COMPUTE:
EACH FP (GROUP) INLET QUANTITY INTO GAS CHANNELS FROM UPPER/LOWER PLENUMS FOR THE NEXT TIME STEP

$$\dot{M}_i = \dot{M}_{net}\, v_i / \Sigma v_i ;$$

WHERE

$\dot{M}_i$ IS THE MASS FLOW RATE ENTERING THE INDIVIDUAL CHANNEL i FROM THE PLENUM, WEIGHTED BY THE VELOCITY v_i OF THE CARRIER GAS IN THAT CHANNEL.

CONVERT:
EACH FP QUANTITY AT EACH NODE TO A HEAT GENERATION RATE

SUM UP:
ALL HEAT GENERATION RATE CONTRIBUTIONS FROM ALL THE FPs (GROUPS) AT EACH NODE

IS IT REGULAR PRINTOUT TIME?

YES | NO

OUTPUT:
REQUESTED INFORMATION

RETURN INFORMATION TO ORECA

Fig. 1 (continued)

these analyses using their independent codes (which have typically checked out well at RWTÜV's codes), and thus no serious problems are expected. Mr. Paul hopes to discuss results of RWTÜV's analyses with ORNL next spring as a part of an overall cooperation between NRC and the Ministry of the Interior (BMI) of FRG. Such collaboration would be very beneficial to both ORNL and RWTÜV and should be pursued

2. TRIP MADE UNDER PROGRAM SPONSORSHIP - VISIT TO PUBLIC SERVICE COMPANY OF COLORADO IN DENVER

R. M. Harrington

R. M. Harrington visited R. M. Bliss of PSC in Denver on October 14, 1983, to discuss several topics pertaining to the FSV reactor, including Bliss' new version of the ORNL ORECA code, FSV data for use in verifying the BLAST steam generator code, the ORNL proposal to do code verification on ORNL's modeling of the PCRV LCS, and the current severe accident version of the ORECA-FSV code.

Bliss' work has been mainly reprogramming ORECA to make it more "user friendly," that is, so it can be used by more utility people. Bliss has recently developed a more elaborate core bypass flow model, incorporated it into the PSC version, and tailored it to accommodate operational data from FSV. ORNL has been reviewing PSC's ORECA model development routinely.

Bliss is assembling the package of FSV steam generator data requested by J. C. Cleveland and will send it to ORNL soon. PSC requested a current version of ORNL's BLAST code to aid in calculating change in steam conditions for various size leaks in the steam generator bellows seals.

PSC will cooperate with our proposed LCS model verification work whenever NRC indicates that this study should be done.

PSC agreed to informally review ORNL's latest ORECA-FSV severe accident model development work.

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NUREG/CR-3492
Volume 4
ORNL/TM-8921/V4
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NRC FORM 335 (11-81)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC) NUREG/CR-3492 ORNL/TM-8921/V4

4. TITLE AND SUBTITLE (Add Volume No., if appropriate)
High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Accident Evaluation Quarterly Progress Report, October 1 - December 31, 1983

2. (Leave blank)

3. RECIPIENT'S ACCESSION NO.

7. AUTHOR(S)
S. J. Ball, J. C. Cleveland, R. M. Harrington, I. Siman-Tov, J. H. Wilson

5. DATE REPORT COMPLETED

MONTH	YEAR
May	1984

9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Oak Ridge National Laboratory
P. O. Box X
Oak Ridge, Tennessee 37831

DATE REPORT ISSUED

MONTH	YEAR
June	1984

6. (Leave blank)

8. (Leave blank)

12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS (Include Zip Code)
Division of Accident Evaluation
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

10. PROJECT/TASK/WORK UNIT NO.

11. FIN NO.
B0122

13. TYPE OF REPORT
Quarterly

PERIOD COVERED (Inclusive dates)
October 1 - December 31, 1983

15. SUPPLEMENTARY NOTES

14. (Leave blank)

16. ABSTRACT (200 words or less)

Development work continued on models and codes for predicting source terms in both the Fort St. Vrain (FSV) and 2240-MW(t) lead plant reactors. Experimental work on fission-product vapor pressures and diffusion rates through graphite continued on temperatures up to 2775 K, and a mathematical model of the experimental system was developed to aid analysis of the results and to guide improvements in the system and experiment design. Benchmarking of the BLAST steam generator code continued using FSV data, and more support work was done for proposed FSV core bypass flow model verification. Progress was made in setting up cooperative high-temperature gas-cooled reactor (HTGR) safety research with the Federal Republic of Germany. A review of a FSV technical specification on limiting maximum core temperature was begun.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited

19. SECURITY CLASS (This report)
Unclassified

20. SECURITY CLASS (This page)
Unclassified

21. NO. OF PAGES

22. PRICE
$

NRC FORM 335 (11-81)

120555078077 1 1ANIR8
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