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High-Temperature Gas-Cooled Reactor Safety Studies for the Division of Reactor Safety Research Quarterly Progress Report, January 1–March 31, 1979

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Prepared for the U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
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HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF REACTOR SAFETY RESEARCH QUARTERLY PROGRESS REPORT, JANUARY 1–MARCH 31, 1979

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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).

J. C. Cleveland, CORTAP: A Coupled Neutron Kinetics—Heat Transfer Digital Computer Program for the Dynamic Simulation of the High Temperature Gas Cooled Reactor Core, ORNL/NUREG/TM-39 (January 1977).

J. C. Cleveland et al., ORTAP: A Nuclear Steam Supply System Simulation for the Dynamic Analysis of High Temperature Gas Cooled Reactor Transients, ORNL/NUREG/TM-78 (September 1977).

S. J. Ball et al., Evaluation of the General Atomic Codes TAP and RECA for HTGR Accident Analyses, ORNL/NUREG/TM-178 (May 1978).

FOREWORD

HTGR safety studies at Oak Ridge National Laboratory (ORNL) are sponsored by the Division of Reactor Safety Research, which is part of the Office of Nuclear Regulatory Research of the Nuclear Regulatory Commission (NRC).

This report covers work performed from January 1 to March 31, 1979. Previous quarterly reports and topical reports published to date are listed on p. v. Copies of the reports are available from the Technical Information Center, U.S. Department of Energy, Oak Ridge, Tenn. 37830.

HIGH-TEMPERATURE GAS-COOLED REACTOR SAFETY STUDIES FOR THE DIVISION OF REACTOR SAFETY RESEARCH QUARTERLY PROGRESS REPORT, JANUARY 1—MARCH 31, 1979

S. J. Ball, Manager
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ABSTRACT

Further development work was done on the ORECA, BLAST, and FLODIS codes. BLAST was adapted to the West German THTR steam generator design. The refurbishing of the FLODIS code was completed, and sample calculations of design-basis depressurization accidents (DBDAs) and loss-of-forced-convection (LOFC) accidents were run to make comparisons with other calculations. Followup investigations were also made of the Fort St. Vrain (FSV) reactor oscillation problems.

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1. HTGR SYSTEMS AND SAFETY ANALYSIS

S. J. Ball

Work for the Division of Reactor Safety Research (RSR) under the High-Temperature Gas-Cooled Reactor (HTGR) Systems and Safety Analysis Program began in July 1974, and progress is reported quarterly. Work during the present quarter included further work on code development and assistance to the Nuclear Regulatory Commission (NRC) on Fort St. Vrain (FSV) reactor licensing questions.

1.1 Development of the ORECA Code for Core Simulations During Emergency Cooling Transients

S. J. Ball

Several improvements and modifications were made to the ORECA code[1] during the quarter. It now includes a calculation of the 12 individual steam generator inlet temperatures based on flow-weighted averages of the appropriate refueling region outlet temperatures. ORECA was also modified to include options for specifying either refueling region orifice positions, orifice loss coefficients, or initial region outlet temperatures as input parameters. Originally, the first two options were not available, and the orifice coefficients used were those required to give the specified initial outlet temperatures. While making these modifications, it was discovered that the flow areas of the 6 five-element refueling regions were not properly represented in the flow calculations. Correction of this problem had only a minimal effect on the results.

1.2 Investigations of the FSV Oscillation Problem

S. J. Ball

Reviews of current FSV oscillation data and analysis of the data by General Atomic (GA) were continued as functions of the special NRC Technical Review Committee. Comments and questions were forwarded to NRC.

A special modification of the ORECA code was written to calculate the equivalent time-varying refueling region orifice coefficients required to yield agreement between calculated and observed region outlet temperatures during oscillation events. The purpose is to try and rationalize the measurements with postulated region leakage flow changes, and hence confirm the validity of the GA "jaws" theory.[2] The major event of interest is the large temperature upset of November 4, 1978, at 0410 hr. The program modifications are in the debugging stage.

1.3 Visit to RWTUV to Assist in Adaptation of the BLAST Steam Generator Simulation to the THTR Design

J. C. Cleveland

J. C. Cleveland visited Rheinisch-Westfalischer Technischer Uberwachungs-Verein e.V. (RWTUV), Essen, West Germany, to assist in adapting the BLAST computer program[3] (for the dynamic simulation of HTGR steam generators) so that it can be used to model the transient behavior of the THTR steam generator. All travel costs associated with the visit were paid by RWTUV.

RWTUV personnel intend to use the BLAST code for analyses of steam generator transient performance in their technical review of the safety of the 300-MW(e) demonstration THTR under construction at Schmehausen. BLAST is a component simulation for the HTGR reheater and steam generator module. It has been used extensively at Oak Ridge National Laboratory (ORNL) in transient analyses for the FSV HTGR, both as a component simulation and as a part of the ORTAP-FSV code, a total system simulation.[4] RWTUV personnel obtained ORTAP, BLAST, and other component simulations and supporting documentation through the request of Heinz G. Seipel, Head of Nuclear Safety Research Section, Der Bundesministerium fur Forschung und Technologie (BMFT) to Saul Levine, Director, Office of Nuclear Regulatory Research, United States Nuclear Regulatory Commission (USNRC). RWTUV personnel will use BLAST during the licensing phases involved in the construction and operation of the THTR-300 at Schmehausen, and they also plan to use BLAST in analyses of future high-temperature reactor steam generator designs.

Additionally, the RWTUV staff, with the assistance of the Institute fur Reaktorentwicklung at KFA, is developing a system simulation for the THTR-300. For this simulation, BLAST will be coupled with models of the core, helium blowers, turbine generator plant, and the control and safety system. The KFA staff is responsible for simulation of the primary circuit thermal hydraulics and neutron kinetics. In addition to the steam generator simulation, RWTUV personnel are responsible for simulation of the turbine plant, the system for feedwater cooling of live steam, the reheater bypass system, and the flash tank. They are also responsible for the plant control and safety system simulation. The system simulation is planned for completion in about one year. It will be used for analyses needed to evaluate the applicant's request for an operation license.

During the 2-week visit, the following tasks were accomplished.

1. The THTR-300 reheater-steam generator module was modeled with BLAST. The model utilized 13 water nodes, 13 tube nodes, and 13 helium nodes. In the THTR steam generator, helium flow is upward, countercurrent to the water/steam flow in the main steam bundle and countercurrent to steam flow in the reheater. Several successful test cases for both steady-state and transient conditions were performed. RWTUV personnel compared results (Reynolds number, heat transfer coefficients, outlet

helium and steam temperatures, and pressure drops) with calculations provided by the manufacturer. Agreement was quite good.

2. A flash tank model was developed and coupled to the steam generator and reheater models. The flash tank is used during startup and shutdown to provide dry steam to the reheater when the high-pressure turbine is being bypassed. This model was tested for both steady-state and transient conditions.

3. A model was developed for the partial steam bypass of the reheater used in the THTR-300. This partial bypass is used to limit the inlet temperature to the intermediate pressure turbine. Bypassed steam is mixed with reheated steam upstream of the high-pressure turbine. Initial programming was performed. Additional checking of the model and results at ORNL may be necessary.

4. A model of a moisture separator in the evaporator section of the main steam bundle was developed and integrated into a special version of BLAST. This model was developed, not for the THTR-300, but for a potential steam generator design of future HTRs. Use of a moisture separator provides a fixed boundary for single-phase steam over the entire operational range, thus increasing steam generator stability. Several test cases were performed.

5. An alternate integration technique (explicit) was inserted into BLAST, and results for transient test cases were compared with results obtained by the standard implicit integration technique. The explicit technique will save computing time for models employing large numbers of nodes. The explicit technique had been developed at ORNL; however, it had not been extensively tested or used, because the implicit technique is preferable for the models used at ORNL, which typically use from 10 to 20 nodes each for water, tubes, and helium.

6. A restart option (both from steady-state and transient conditions) was developed for BLAST. Several test cases were executed. RWTUV gave this fairly high priority due to the fact that their current computer is fairly slow.

7. The functioning of certain subroutines within BLAST was discussed with the RWTUV staff so that they can make changes as desired (e.g.,

inclusion of different tube material properties or alternate heat transfer correlations).

These tasks were achieved with the efficient help of two full-time (and overtime) computer programmers. Each model and modification to BLAST developed during the 2-week period will be made available to ORNL personnel by the RWTUV staff. Also, a version of BLAST that has been prepared by RWTUV personnel with input and output in SI units will be provided to the ORNL staff. RWTUV personnel have recently informed those at ORNL that improvements have been made in a matrix inversion subroutine within BLAST resulting in roughly a 50% decrease in computer time requirements. This modification will be sent to ORNL after calculations are checked by the RWTUV staff for incorporation into the ORNL version.

RWTUV personnel expressed considerable interest in ORNL comparisons of BLAST, ORECA, and ORTAP results with data from FSV transients, and they will attempt to obtain AVR steam generator transient data for comparisons with RWTUV predictions obtained with BLAST. Following startup of the THTR-300, RWTUV personnel also plan comparisons of BLAST with measured plant data. These comparisons would be of considerable interest at NRC and ORNL.

Discussions were held with members of the Institute fur Nukleare Sicherheitsforschung and the Institut fur Reaktorentwicklung at Kernforschungsanlage (KFA), Julich, concerning their THTR analysis program and the ORNL effort in the HTGR Safety Research Program. Also, experimental facilities in the Institut fur Reaktorbauelemente at KFA were visited. These facilities involved measurements of heat transfer and pressure drop characteristics of pebble bed reactors as well as flow-induced vibration characteristics of heat exchanger tube banks.

A meeting was held with the staff of Hochtemperatur Kernkrafwerk Gesellschaft (HKG) at Schmehausen, and a tour of the THTR was provided.

Many documents were provided by RWTUV, KFA, and HKG based on questions asked by the traveler. These are listed in the traveler's trip report.[5]

1.4 Impact of Control System Behavior on Plant Safety

S. J. Ditto

During the reporting period, a small effort has begun on an evaluation of the control schemes of the HTGR with the aim of a better understanding of the impact of control system behavior on plant safety. Not only can the protection be enhanced by timely and proper operaton of the controls during minor disturbances, but the control system can contribute to the initiation of transients by its malfunction. Both areas need to be explored.

Initial examination of the system leads one to suspect that interactions between, or imbalances among, the 12 steam generators, 2 helium loops, and 4 circulators could be troublesome. Analyses that assume well-behaved and symmetrical systems could fail to disclose certain types of instability. An example that might be considered involves control of both boiler feed pumps from a selected (low select) differential pressure across the feedwater valves. Other examples exist and will be identified in future reports.

1.5 Development of the FLODIS Code for FSV Emergency Core-Cooling System (ECCS) Analysis

J. C. Conklin S. J. Ball

The rewrite and debugging of the FLODIS code[6] was completed as of Feb. 28, 1979. Additional subroutines and modifications were implemented to compare the results of ORECA and FLODIS in a meaningful manner.

FLODIS models four rectangular subregions for each of the 31 seven-column refueling regions and three rectangular subregions for each of the 6 five-column refueling regions. Hence, FLODIS can calculate the intra-regional flow distribution as well as the interregional flow distribution.

Two accidents were investigated, the design-basis depressurization accident (DBDA) and the loss-of-forced-convection (LOFC) accident, and, for each accident, two cases with different initial core pressure drops are presented. For the first case, refueling region 6 has a wide-open orifice with a pressure loss of 4.0 velocity heads.[7] The remaining orifice loss coefficients are set so that the refueling region flow rate results in the specified EQSB3 refueling region outlet temperature. This resulted in an initial core pressure drop of 6.25 psi. For the second case, the orifice loss coefficient for region 6 was set so that the initial core pressure drop was 10.0 psi. The remaining loss coefficients were set as in the first case.

1.5.1 DBDA analyses

The results of FLODIS and ORECA for the first 6 hr of the DBDA are presented in Figs. 1 through 4. The basic core performance parameters are as follows:

1. EQSB3 peaking factors, region outlet temperatures, helium inlet temperature, and helium flow from Ref. 8;
2. helium pressure at 12.3 psia for the duration of transient;[8]
3. graphite thermal conductivity of 10.0 Btu/hr-ft°F from the FSAR;[9]
4. core afterheat as a function of time from LTR-1.[10]

Figure 1 presents the FLODIS calculated core and coolant temperatures for the 6.25-psi case. The maximum calculated subregion core temperature for the DBDA is 2725°F and occurs at 182.5 min into the transient, and the maximum refueling region average temperature is 2635°F and occurs at 172.5 min. Both of these differences are due to a "positive feedback," or autocatalytic, effect of helium temperature on flow distribution.

This autocatalytic effect arises from helium physical property relationships. The viscosity of helium increases with increasing temperature; therefore, the helium flow resistance would increase with hotter helium temperature, which would, in turn, decrease the helium flow for that subregion and consequently increase the helium temperature. Conversely, a cold subregion, having less flow resistance, would have an increased helium flow and would cool faster than the hot subregions.

Figure 1 shows this effect for intraregional flow within the core refueling region having the maximum temperature. The difference between the maximum subregion temperature and the maximum region average (which includes the maximum subregion) temperature is zero at the start of the transient and increases to 140°F at 240 min into the transient. After

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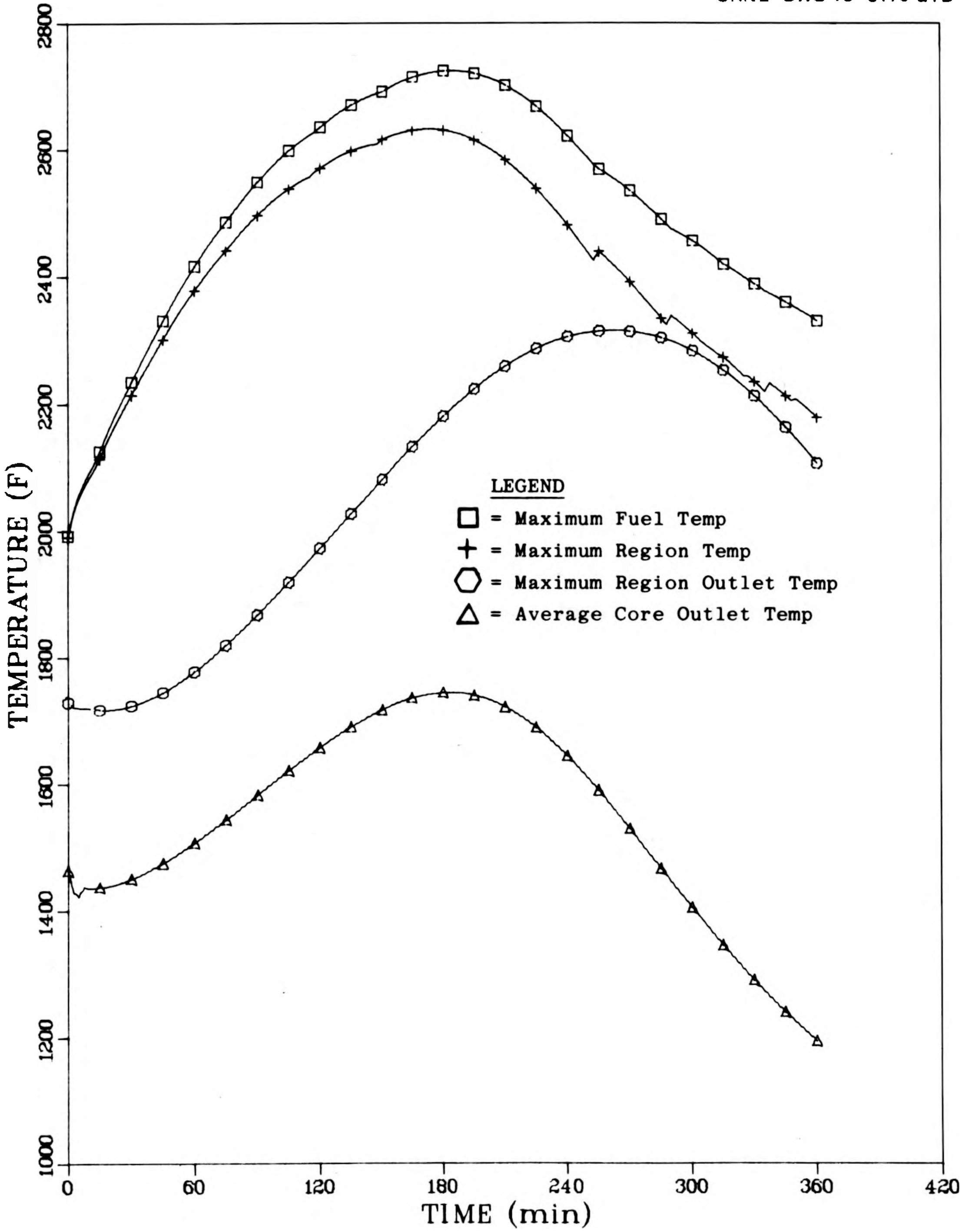

Fig. 1. FLODIS core thermal response for a 6.25-psi initial pressure drop, DBDA.

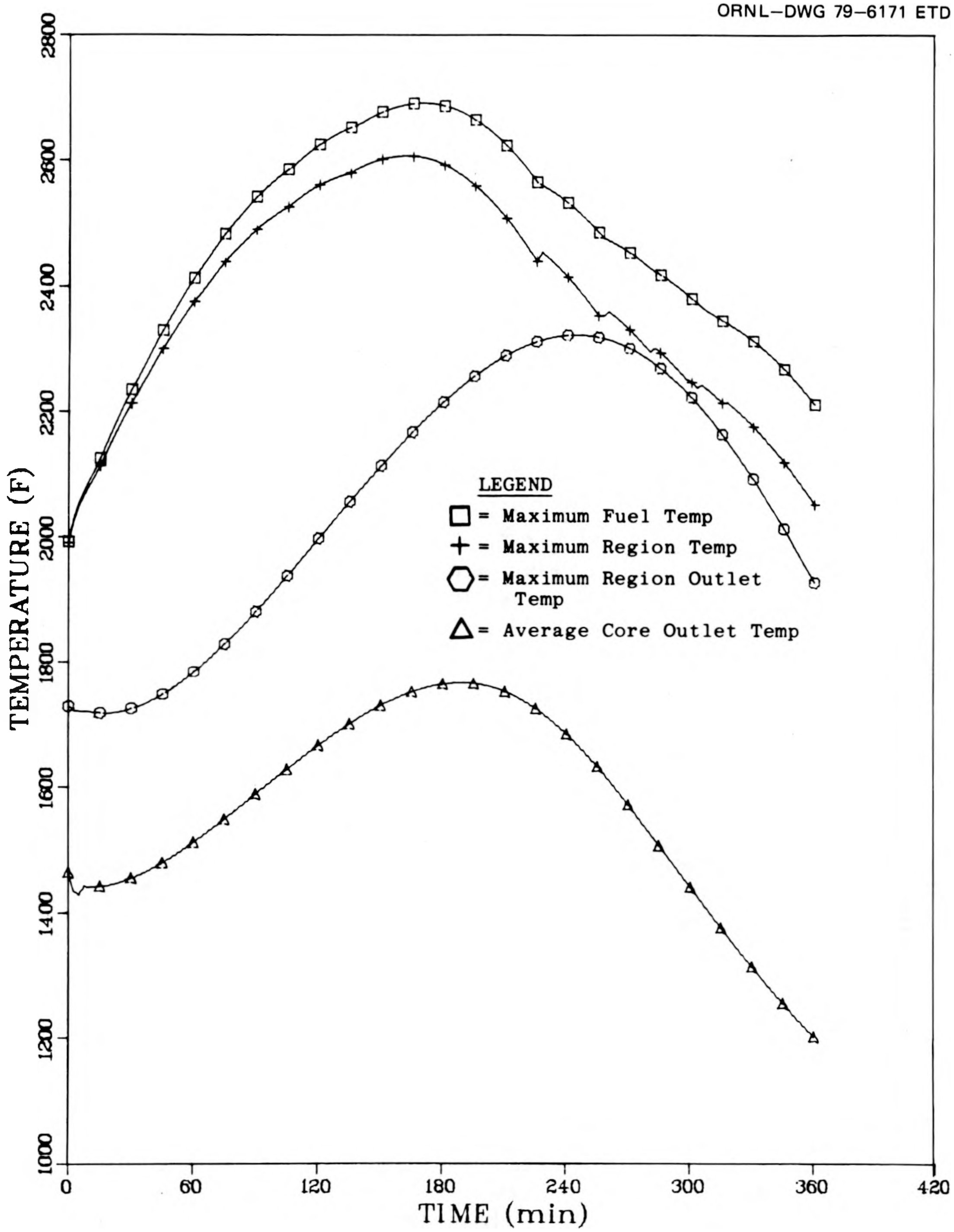

Fig. 2. FLODIS core thermal response for a 10.0-psi initial pressure drop, DBDA.

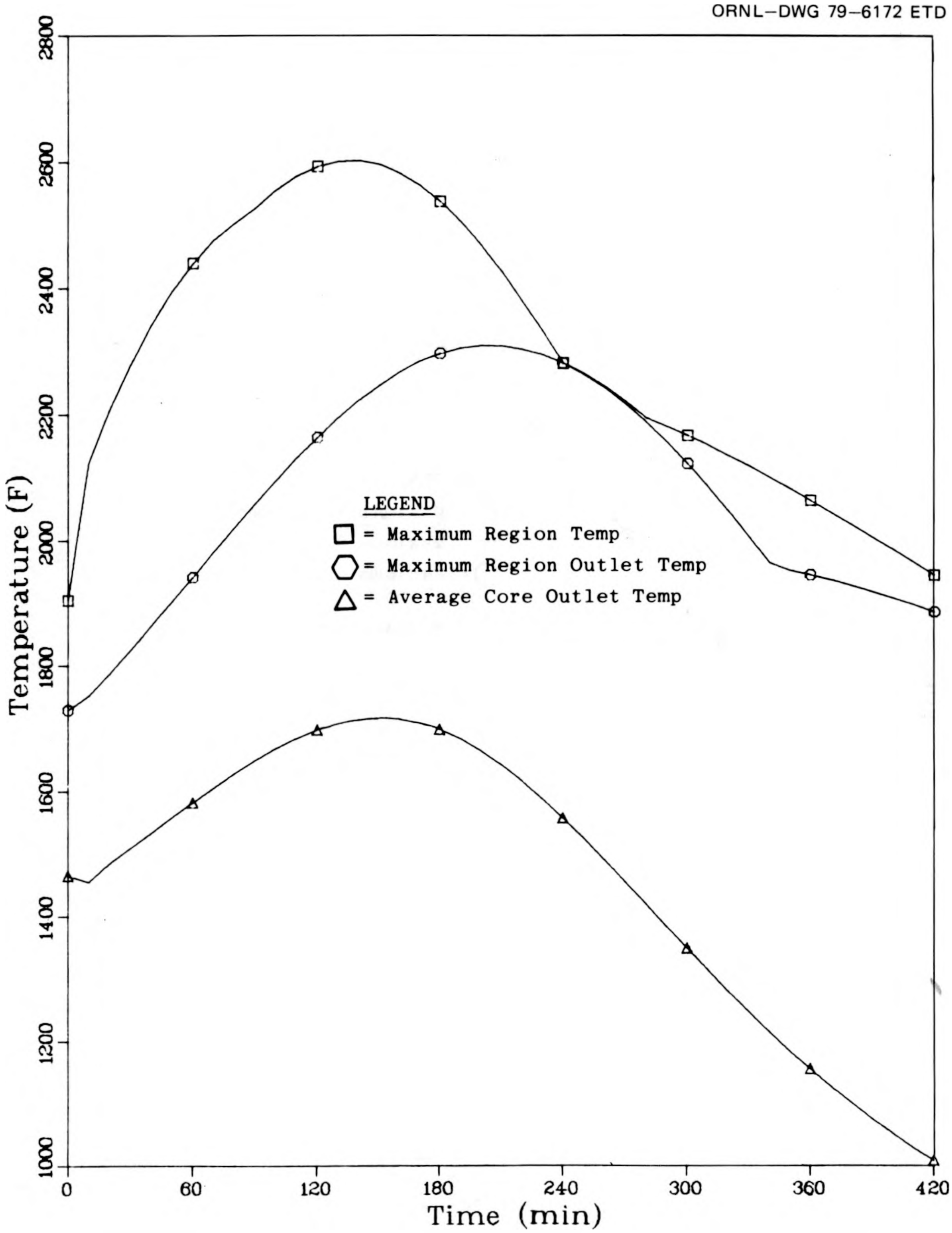

Fig. 3. ORECA core thermal response for a 6.25-psi initial pressure drop, DBDA.

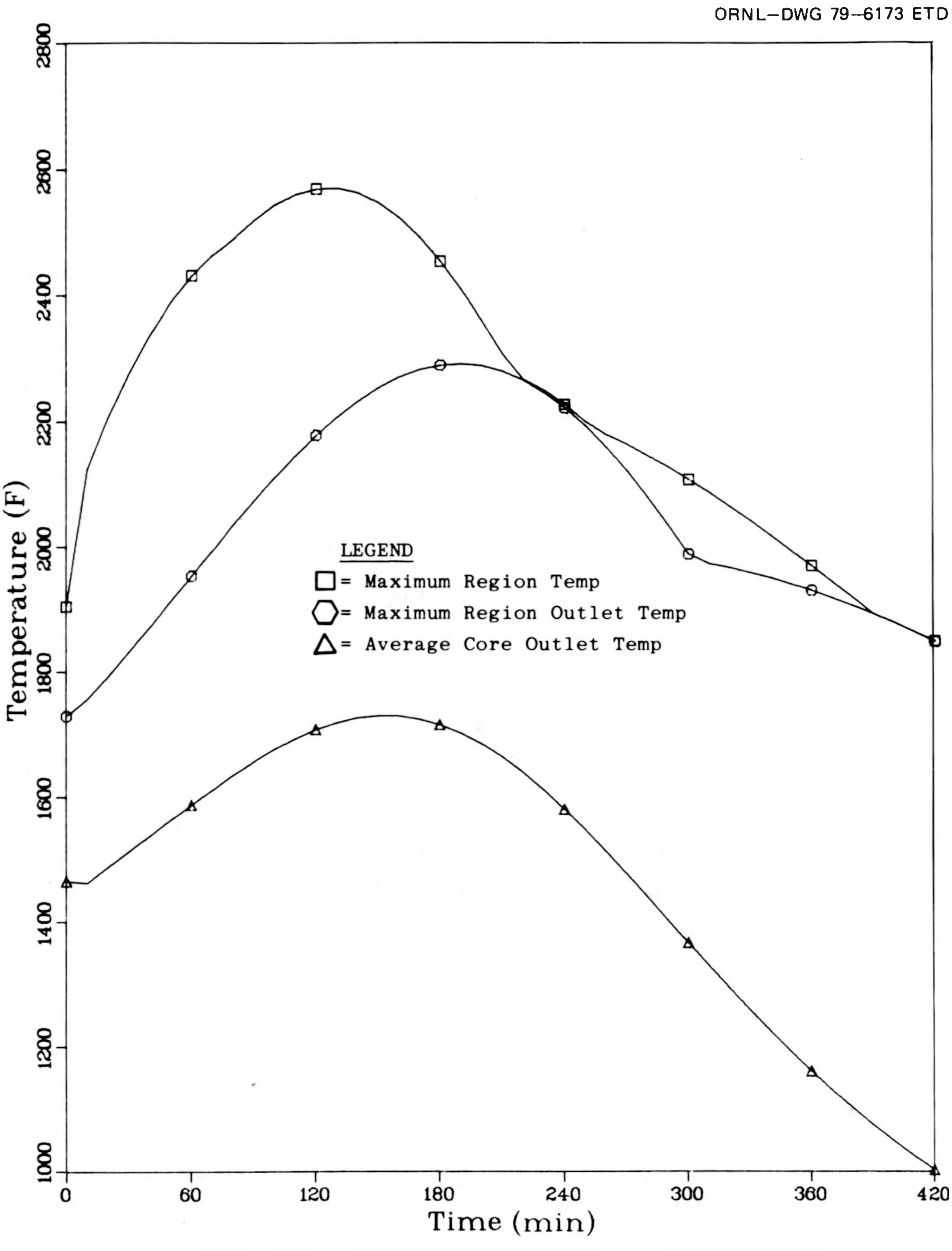

Fig. 4. ORECA core thermal response for a 10.0-psi initial pressure drop, DBDA.

240 min, the maximum subregion lower reflector block temperature is greater than the fuel temperature. The reason for the sawtooth appearance of the maximum average core temperature is that the maximum average core temperature appears in different refueling regions as the transient progresses.

Another important parameter is the maximum coolant outlet temperature. The concern here is for damage to the steam generator inlet liners caused by hot streak impingement. The maximum coolant outlet temperature is 2318°F and occurs at 260 min into the transient, while the average reactor coolant outlet temperature is 1745°F and occurs at 185 min. The resulting hot streaks, computed via a GA experimentally derived equation, would not damage the liners.

Figure 3 presents the ORECA calculated core and coolant temperatures for the 6.25-psi ΔP case. The maximum average fuel temperature is 2600°F and occurs at 130 min into the transient. ORECA is less detailed than FLODIS and does not split up the refueling regions into rectangular subregions. The maximum average temperature calculated by FLODIS, 2725°F, is about the same; however, it occurs at 182.5 min. This difference in time is probably due to the different modeling techniques of the two codes.

The maximum ORECA region outlet temperature is 2300°F and occurs at 210 min. This value agrees very well with the previous FLODIS calculation; however, the FLODIS calculated temperature occurs $\sim$50 min later, as did the maximum core region temperature. The ORECA maximum average region coolant outlet temperature is 1700°F, agreeing well with FLODIS, and it occurs at 150 min, 35 min earlier than FLODIS.

In the DBDA calculations using 10-psi initial core ΔP, the FLODIS calculations of maximum fuel and maximum fuel average temperatures (Fig. 2) occur $\sim$15 min earlier and are lower in value than those of the 6.25 ΔP case (Fig. 1). The maximum coolant outlet temperature peaks $\sim$15 min earlier with a slightly higher temperature (2325°F) than for the 6.25 ΔP case. The average core outlet temperature, however, peaks 10 min later than the low ΔP case with an also slightly higher temperature of 1770°F. ORECA calculations (Figs. 3 and 4) agree very well with those of FLODIS; however, the maximum temperatures again peak earlier. Table 1 summarizes these values for both cases and both codes.

Table 1. Design-basis depressurization accident (DBDA) data

Parameters	FLODIS initial core ΔP (psi)				ORECA initial core ΔP (psi)			
	6.25		10.0		6.25		10.0	
	Maximum temperature (°F)	Time (min)	Maximum temperature (°F)	Time (min)	Maximum temperature (°F)	Time (min)	Maximum temperature (°F)	Time (min)
Fuel temperature	2725	182.5	2690	170				
Region fuel temperature	2635	172.5	2610	160	2600	140	2550	130
Maximum region coolant outlet temperature	2318	260	2325	245	2300	210	2300	190
Average core outlet temperature	1745	185	1770	190	1725	150	1725	150

In summary, the results of both codes show that changes in the initial core flow resistance have only a minor effect on the temperature and flow distribution for the DBDA, where the static head is insignificant. However, both codes do show that higher orifice loss coefficients result in somewhat lower peak temperatures.

1.5.2 LOFC analyses

The results of FLODIS and ORECA calculations for the first 4 hr of the 90-min LOFC/FWCD accident are presented in Figs. 5 through 16. The basic core performance parameters are as follows:

1. EQSB3 peaking factors, region outlet temperatures, helium inlet temperature, and helium flow from Ref. 8;
2. helium pressure as a function of time from Ref. 8 (the primary coolant system is not depressurized);
3. graphite thermal conductivity of 10.0 Btu/hr-ft°F from the FSAR;[9]
4. core afterheat as a function of time from LTR-4.[11]

As shown in Fig. 5, in FLODIS calculations for the 6.25-psi core ΔP case, the maximum fuel temperature for a subregion is 3050°F and occurs at 170 min into the transient (80 min after forced convection is restored). The maximum average fuel temperature for an entire refueling region is 2928°F and occurs at 145 min into the transient. Intraregional flow distribution, as in the DBDA, causes the subregion maximum fuel temperature to peak 25 min later than the refueling region average maximum fuel temperature. The positive feedback temperature effect on helium viscosity is also responsible for the maximum refueling region average fuel temperature peaking 55 min after forced convection is restored. Figure 6 clearly demonstrates this viscosity effect on the interregional flow distribution. The static head of a relatively hot region is less than that of a relatively cold region, which also tends to decrease flow in the hot region.

Regions 2, 19, and 24 have power peaking factors greater than unity, (1.25, 1.83, and 1.42 respectively) and are therfore "hot" regions. Region 5 has a power peaking factor less than unity (0.65) and is a "cold" region. During the first 90 min of the transient (no forced convection), the hot regions have upflow(-), with the amount of upflow being higher

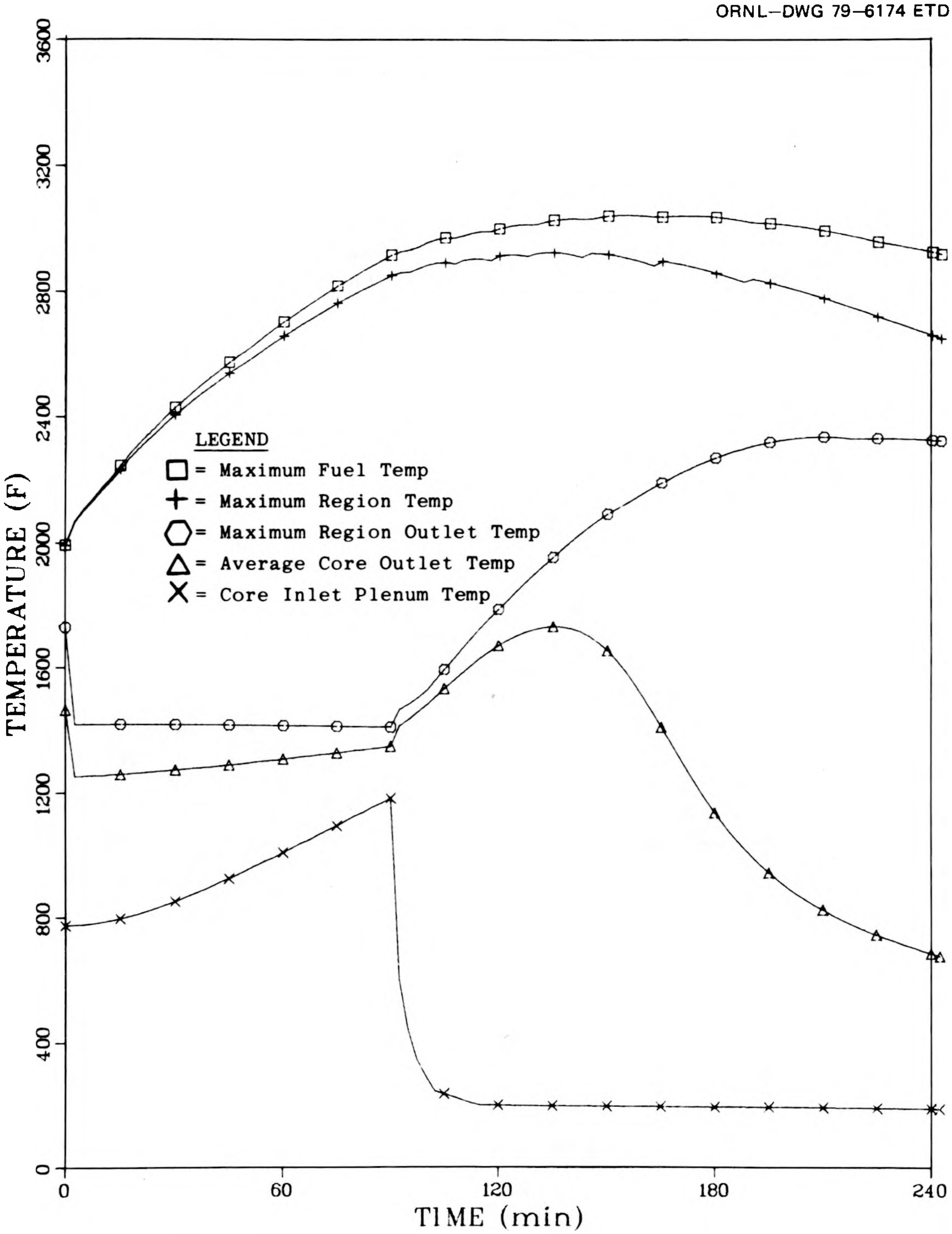

Fig. 5. FLODIS core thermal response for a 6.25-psi initial pressure drop, LOFC/FWCD.

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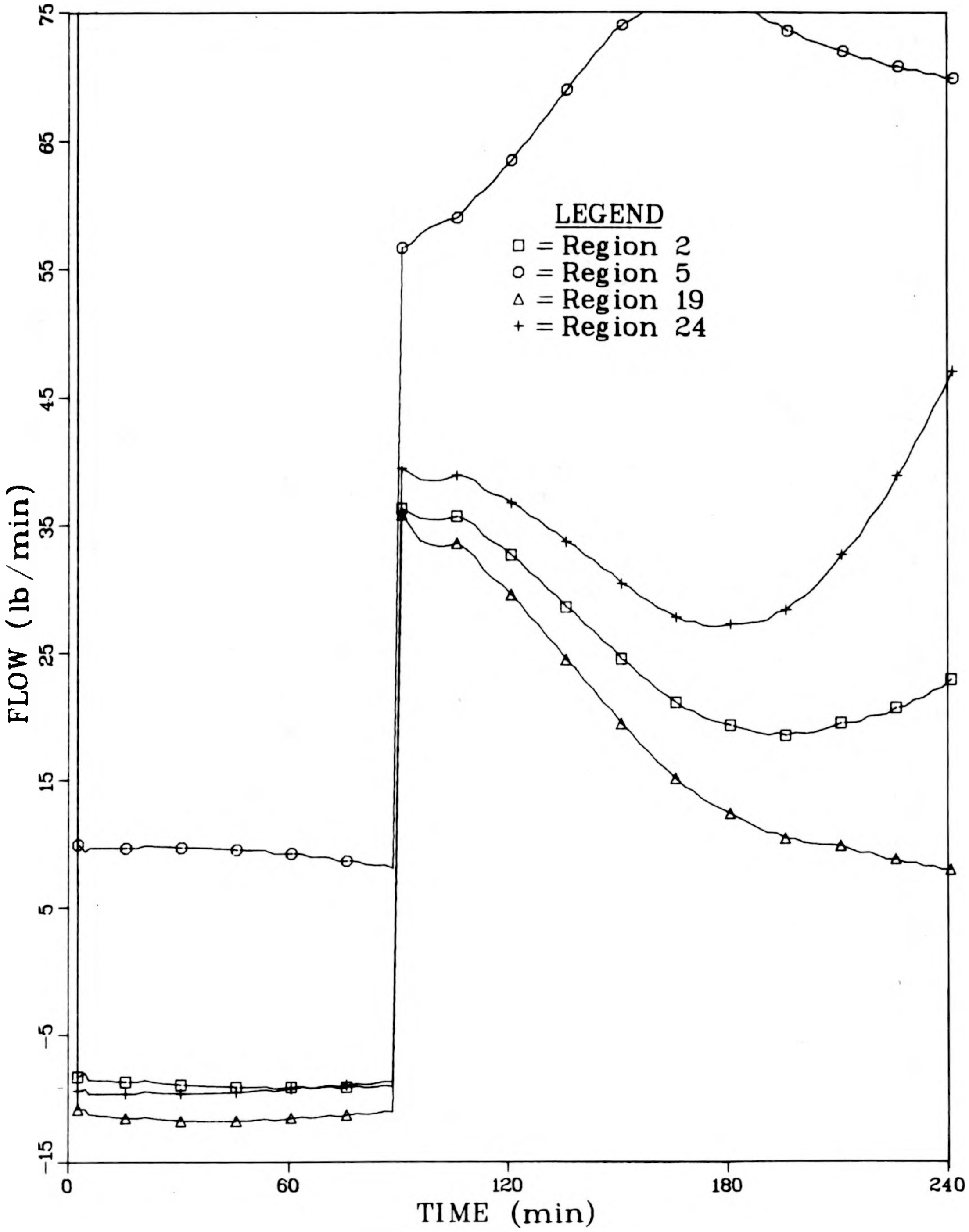

Fig. 6. FLODIS calculated region flows for a 6.25-psi initial pressure drop, LOFC/FWCD.

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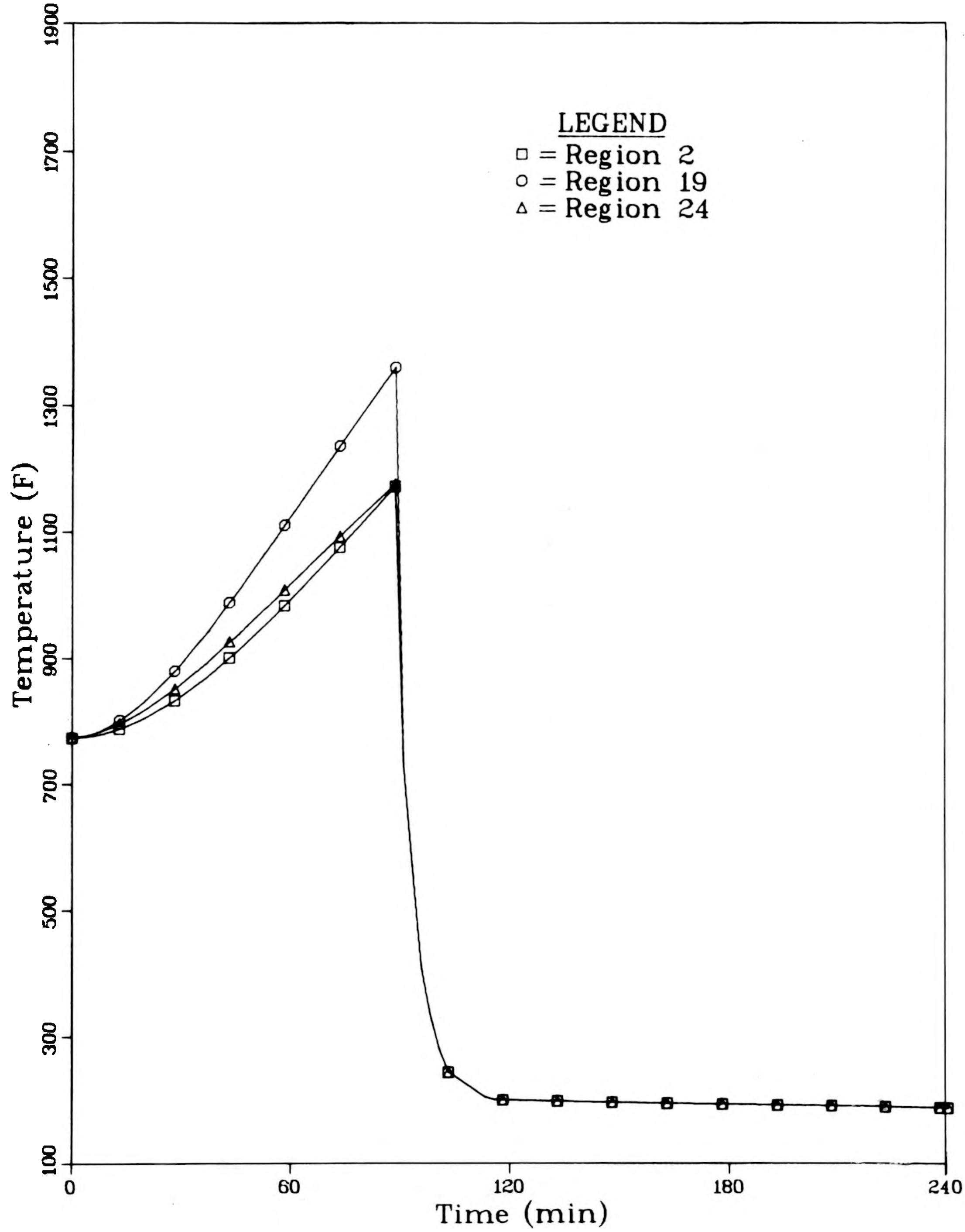

Fig. 7. FLODIS calculated reverse flow temperatures for a 6.25-psi initial pressure drop, LOFC/FWCD.

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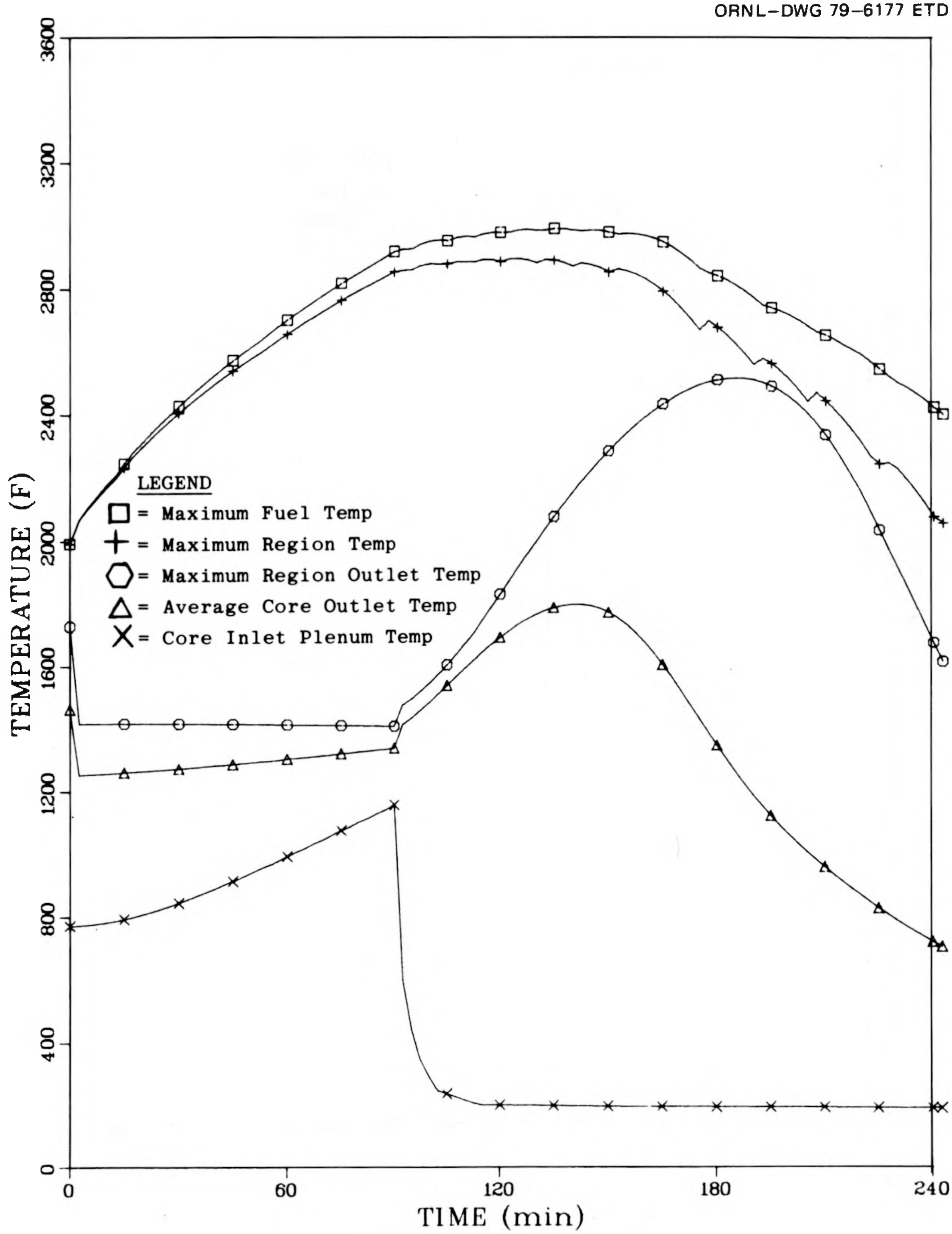

Fig. 8. FLODIS core thermal response for a 10.0-psi initial pressure drop, LOFC/FWCD.

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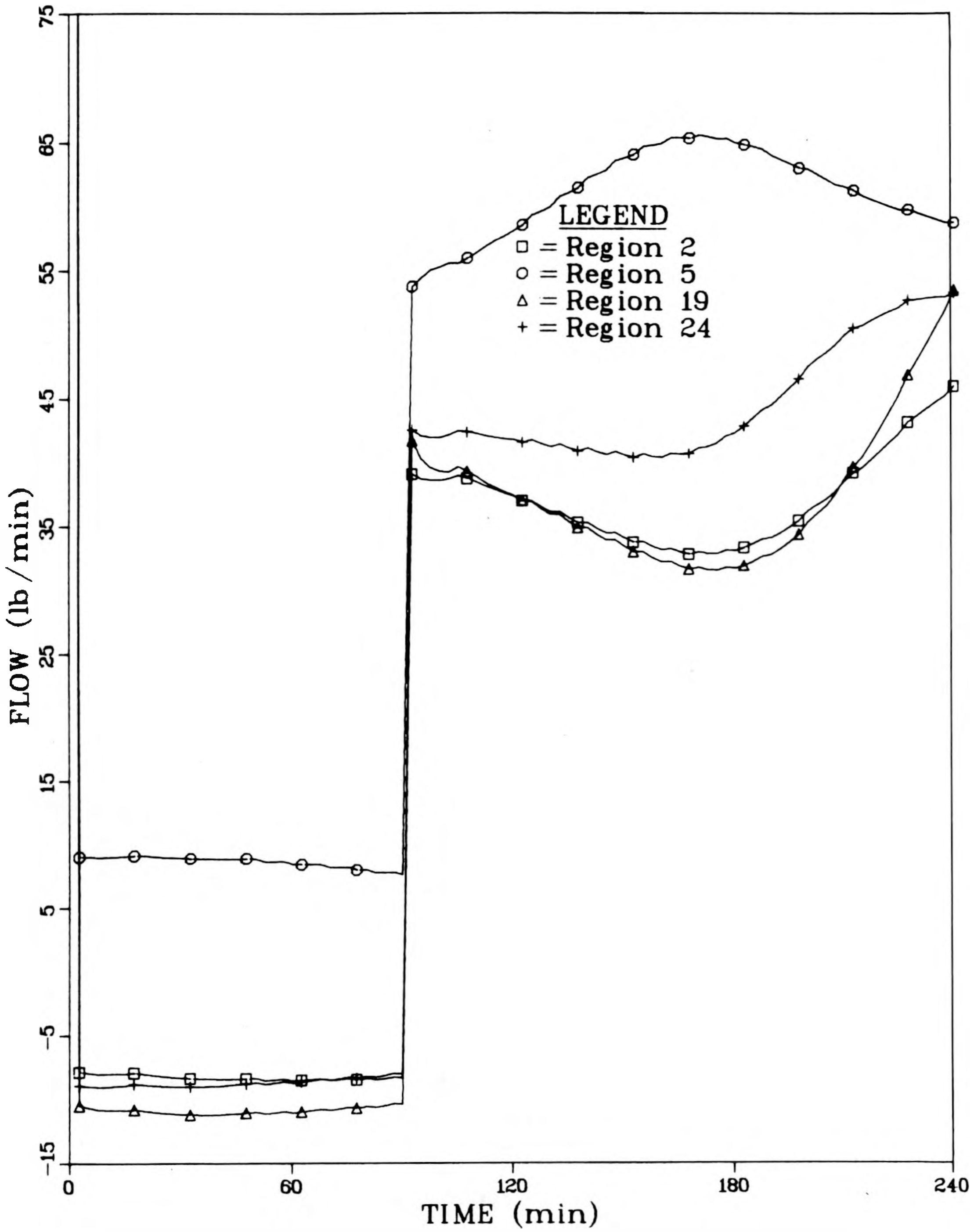

Fig. 9. FLODIS calculated region flows for a 10.0-psi initial pressure drop, LOFC/FWCD.

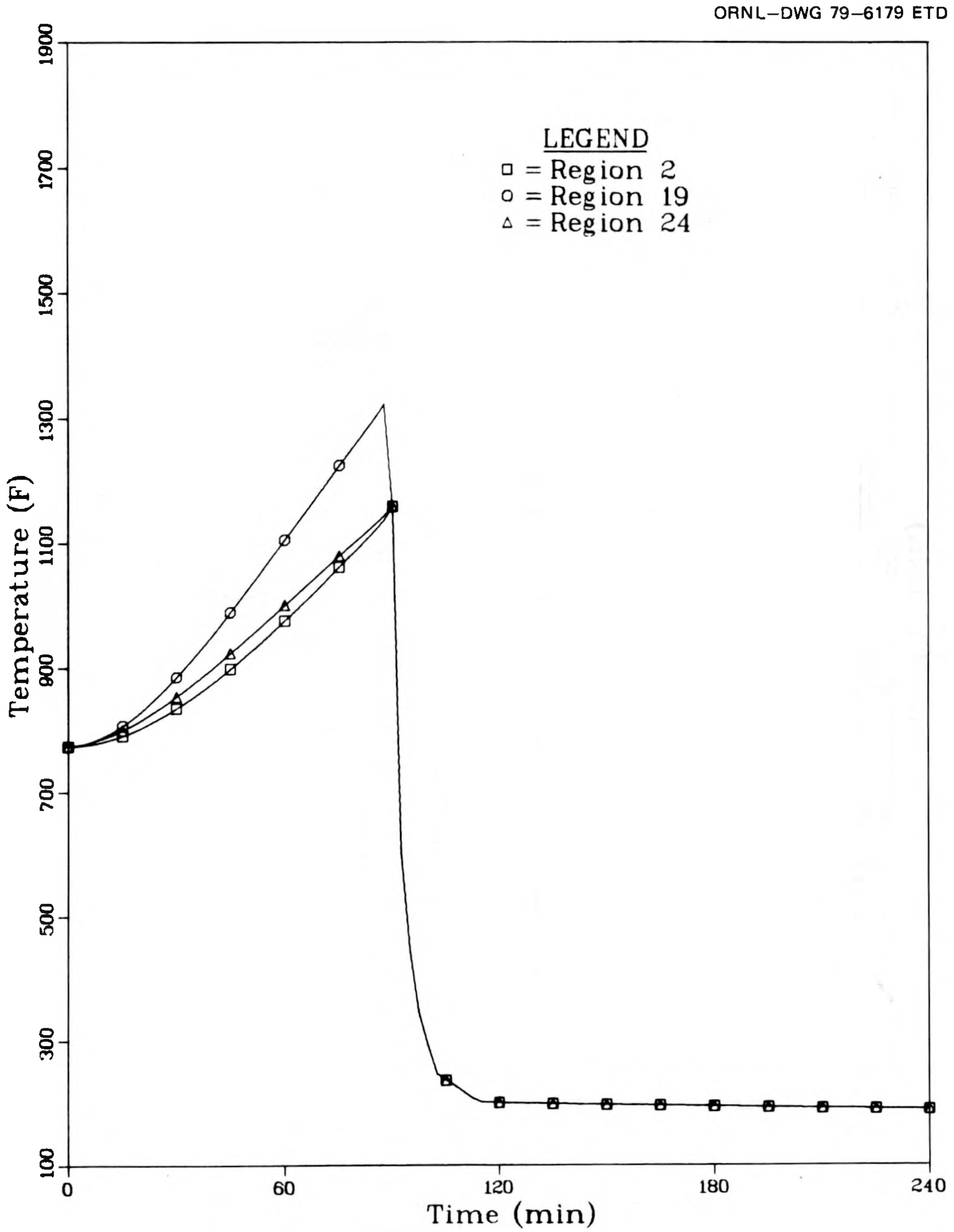

Fig. 10. FLODIS calculated reverse flow temperatures for a 10.0-psi initial pressure drop, LOFC/FWCD.

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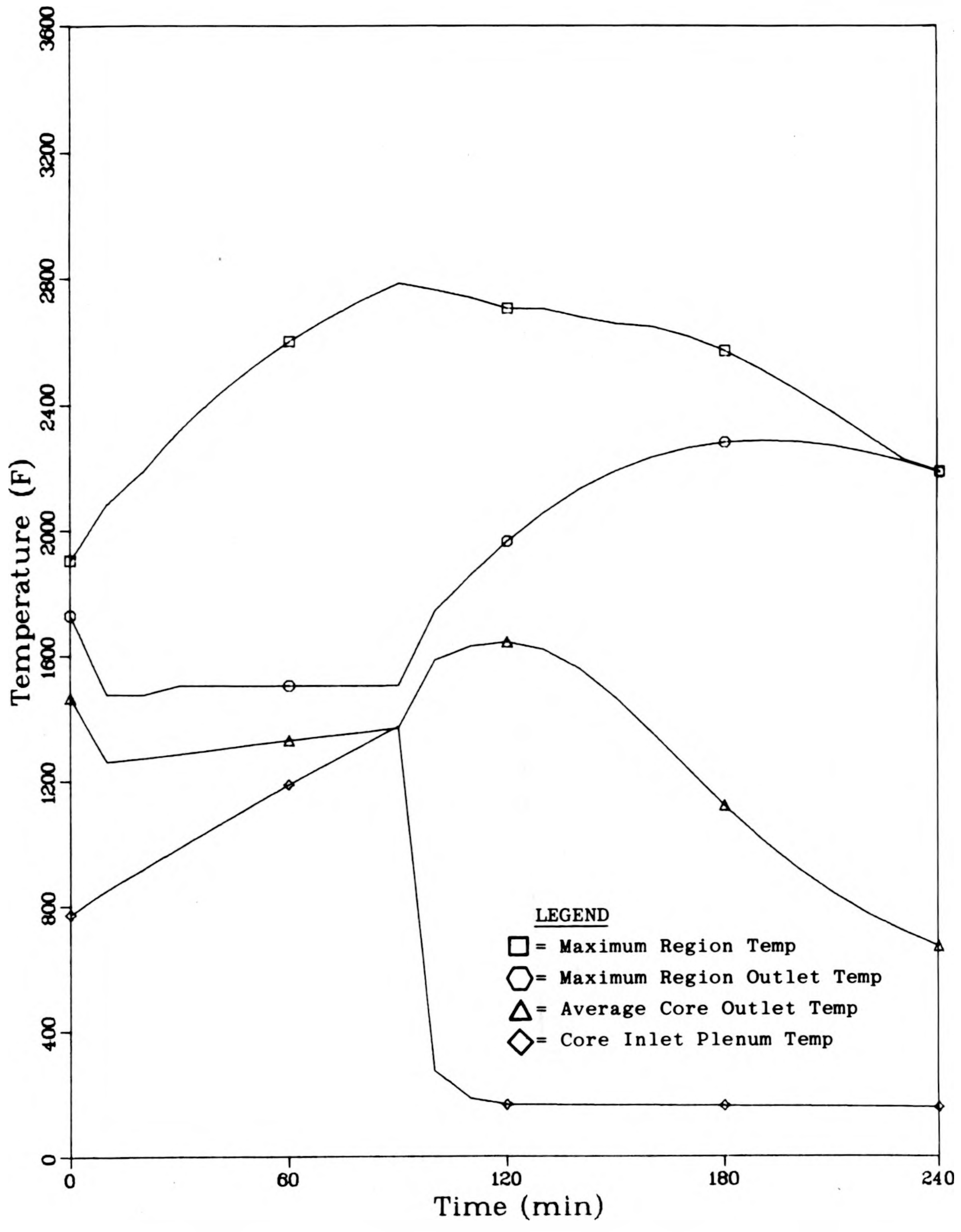

Fig. 11. ORECA core thermal response for a 6.25-psi initial pressure drop, LOFC/FWCD.

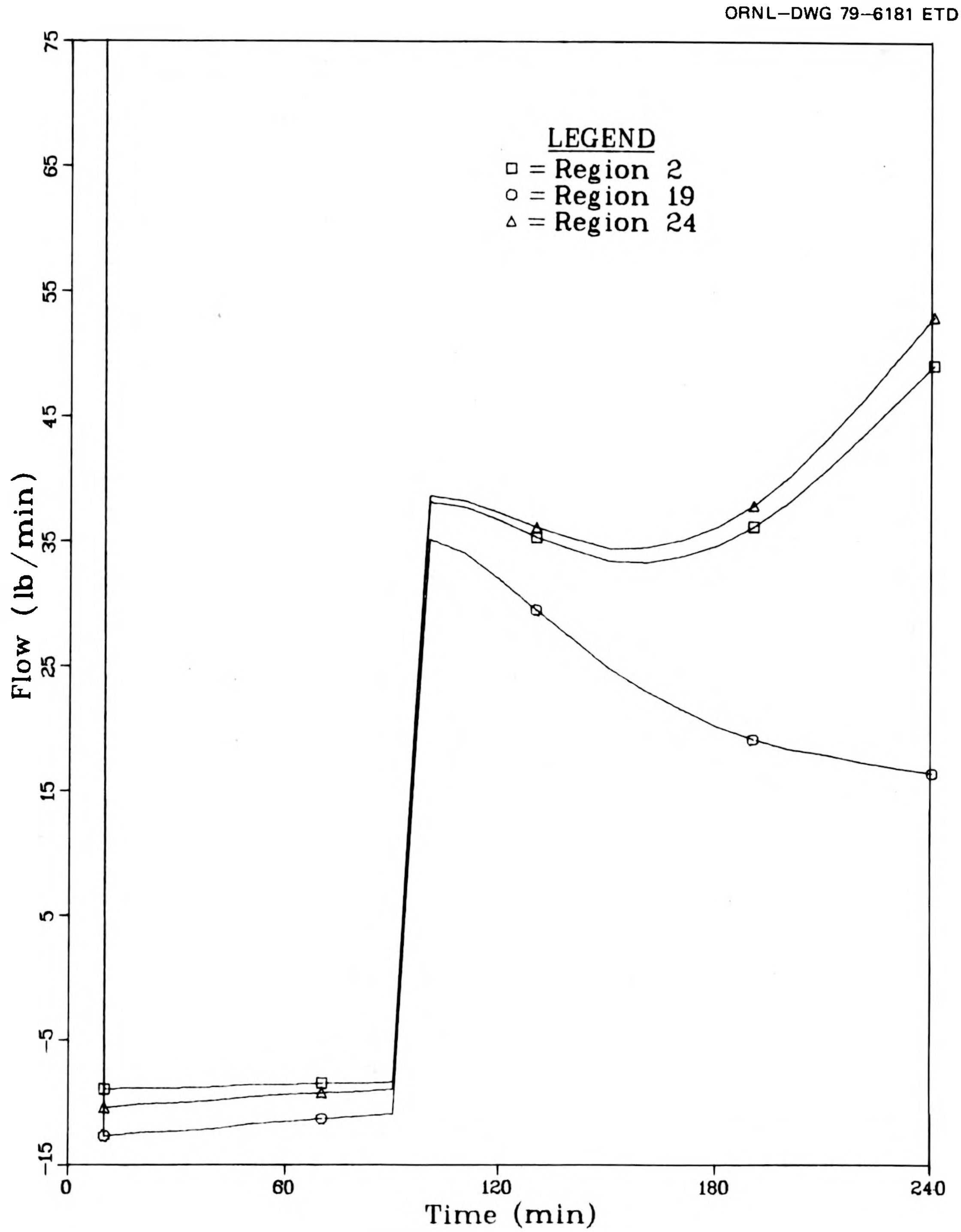

Fig. 12. ORECA calculated region flows for a 6.25-psi initial pressure drop, LOFC/FWCD.

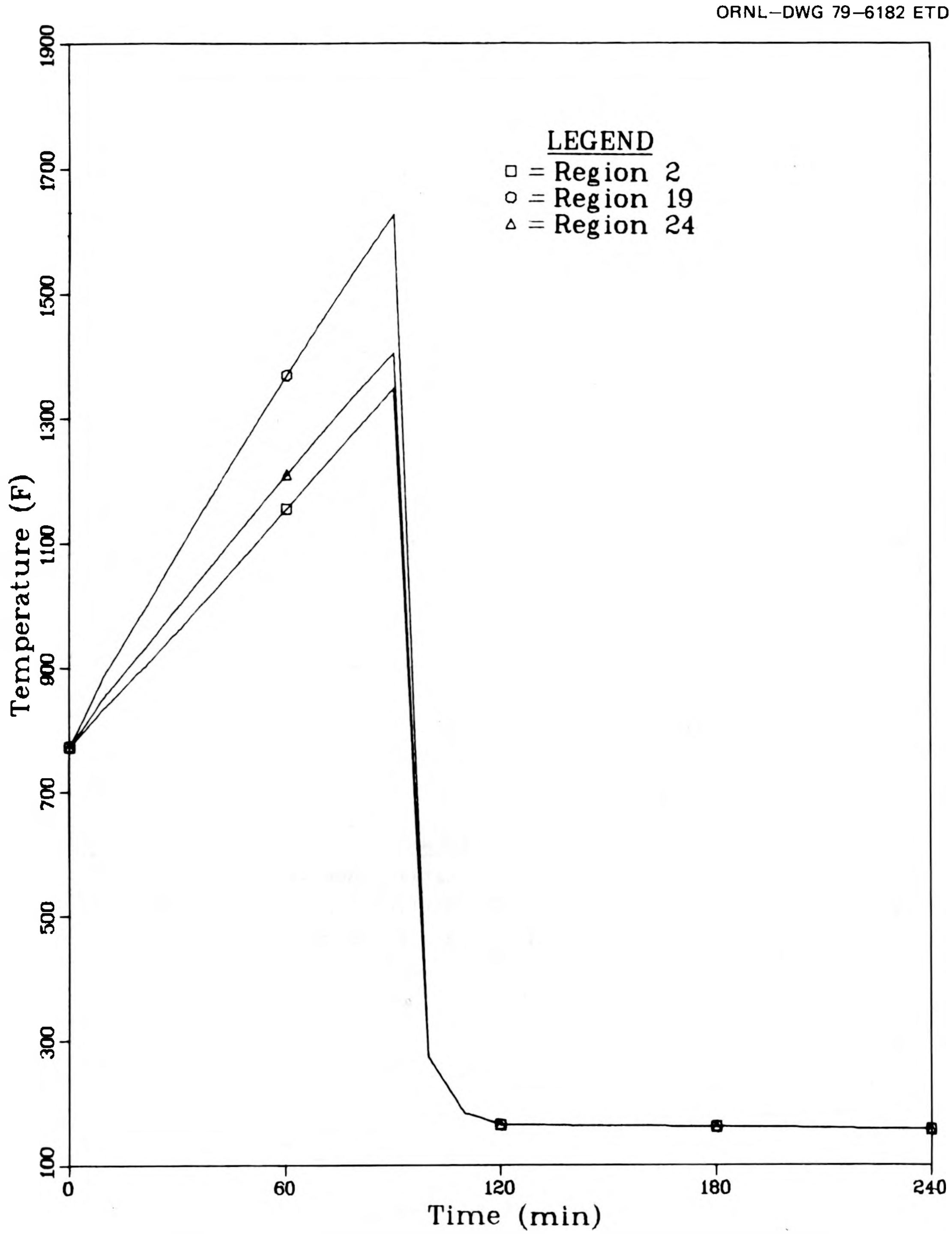

Fig. 13. ORECA calculated reverse flow temperatures for 6.25-psi initial pressure drop, LOFC/FWCD.

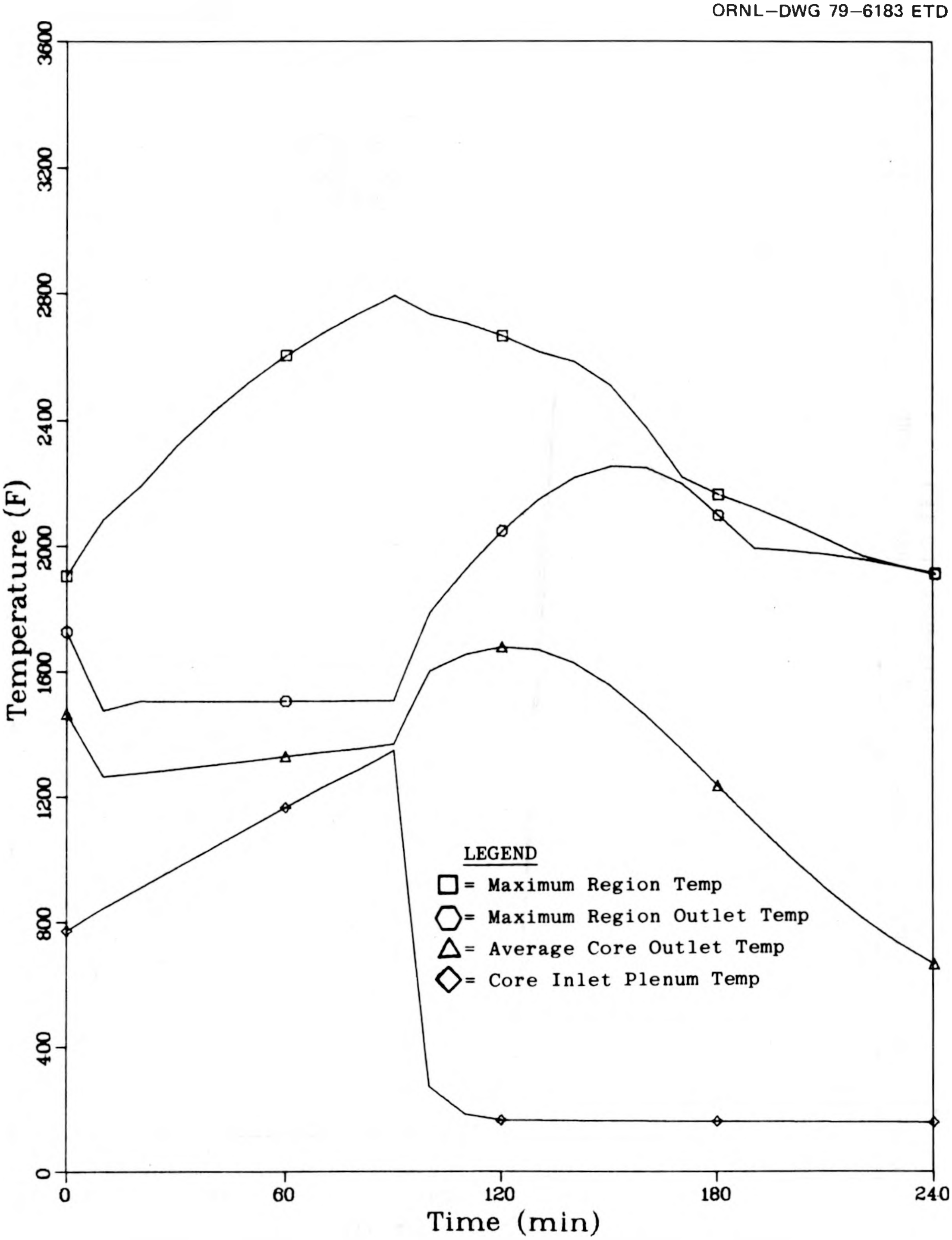

Fig. 14. ORECA core thermal response for a 10.0-psi initial pressure drop, LOFC/FWCD.

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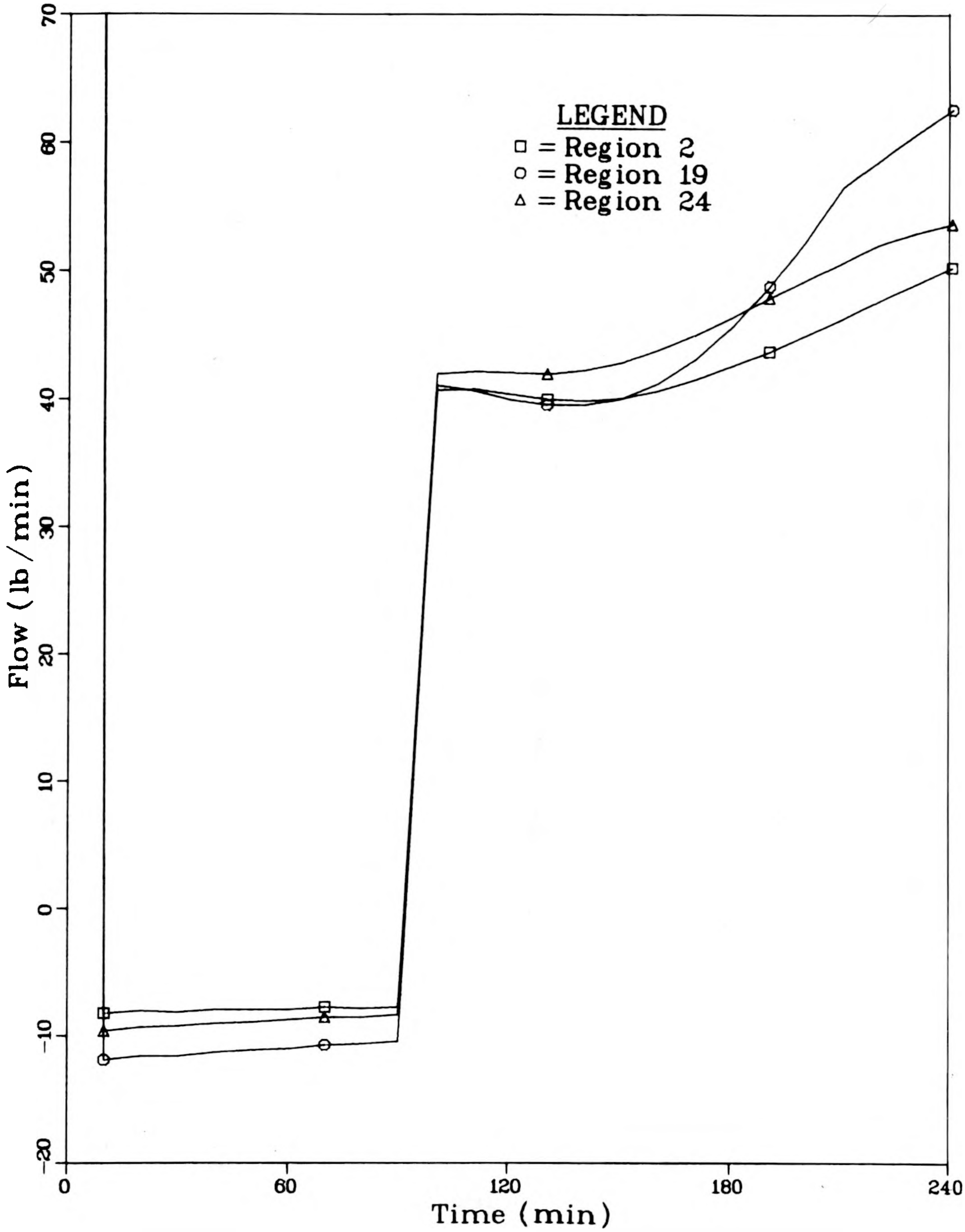

Fig. 15. ORECA calculated region flows for a 10.0-psi initial pressure drop, LOFC/FWCD.

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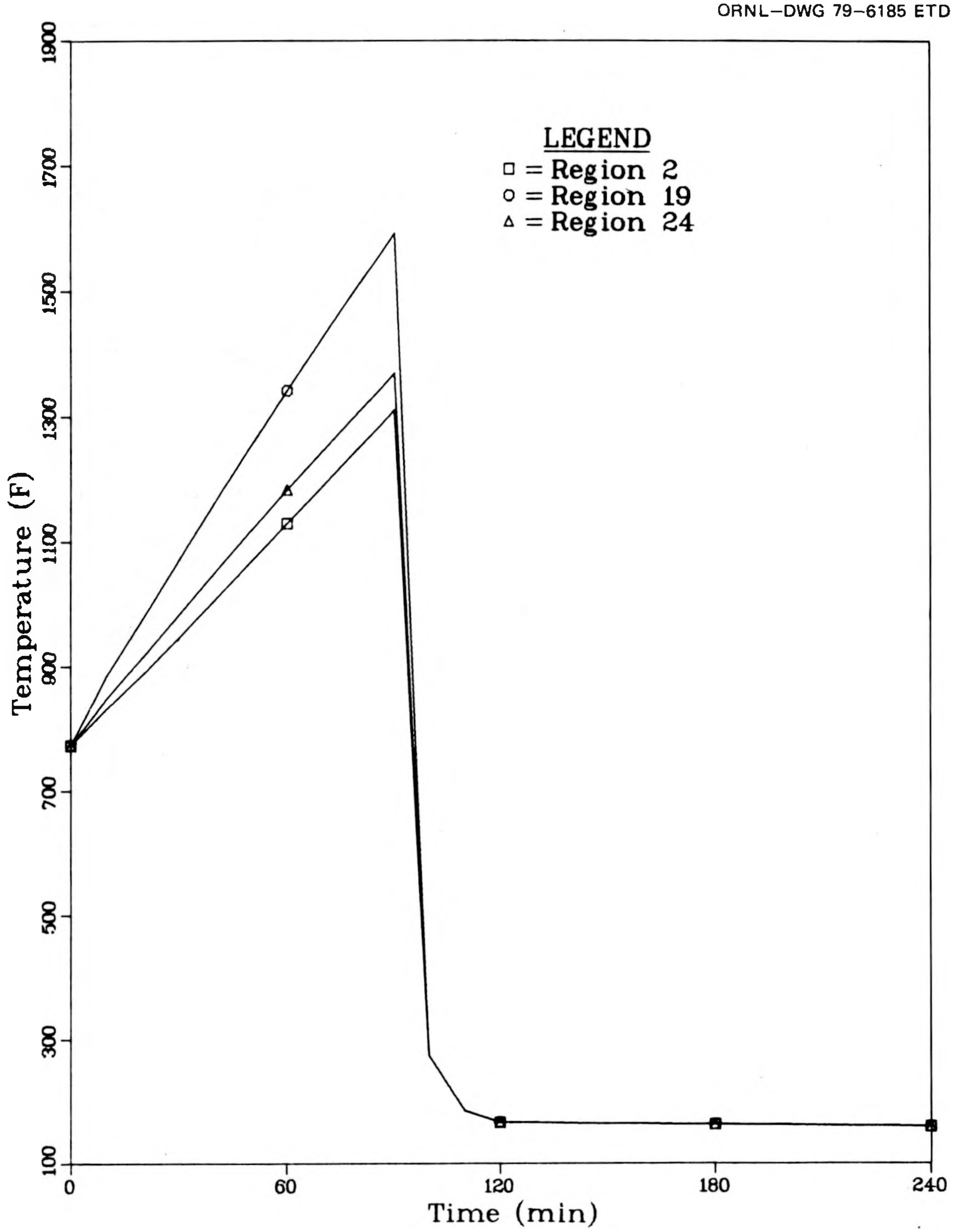

Fig. 16. ORECA calculated reverse flow temperatures for a 10.0-psi initial pressure drop, LOFC/FWCD.

with higher-power peaking factors. The cold region has downflow(+). Immediately upon reestablishment of forced convection, the downflows for all four regions are within 15 lb_m/min and are lower for the lower-power peaking factor regions. From 90 to 180 min, the hot regions get progressively less flow and the cold region gets progressively more, as the previously explained temperature viscosity effect on the friction pressure drop dominates the plenum-to-plenum pressure drop.

At 180 min, the flow in region 5 starts to decrease with time, and the flow in region 24 starts to increase with time. As the core cools, the temperature viscosity effect on the friction pressure drop becomes less dominant and the flow control orifice pressure drop becomes a more dominant part of the plenum-to-plenum pressure drop. Region 24 is located adjacent to three regions with power factors less than unity (0.95, 0.49, and 0.74), to one region with power factor slightly greater than unity (1.02), and to the cooler side reflector. Thermal conduction from region 24 to the cold surrounding regions is apparently an important heat transfer mode.

The flow in region 2 does not stop decreasing until ∿200 min into the transient. Region 2 is located adjacent to three regions with power factors substantially greater than unity (1.29, 1.51, and 1.83) and three regions with power factors slightly less than unity (0.89, 0.86, and 0.92). Region 2 has a power factor of 1.25, which is less than region 24 (1.42), but it requires more time to cool, because of the important effect of interregional thermal conduction. Regions 2 and 24 start out the transient with approximately the same fuel temperature. When forced convection is restored at 90 min, region 2 is ∿65°F hotter than region 24, and at 240 min region 2 is ∿435°F hotter than region 24.

The orifice loss coefficient for region 2 (24.1) is greater than that for region 24 (17.5); however, the pressure drop across the orifice for both regions is approximately equal for both regions for most of the transient and is a small fraction (15 to 5%) of the friction pressure drop for the entire transient. At 240 min, the orifice pressure drop for region 24 is triple the orifice pressure drop for region 2, resulting from the higher mass flow through region 24 at that time. Therefore, the pressure drop due to the orifice is less important than the pressure drop due to the viscosity effect of temperature on the friction factor on the plenum-to-plenum pressure drop at the high temperatures for this case.

The mass flow for region 19 is continuing to decrease at 240 min. The computations were not continued beyond 240 min because of the large cost for FLODIS runs (∿$100).

A very important temperature during the period of no net reactor coolant flow is the temperature of the helium emerging from the flow control orifice of the high-power regions having reverse flow. The concern is that these high-temperature plumes will have an adverse impact on the PCRV carbon steel liner in the inlet plenum. The temperature of the helium flowing through the flow control orifice for three high-power regions is plotted in Fig. 7. This temperature, of course, drops rapidly to the inlet plenum helium temperature after forced convection is restored at 90 min.

As shown in Fig. 8, in FLODIS calculations for the 10-psi core ΔP case, the maximum fuel temperature for a subregion is 2995°F and occurs at 145 min into the transient (55 min after forced convection is restored). The maximum average fuel temperature for the entire refueling region is 2905°F and occurs at 125 min into the transient.

Evidence of the importance of the orifice pressure drop is seen on Fig. 9, as compared to Fig. 6 for the 6.25 initial core pressure drop case explained earlier. During the first 90 min, when there is no net reactor coolant flow, the corresponding region flows are approximately equal for both cases. However, the region flows after forced convection is restored are considerably different. The cold region has less flow at restoration, as compared to the 6.25-psi case, and the hot regions have more.

The flow of region 19, the hottest one plotted, also starts to decrease upon resumption of forced convection, but at a lower rate than the 6.25-psi case. At 180 min into the transient, the flow of region 19 starts to increase dramatically. The maximum coolant outlet temperature plotted on Fig. 8 has its maximum at 180 min also.

The flow of region 5, the coldest one plotted, increases upon resumption of forced convection, but at a lesser rate of increase than the 6.25-psi case. At 180 min, the flow starts to decrease as the orifice pressure drop becomes predominant over the friction pressure drop of the plenum-to-plenum pressure drop. After 180 min, the rates of decrease for the two cases are essentially identical for region 5, but the absolute values are different, of course.

The flow in region 2 also decreases upon resumption of forced convection, but at a lesser rate than in the 6.25-psi case. At 180 min, the flow starts to increase at a greater rate than in the 6.25-psi case. The flow of region 24 is also similar, except that, after 230 min, the flow is no longer increasing. This would then mean that region 24 has cooled off.

Figure 10 plots the reverse flow temperatures from regions 2, 19, and 24. These temperatures are slightly lower than those for the 6.25-psi case plotted in Fig. 7.

For the ORECA LOFC calculations, the core performance parameters of maximum region fuel temperature, maximum region coolant outlet temperature, average core outlet temperature, and inlet plenum temperature are plotted in Fig. 11 for the 6.25-psi initial core pressure drop case and in Fig. 14 for the 10.0-psi initial core pressure drop case. The maximum and the times at which they occur for each parameter are tabulated in Table 2, along with those for FLODIS. As can be seen from this table, the values from FLODIS are slightly higher than those of ORECA (but still in close agreement), and they occur later in the transient.

The region flows for three hot regions are plotted in Fig. 12 for the 6.25-psi case and in Fig. 15 for the 10.0-psi case. Comparisons of these plots with those of FLODIS (Fig. 12 with Fig. 6 and Fig. 15 with Fig. 9) show that, while the absolute values of the region flow may disagree, particularly at 180 min, the trend of the flow behavior is similar for both codes for both cases. The FLODIS minimum flows for the hot regions occur later than the ORECA calculated values.

The reverse flow temperatures for three hot regions are plotted in Fig. 13 for the 6.25-psi case and in Fig. 16 for the 10.0-psi case. The maximum reverse flow temperature of 1625°F for the 6.25-psi case is only slightly higher (25°F) than that for the 10.0-psi case. However, the ORECA calculated values are significantly higher ($\sim$200°F) than those calculated by FLODIS (compare Fig. 7 with Fig. 13 and Fig. 10 with Fig. 16.) The reason for these discrepancies is not known at present.

Table 2. Loss-of-forced-convection (LOFC) accident data

Parameters	FLODIS initial core ΔP (psi)				ORECA initial core ΔP (psi)			
	6.25		10.0		6.25		10.0	
	Maximum temperature (°F)	Time (min)	Maximum temperature (°F)	Time (min)	Maximum temperature (°F)	Time (min)	Maximum temperature (°F)	Time (min)
Fuel temperature	3050	170	2995	145				
Region fuel temperature	2928	145	2905	125	2800	90	2800	90
Maximum region coolant outlet temperature	2345	210	2525	185	2300	190	2250	150
Average core outlet temperature	1735	135	1803	140	1650	120	1650	120
Inlet plenum temperature	1184	90	1162	90	1350	90	1300	90

In conclusion, both codes indicated that the initial core flow resistance, as determined by the positions of the flow control orifices (loss coefficient), can significantly affect the flow and temperature distributions subsequent to an accident. It was shown for the DBDA that these orifice loss coefficients had only a slight influence on the post-accident flow and temperature distribution, while they had a profound effect on the flow and temperature distributions for the LOFC subsequent to restoration of forced convection. For the LOFC, the positive feedback effect of helium viscosity coupled with the static head of high-pressure helium indicates the high sensitivity of the reactor flow distribution to the orifice loss coefficients. The differences between the two codes, due to the finer detail of FLODIS (80 calculation nodes for a full seven-column region), as compared to ORECA (8 nodes), also support this conclusion. The apparently important assumption that the orifice loss coefficient is not a function of flow (especially in the laminar flow regime) might need reevaluation in the light of this high sensitivity.

2. MEETING ATTENDED UNDER PROGRAM SPONSORSHIP: RSR HTGR SAFETY PROGRAM MIDYEAR REVIEW, BETHESDA, MARYLAND, FEBRUARY 14, 1979

S. J. Ball

The meeting was held in Bethesda (rather than Silver Spring) to facilitate attendance by NRR personnel, because at least partial sponsorship of the program by NRR was under discussion. The agenda included presentations and discussions on FSV accident code development and verification, program assistance to NRC on FSV licensing questions, FSV oscillation problem investigations, and foreign information exchange.

Discussions of future work resulted in the following outline:

1. FSV licensing questions — on call assistance
2. Code verification
    a. FSV scram tests (ORECA)
    b. FSV steam generator transients (BLAST) (especially as they relate to allowable temperature transients)
    c. Upper plenum reverse-flow plume heat transfer characteristics
    d. Lower plenum hot-streak mixing factor determination
    e. Region outlet thermocouple dynamic model
3. FSV oscillation problem review
4. Assessment of postulated DBA assumptions
5. Code update and documentation

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