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

J. P. Sanders

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

J. P. Sanders, Manager

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

MARCH 1976

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

Type	For the Period Ending	Designation
Bimonthly	January 31, 1973	ORNL/TM-4752
Bimonthly	March 31, 1973	ORNL/TM-4753
Quarterly	June 30, 1973	ORNL/TM-4741
Quarterly	September 30, 1973	ORNL/TM-4739
Quarterly	December 31, 1973	ORNL/TM-4738
Quarterly	March 31, 1974	ORNL/TM-4736
Quarterly	June 30, 1974	ORNL/TM-4737
Quarterly	September 30, 1974	ORNL/TM-4806
Quarterly	December 31, 1974	ORNL/TM-5062
Quarterly	March 31, 1975	ORNL/TM-5075
Quarterly	June 30, 1975	ORNL/TM-5120
Monthly	July 31, 1975	ORNL/TM-5061
Monthly	August 31, 1975	ORNL/TM-5085
Monthly	September 30, 1975	ORNL/TM-5108
Monthly	October 31, 1975	ORNL/TM-5155
Monthly	November 30, 1975	ORNL/TM-5205
Monthly	December 31, 1975	ORNL/TM-5238
Monthly	January 31, 1976	ORNL/TM-5268
Monthly	February 29, 1976	ORNL/TM-5326

Related Topical Reports Published
During the Current Fiscal Year

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).

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).

S. K. Iskander, C. W. Collins, and J. P. Sanders, Inelastic Analysis of Two Plates Under Deformation Dependent Loads, ORNL/TM-5206 (February 1976).

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).

FOREWORD

This is the seventh in a series of monthly reports that describe work performed concerning the analysis of the transient response of High-Temperature Gas-Cooled Reactors (HTGR) under abnormal conditions. The work was initiated in November 1972 under a technical assistance contract to the Gas-Cooled Reactors Branch (GCRB) of the Directorate of Licensing (DOL) of the U.S. Atomic Energy Commission. Following the reorganization of the licensing agency, the work was reported to the GCRB of the Division of Reactor Licensing (DRL) of the Office of Nuclear Reactor Regulation (NRR) of the U.S. Nuclear Regulatory Commission (NRC). Work during this period was reported in a series of bimonthly and quarterly reports.

At the beginning of the current fiscal year, the technical assistance work was transferred to the Division of Technical Review (DTR) of NRR, and progress reports were issued on a monthly basis. All coherent material has been collected and reported in a series of topical reports. Reorganization of the NRR in December 1975, resulted in the establishment of the Division of Systems Safety (DSS), to which this task is now assigned.

A complete list of all regular progress reports and current topical reports related to this task are shown in the front of this report.

During the past fiscal year, the bulk of monthly progress reports designated for distribution external to the Laboratory has been transmitted to the responsible Division of NRR. These reports were then forwarded to appropriate persons or agencies. Concomitant to the recent reorganization, the Laboratory has been requested to make the complete external distribution in accordance with a list supplied by the cognizant engineer. Any request relative to modifications in this distribution should be addressed to Mr. Wayne D. Lanning, who is serving in that capacity. Mr. Lanning is in the Division of Systems Safety, NRR, USNRC.

MONTHLY PROGRESS REPORT FOR FEBRUARY 1976

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

I. INTRODUCTION (J. P. Sanders)

During the reporting period, work has continued on the development, evaluation, and documentation of previously developed computational methods developed under this program.

Work has also continued on the ad hoc assignment that was begun during the previous reporting period relative to the evaluation of the thermal conditions in the Fort St. Vrain reactor associated with a loss of forced convection cooling (LOFC) and loss of PCRV liner cooling for a limited interval during the rise-to-power sequence. During the previous reporting period, temperatures were determined in the thermal barrier cover plates, the PCRV liner, and the concrete of the PCRV during the postulated transient. These calculations considered the conduction of heat through the core to the core barrel and across the annular helium flow channel to the wall of the PCRV. These calculations would be directly applicable if a primary system depressurization occurred during the postulated transient or if the primary system pressure could be lowered by pumping the gas through the clean-up system to storage.

If it is assumed that the primary system retains the operating inventory of helium, then natural convection currents within the PCRV cavity will redistribute the heat and produce more severe conditions on the upper thermal barrier. An evaluation of this effect has been made both for a simple model of the upper head of the PCRV and for a model that includes the added heat-transport path provided by the plug in the refueling hole penetration.

On February 24, a summary of the computational methods developed under this program was presented to Richard M. Lobel, who has been the cognizant engineer for the work during this fiscal year, and to Wayne D. Lanning, who is assuming these duties following the recent reorganization of NRR. Direction for future work under this task was discussed during this meeting.

II. THERMAL ANALYSIS FOR THE FORT ST. VRAIN HTGR (K. W. Childs, W. D. Turner)

Analysis of the loss of forced cooling (LOFC) accident coupled with the temporary loss of PCRV liner cooling for the Fort St. Vrain reactor has continued. Three models have been used in conjunction with the HEATING5 heat conduction code to evaluate the consequences of this accident during the initial rise-to-power sequence. The calculations were based on an operating power level that was 40% of the design value; allowances were made for the fact that, in the new core, afterheat generation would be less than for the equilibrium core and effective thermal conductivities in the core would be greater.

The calculations represented a 48-hr period following LOFC and reactor trip. During this period, it was assumed that the PCRV liner cooling system was inoperative. The temperatures of the thermal barrier cover plate, the PCRV liner, and the concrete of the PCRV were calculated. At the end of this 48-hr period, it was assumed that the PCRV liner cooling would be restored so that subsequent temperatures in these components would be no greater than the values calculated at the end of the 48-hr transient.

The first model developed represented the temperatures that would be developed due to the generation and conduction of heat through the reactor core. This model represented the conduction of heat through the reflectors and the core barrel and across the annular helium flow channel and the upper and lower plenum. This calculation would be applicable to the case when the LOFC was accompanied by a depressurization of the primary system. This model is shown in Fig. 1.

This first model is a 2-dimensional R-Z model. The active core is represented by four annular rings with the radial power peaking factor in each ring set equal to the average of all of the refueling regions in the ring. The axial power distribution is represented by using an average axial peaking factor for each of the six levels of fuel assemblies in the active core. The total power generated in the model is in agreement with the total power generated in the actual reactor.

Radiation boundary conditions were applied across the upper and lower plenum. Radiation and natural convection boundary conditions were

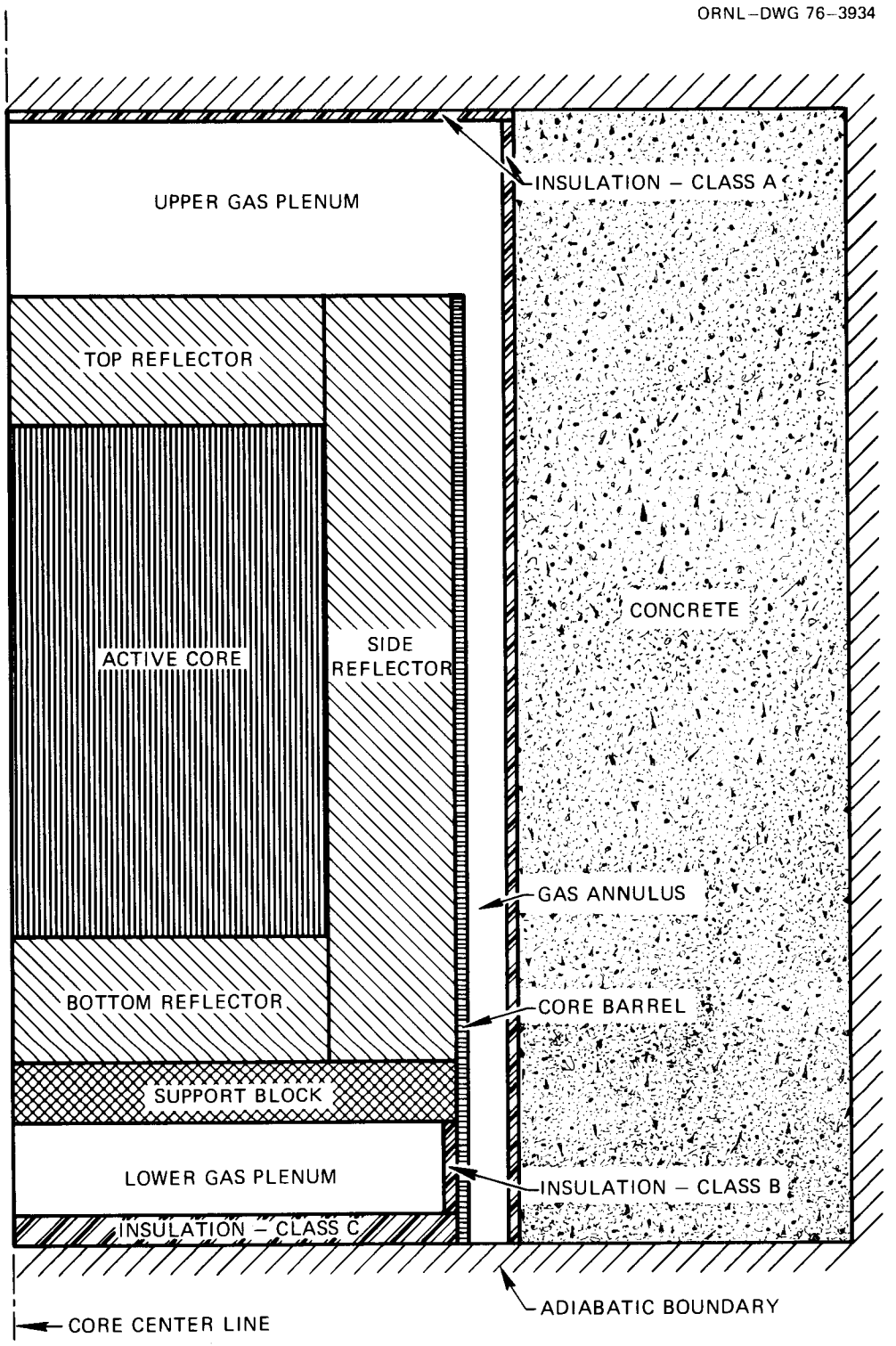

Fig. 1. Model of entire core and surrounding structures used to determine the conduction of heat to the PCRV concrete and liner.

represented in the gas annulus between the core barrel and the PCRV. The outer surfaces of the model were specified as adiabatic boundaries.

A plot of the temperatures of interest as a function of time is presented in Fig. 2. The concrete temperature predicted is slightly higher than the value that would be expected in the actual transient since only a portion of the PCRV was represented.

If the primary system of the reactor remains pressurized following the LOFC, natural convective loops are established in the core cavity due to the nonuniform power generation rate. In the regions with larger power generation rates, the helium will be heated to higher temperatures and tends to rise due to buoyant forces. In the regions with lower power, the helium, which is at lower temperatures and thus at a higher density, flows downward. These natural convection loops tend to redistribute the heat within the core from the high-power, hotter regions to the low-power, cooler regions. As a consequence, the calculation of the maximum temperatures in the absence of natural convection effects will be conservative.

The major effect of these convective loops with regard to this analysis is the formation of a column of hot helium rising from the hot region and impinging on the upper thermal barrier cover plate. The highest temperature in the cover plates will be directly over the hottest region.

The distribution of temperatures among the various regions as a function of time was determined using the HEXEREI computer program; the redistribution of heat due to natural convection transport was not included in this calculation. Using a volume-averaged, effective temperature for the coolant in each of the regions as a function of time, an independent computer program was written to estimate both the amount of coolant flow from the hottest region and the exit temperature of this helium. Based on this calculation, it was possible to estimate both the surface heat transfer coefficient and the temperature of the hot column of helium impinging on the upper thermal barrier cover plate.

In making these calculations, it was found that one of the most important assumptions used in the calculations was the effective diameter of the column of hot gas rising from the top of the core. In making these calculations, it was assumed that this was a cylindrical column with the same flow area as the vertical segment of the orifice valve at

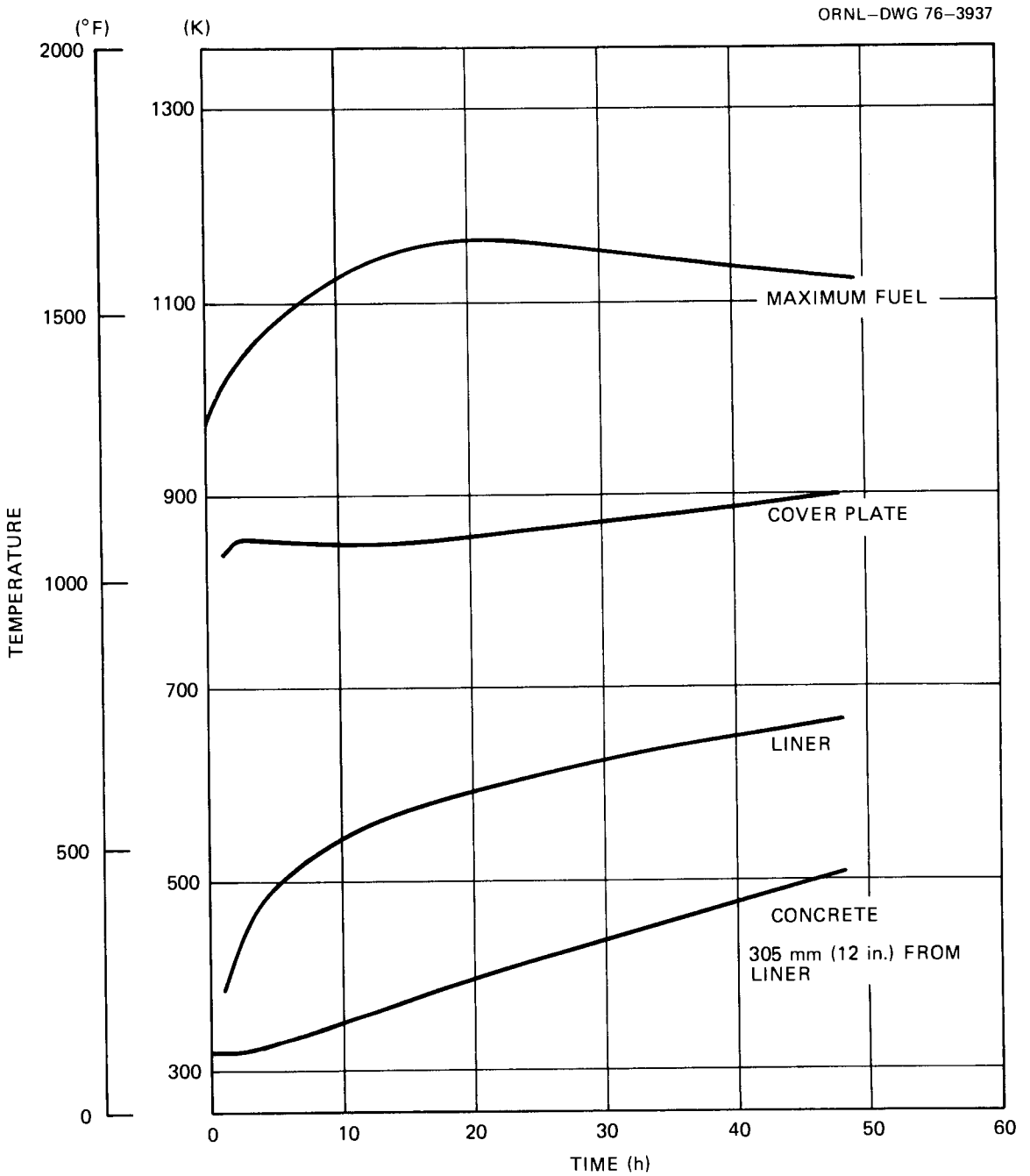

Fig. 2. Temperatures in the PCRV liner and concrete and the thermal barrier cover plate due to conduction near the midplane of the reactor.

the top of each refueling region. Values for the gas temperatures and heat transfer coefficients through the transient are given in Table 1.

Figure 3 shows the model of the upper PCRV head and insulation that was used in conjunction with these gas temperatures and heat transfer coefficients to determine the resultant temperatures in these components. This is a 1-dimensional model of a portion of the PCRV above the reactor core. The thermal barrier is modeled as a composite material rather than layers of Kaowool and steel plates. A correction factor of 1.8 is applied to the calculated effective conductivity to account for fasteners and penetrations of the thermal barrier. A radiation boundary condition is specified between the top reflector and the thermal barrier. Also a convective boundary condition is applied to the thermal barrier. The top reflector temperature, hot channel exit temperature, and convective heat transfer coefficient are taken from Table 1. The results of the analysis are presented in Fig. 4.

From a review of this analysis, it was obvious that the plugs in the upper-head penetrations of the PCRV would provide a path for the transport of heat around the thermal barrier to the PCRV concrete. To represent this effect, a third model was used. It is illustrated in Fig. 5. This is a 2-dimensional R-Z model of a symmetric portion of the PCRV above the reactor core. The center of the model is located at the center of the refueling penetration. The plug in the penetration is modeled as a cylinder of granular graphite with an annular, gas-filled ring to represent the cavities into which the control rods are withdrawn. The model is constructed so that the appropriate cross-sectional areas for the transport of heat are maintained. The radius of the model is equal to one-half of the pitch of the refueling hole penetrations. The outer surface is modeled as an adiabatic boundary. Radiation and natural convection are specified in the annular helium gap representing the control rod cavities and in the gap between the plug and the penetration liner. All other boundary conditions are the same as those used in the second model.

The temperatures that were determined from this model are shown in Fig. 6. The temperatures for the critical components determined from this model at the end of the 48-hr period are lower than those indicated by analyses submitted by the applicant. The remaining effect that is not

Table 1. Calculated effective heat transfer coefficients at upper thermal barrier

Time (hr)	Temperature (K)			Reynolds number	Heat transfer coefficient (mW/m·K)[a]
	Hot channel	Cold channel	Top reflector		
0.0	850	836	595	897	5.4
0.8	1 109	878	597	6 894	20.8
1.7	1 122	881	607	6 945	21.1
2.7	1 127	883	618	6 929	21.1
3.8	1 131	882	628	7 010	21.4
8.2	1 157	883	667	7 249	22.2
10.4	1 163	886	686	7 196	22.2
15.0	1 172	894	722	6 954	21.6
19.6	1 188	905	759	6 685	21.2
25.5	1 202	920	796	6 284	20.3
30.4	1 210	931	829	5 982	19.6
35.3	1 215	945	857	5 588	18.6
40.2	1 218	957	883	5 246	17.7
45.0	1 220	969	905	4 930	16.9
50.3	1 221	981	926	4 617	16.0

[a]To convert to Btu-hr^{-1}ft^{-2}°F^{-1}, multiply by 0.1761.

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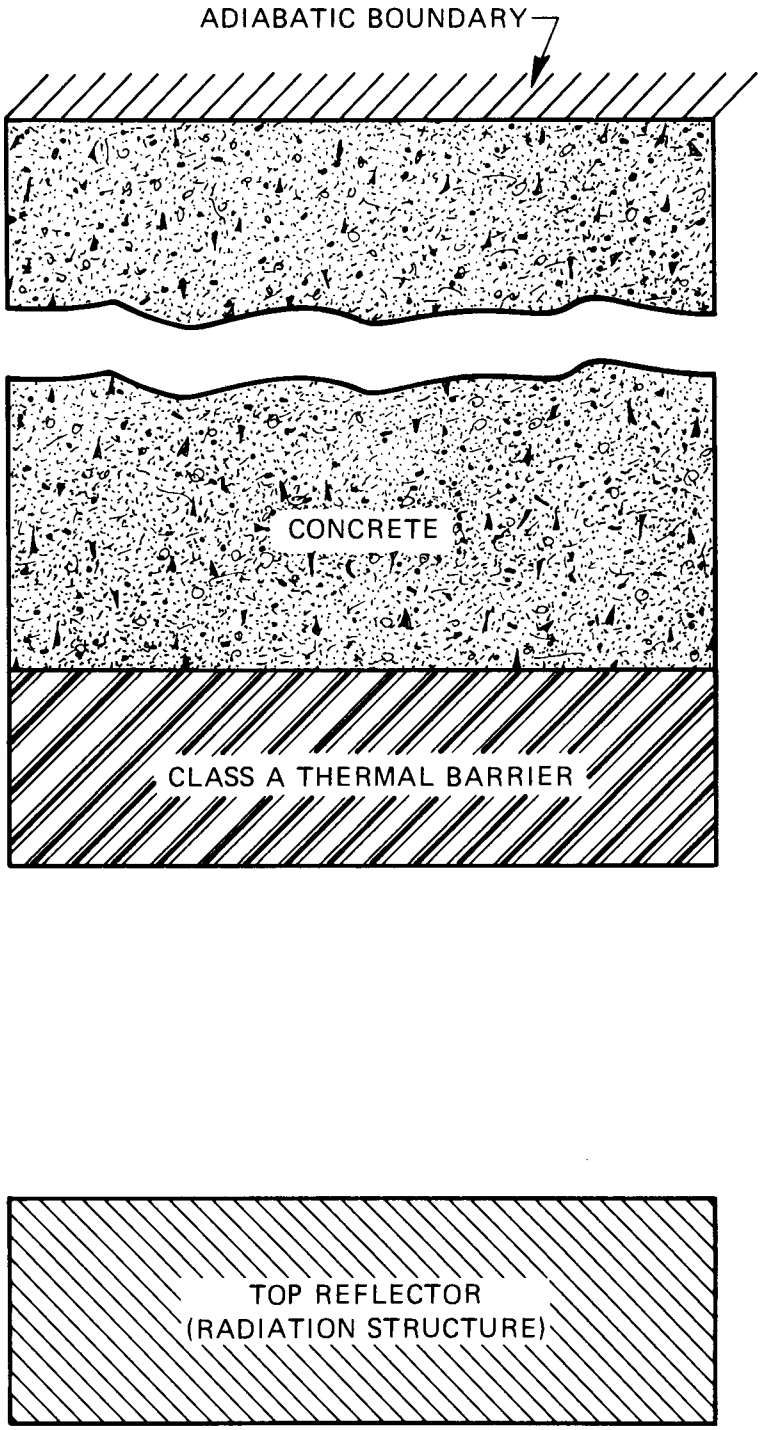

Fig. 3. Model for the transport of heat to the upper head of the PCRV including effects of natural convection and radiation from the top of the core.

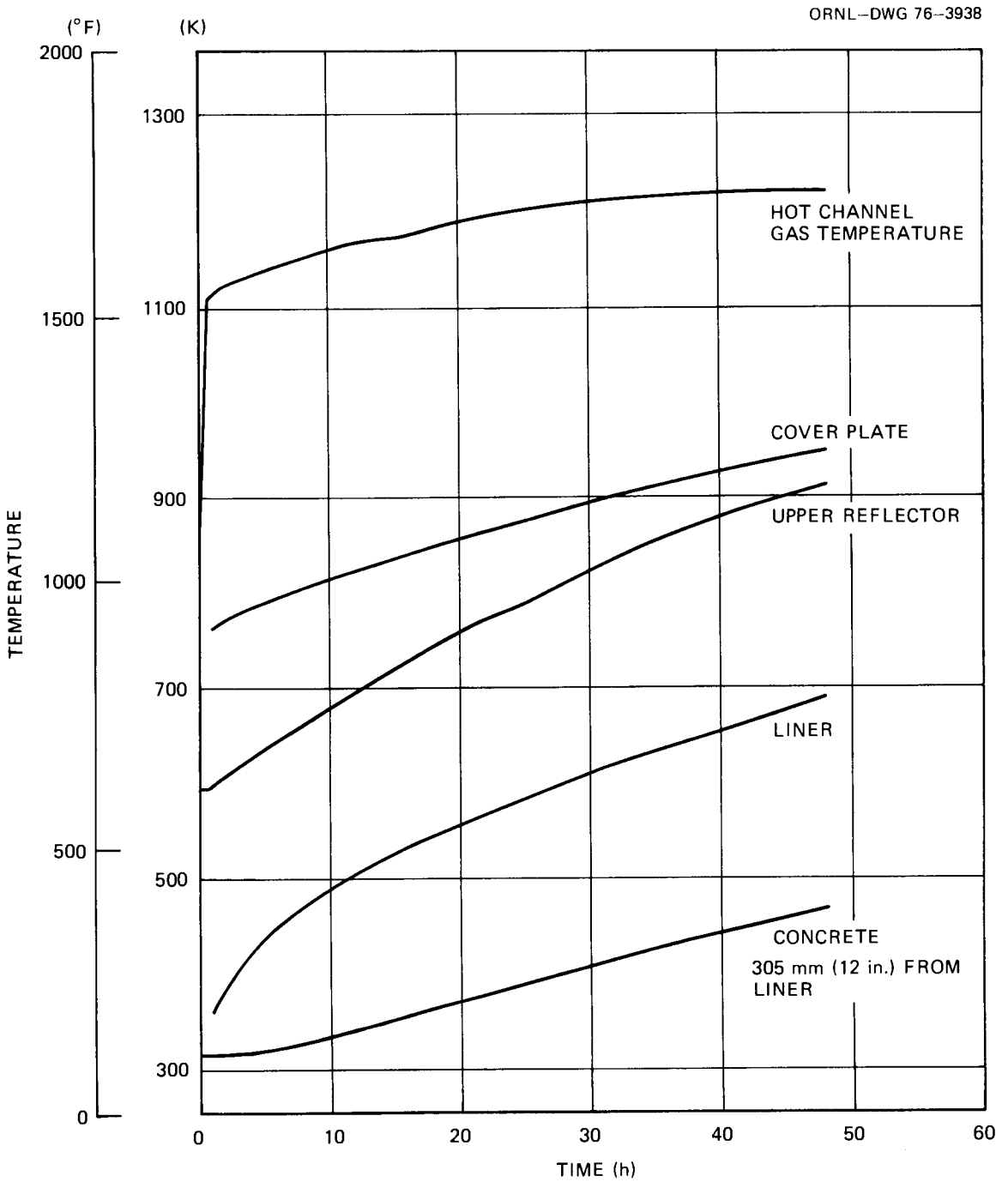

Fig. 4. Temperatures calculated in the concrete and liner of the upper head of the PCRV and in the upper thermal barrier cover plate.

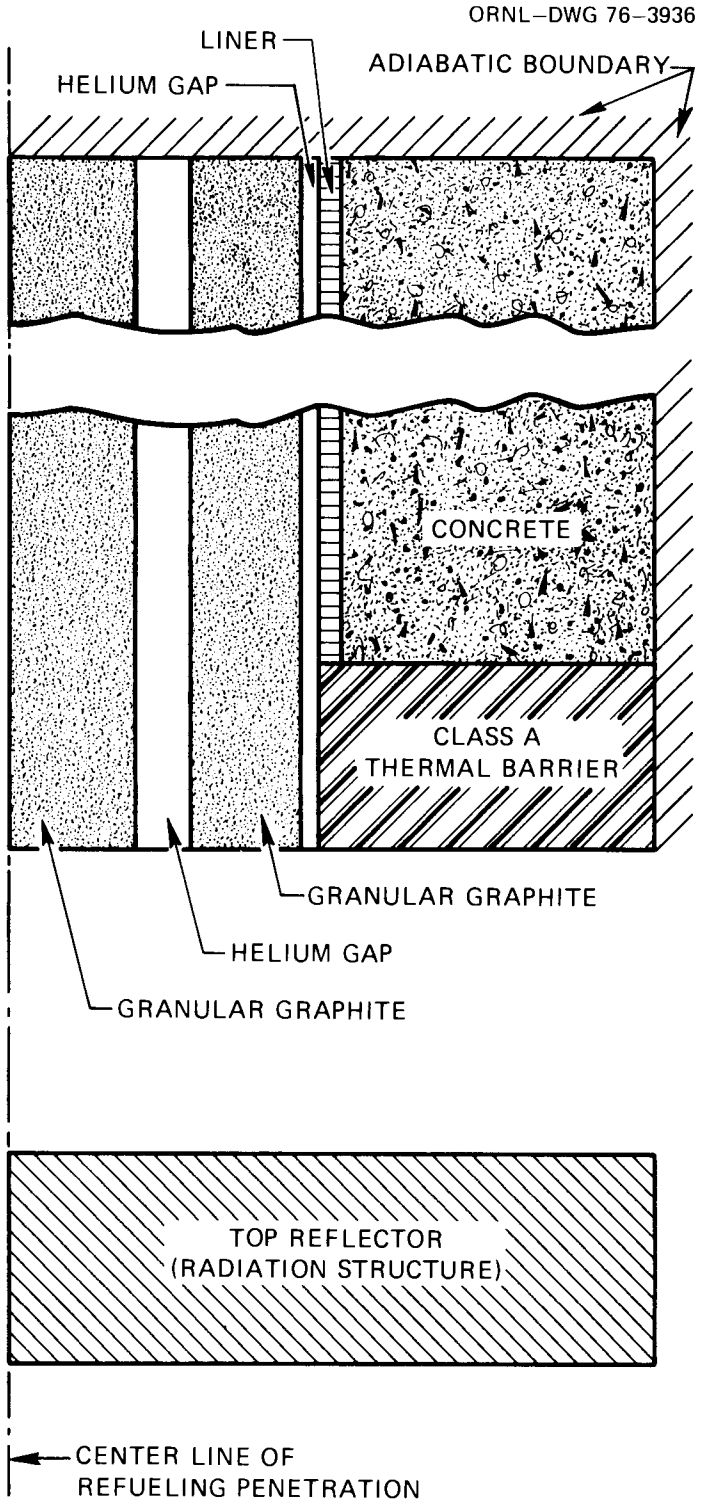

Fig. 5. Model used to represent the refueling penetration and plug of the upper head of the PCRV.

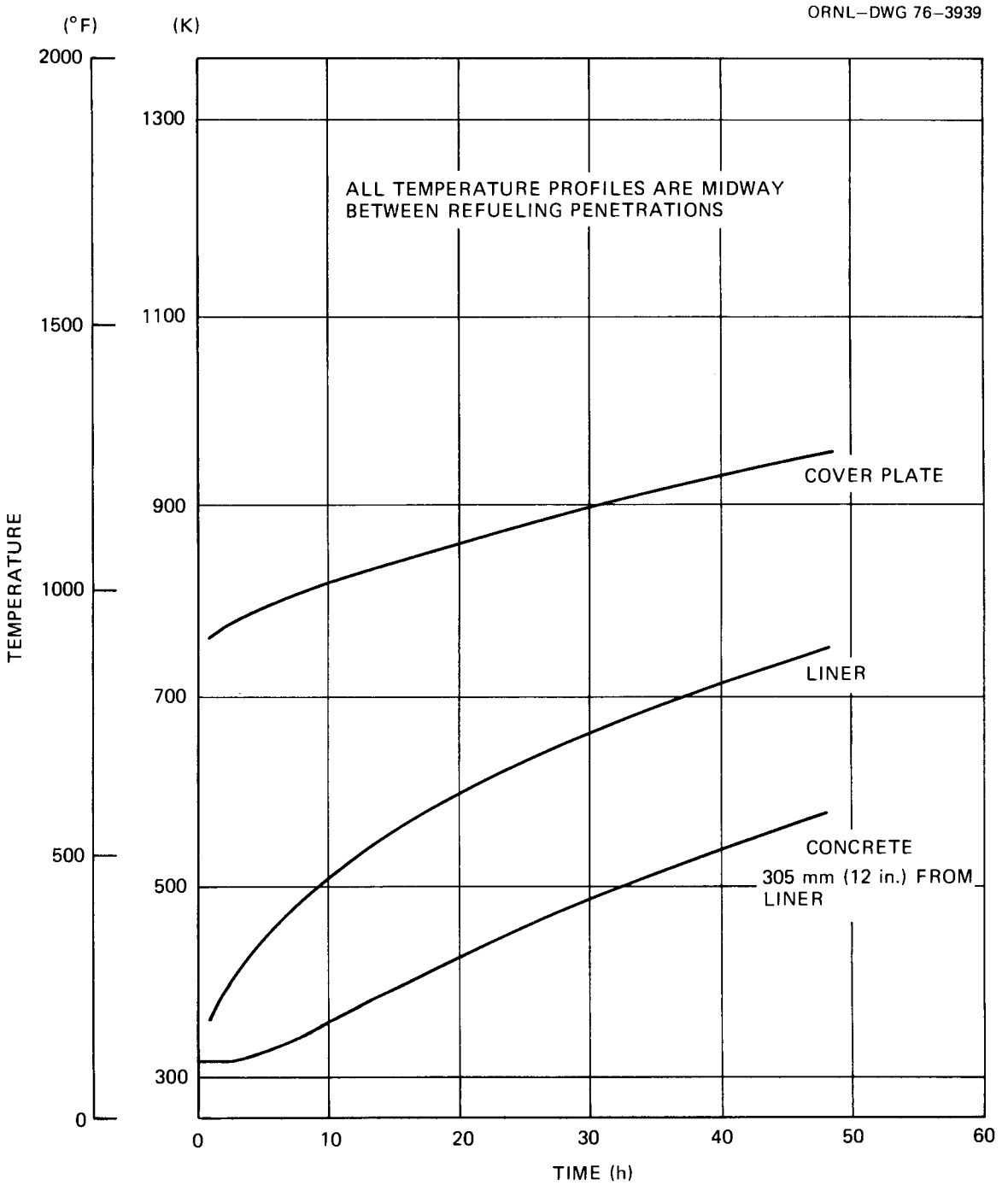

Fig. 6. Temperatures calculated in the upper head of the PCRV including the effect of heat transport in the plug for the refueling penetration.

represented in these calculations is the transport of heat by natural convection of the hot gas into the annular gap around the plug of the penetration. The end of these plugs were modified within the last year to inhibit the ingress of gas from the upper plenum. Detailed dimensions for the modified plug were not available so further meaningful computations could not be made.

III. HEXEREI CODE DEVELOPMENT (G. E. Giles, R. M. DeVault, B. R. Becker)

Output routines have been modified to give more descriptive information for the Fort St. Vrain 40% power accident. The generalized boundary conditions for a surface to a boundary temperature as exists in HEATING5 have been added. The modifications for the modeling of the core support blocks have been made, and the modifications to the solid node temperature calculations have been verified. The implicit transient technique works for the uniform flow heat generation test case. Work is now progressing to add the flow redistribution and ACHERONS to the implicit technique. The flow redistribution technique in HEXEREI underwent severe modification to correct convergence problems. The technique now works on all Fort St. Vrain test cases experienced to date (DBDA2 with 7000 rpm and 8000 rpm circulator speed and the 40% accident).

Further work is being done in analytical verification of HEXEREI boundary conditions, and coolant temperature calculation is progressing. Documentation of test cases to date is in the final stage before printing.

IV. FLODIS RESULTS FROM THE FORT ST.VRAIN DBDA (D. D. Paul)

The Fort St. Vrain rapid depressurization accident with the Pelton circulators operating at 8000 rpm was analyzed using the FLODIS code. A comparison of the results of the analysis with the critical safety limits is given in Table 2. None of the critical safety limits have been exceeded during this most severe accident sequence.

Table 2. Critical safety limits

Material or component	Critical safety limit	FLODIS maximum temperature
Coated fuel particles		
Value, °C	1870	1550
Value, °F	2900	2323
Silica thermal barrier on the surface of the core support floor		
Value, °C	1640	1480
Value, °F	2500	2200
Thermal barrier lining of the steam generator inlet ducts		
Value, °C	1370	1280
Value, °F	2000	1850[a]

[a]This result should be considered the best estimate based on experimental mixing data from GA-7638.

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