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WRAP-PWR Verification Studies

M. V. Gregory, P. L. Ames, F. Beranek, N. H. Kuehn, P. B. Parks

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ABSTRACT

A modular computational system known as the Water Reactor Analysis Package - Evaluation Model (WRAP-EM) was developed for the Nuclear Regulatory Commission (NRC) to interpret and evaluate reactor vendor EM methods and computed results. A subset of the system (WRAP-PWR-EM) provides the computational tools to perform a complete analysis of loss-of-coolant accidents (LOCA's) in pressurized water reactors (PWR's). A set of calculations modeling experimental tests in the Semiscale and LOFT facilities, and calculations of a large break in a typical four-loop Westinghouse PWR plant have verified that the WRAP-PWR-EM system is functioning as intended.

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M. M. Anderson	R. L. Reed
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1. INTRODUCTION

The Water Reactor Analysis Package - Evaluation Model (WRAP-EM) is a modular system of codes which provides the computational tools required to perform complete licensing type analyses of postulated loss-of-coolant accidents (LOCA's) in light water nuclear power reactors. The system was developed at the Savannah River Laboratory (SRL) for use primarily by the Nuclear Regulatory Commission (NRC) for interpreting and auditing reactor vendor evaluation model (EM)[1] methods and computed results. The system for pressurized water reactors, WRAP-PWR-EM, is described in Reference 2. A similar document describing the system for boiling water reactors, WRAP-BWR-EM, is described in Reference 3. The final step in the PWR development program, verification of WRAP-PWR-EM, is documented in this report.

NRC specified a series of analyses to be run to verify that the WRAP-PWR-EM system was functioning properly and was capable of correctly modeling physical phenomena in a number of different PWR systems. Input for these analyses was derived from RELAP4/MOD5 input data decks prepared by the Idaho National Engineering Laboratory (INEL). The transients analyzed were two experiments: Semiscale Test S-06-3 and LOFT Test L2-3, and a double-ended, large pipe break in a typical four-loop Westinghouse PWR plant. In general, a single analysis was run for each transient; thus, sensitivity to input parameters was not determined.

2. SUMMARY

WRAP PWR-EM successfully calculated transients for two experiments, Semiscale S-06-3 and LOFT L2-3, and a four-loop PWR. The calculated behavior of all the systems was physically reasonable and the results of the calculations were selfconsistent. Thus, the WRAP-PWR-EM system was judged to be an acceptable tool for interpreting and evaluating reactor vendor licensing calculations. However, additional work is recommended on the reflood models which currently are overly conservative.

The WRAP-PWR-EM analysis of the Semiscale S-06-3 experiment predicted a peak clad temperature (PCT) of 1093°K at 43 sec, compared to the measured value of 1155°K at 20 sec. However, this nonconservative result was not necessarily an indictment of WRAP-PWR-EM because all Appendix K rules required for an EM calculation were not applied. Only a single run with a discharge coefficient (C_D) of 1.0 was performed; whereas, Appendix K requires a series of computations, varying C_D to determine the maximum PCT. Also, because the experiment was electrically heated, the conservative EM nuclear fuel models could not be invoked.

For the LOFT L2-3 experiment, the WRAP-PWR-EM analysis predicted an adequately conservative response: the computed maximum clad temperatures remained approximately 100°K higher than the experimental value throughout the transient. The experiment resulted in a PCT of 914°K at five sec while WRAP-EM computed a local maximum of 970°K at three sec and went on to compute a PCT of 1030°K at 43.5 sec (during reflood). The conservatism was a result of both the evaluation model disallowing a return to nucleate boiling in the rewetting phase following the attainment of critical heat flux (CHF) and the adiabatic heatup assumed in the refill phase.

The analysis of the PWR plant system represented a more formal verification of the entire WRAP-PWR-EM system. Parallel calculations were performed at INEL and SRL. WRAP-PWR-EM computed essentially the same results as the RELAP4/WRAP analogue at INEL. This demonstrated that the EM coding developed at SRL and INEL and subsequently added to WRAP-PWR-EM was functioning as intended. Unlike the other two verification calculations, this analysis was carried through the hot pin analysis by using the FRAP module and thus represented a test of the entire WRAP-PWR-EM system. The FRAP calculation gave very high clad surface temperatures (greater than 2300°F) for the upper portion of the pin and indicated that some models in FRAP and, thus, in WRAP-PWR-EM may be excessively conservative since similar analyses with NRC-approved codes yielded lower temperatures.

Follow on studies have identified and corrected errors in the WRAP-PWR-EM reflood models based upon FLECHT, but these changes did not alter the overly conservative behavior.

3. THE WRAP-PWR-EM SYSTEM

The WRAP-PWR-EM system[2] is a major extension of the WRAP[4,5] system developed at SRL during 1977. WRAP is a modified version of the RELAP4* code[6] with an extensively restructured input format, a dynamic dimensioning capability and additional computational capabilities such as an automatic steady-state option and an automatic restart capability with provision for renodalization. The capabilities of the WRAP-PWR-EM system include:

- Calculation of the initial fuel condition,
- Calculation of the initial thermal-hydraulic state of the system,
- Calculation of the blowdown phase of the LOCA,
- Calculation of the refill phase of the LOCA,
- Calculation of the reflood phase of the LOCA, and
- Calculation of the temperature of the fuel in the hottest pin in the core.

The overall structure of the WRAP-PWR-EM system is shown in Figure 1. Initial fuel conditions are calculated as a function of burnup by the GAPCON[7] module. These conditions are passed to FRAP[8] and WRAPIT, the generalized input processor, for initialization of the transient fuel models. GAPCON results are also stored on magnetic tape or disk for use in subsequent calculations.

The WRAP PWR steady-state initialization procedure, PWRSS,[9] as well as the RELAP4 initialization in which residual flow resistances are computed to balance the system, is contained in the WRIN module. The blowdown phase of the LOCA is calculated by the TWRAM module (most of the RELAP4/MOD5 code is contained in this module) with transient results stored on tape and/or disk for use in plotting via the WROP module and to provide fuel thermal conditions for the hot pin analysis by the FRAP module.

At the end of bypass, when the direction of flow through the downcomer again is into the lower plenum, the refill models within TWRAM are invoked to calculate the time at which the lower plenum is full. During this interval, the heat conduction equations for

*The WRAP-PWR-EM system used for these verifications studies is based upon RELAP4/MOD5 Version 84.

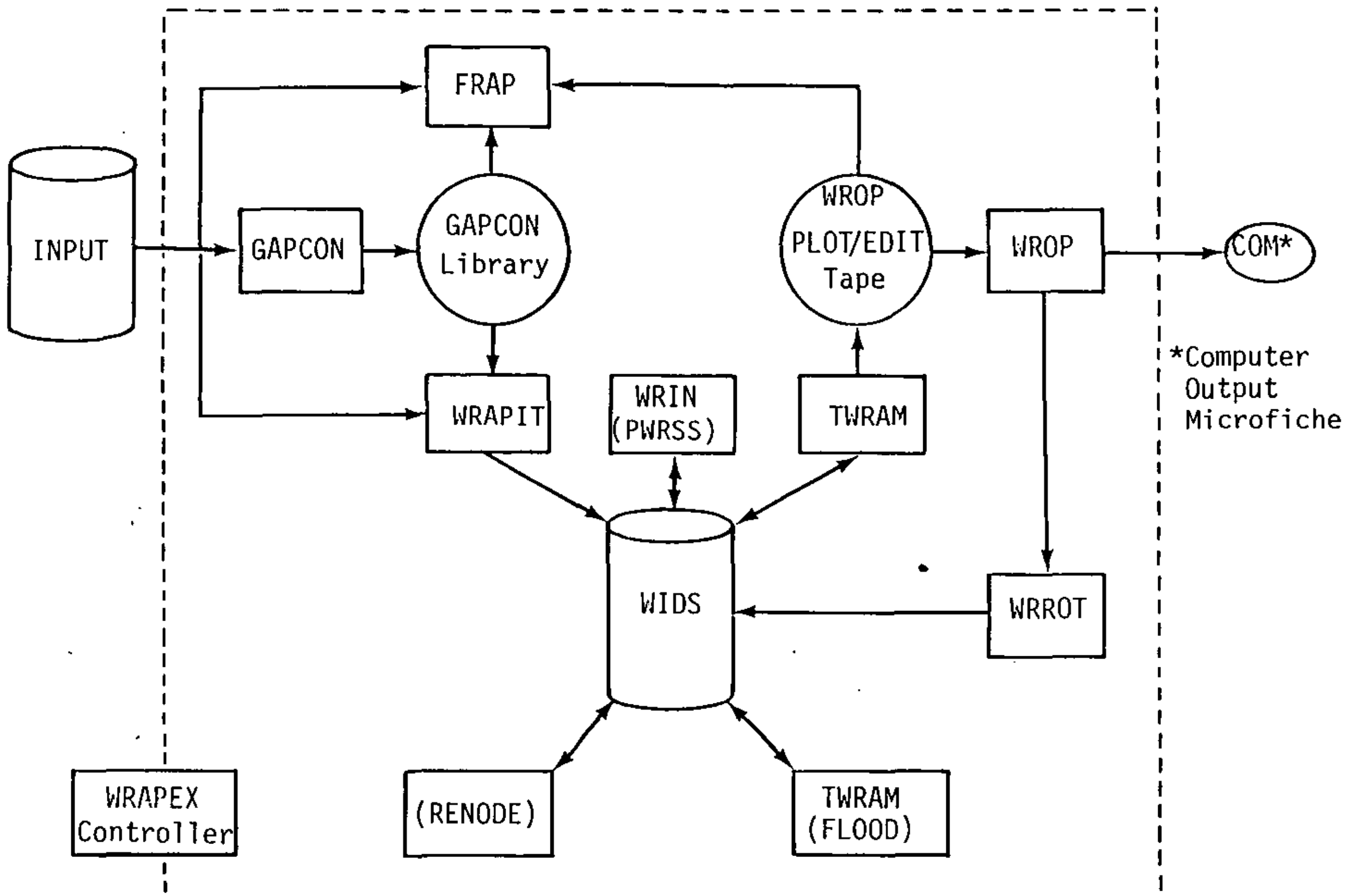


Figure 1 WRAP-PWR-EM system overview

the core are solved by using adiabatic boundary conditions. The time at which the lower plenum is full is denoted as the beginning of core recovery (BOCREC).

Following BOCREC, system renodalization is performed either manually or with the assistance of a utility module; and the reflood phase of the accident is calculated by the FLOOD option in the TWRAM module. The fuel thermal conditions during reflood are passed to FRAP for use in determining the thermal-hydraulic behavior of the hot pin. Other capabilities within the system include the transient restart capability provided by WRROT and by MWRROT, with the added capabilities of system renodalization and problem respecification. The overall execution of the various modules is controlled by the executive module, WRAPEX.

A detailed discussion of the component modules and input data requirements are given in References 2 and 10.

4. VERIFICATION CALCULATIONS

In the following subsections, each of the analyses for the verification of the WRAP-PWR-EM system is discussed in detail. For each analysis, pertinent background information is supplied; results are summarized, and conclusions are drawn; input specifications are given; and, selected output data are discussed. Where appropriate, results of additional analyses and uncertainties in the calculations are reported.

4.1 Semiscale Test S-06-3

4.1.1 Background

Comparing calculations to measured data is a key step in code verification. For an evaluation model (EM) code, models are selected to ensure a conservative calculation; thus, EM calculations are not expected to match experiments but the calculations should have the same general behavior and yield higher peak surface temperatures than the experiments.

For the verification of the WRAP-PWR-EM system, calculations were made of the Semiscale S-06-3 experiment. The WRAP results were to be compared with both the experimental data and with best-estimate (BE; i.e., no conservatisms present) calculations at INEL with RELAP4/MOD6. This test was somewhat limited because Semiscale was a non-nuclear, electrically heated facility. Therefore, only the blowdown, refill, and reflood modules were used. The GAPCON and FRAP modules, used only for nuclear PWR systems, were not used.

An isometric drawing of Semiscale taken from Reference 11 is pictured in Figure 2. As shown, the system has two distinct loops: an intact loop with a pressurizer, pump, and steam generator, and a blowdown loop with a simulated pump and steam generator. Though not shown, Semiscale has an accumulator, a low pressure injection system, and a high pressure injection system for introducing emergency core coolant into the cold leg of the intact loop. The electrically heated core consists of 40 heater rods; the central four are high power rods; the outer 32 are low power rods; and four are unheated.

4.1.2 Summary and Conclusions

WRAP-PWR-EM was run successfully for all phases of the S-06-3 test; however, the calculation was nonconservative because the calculated peak clad temperature was lower than both experiment and INEL-BE calculations. WRAP predicted PCT of about 1093°K at 43 sec, at the start of the reflood phase. The measured PCT reached 1155°K at about 20 sec, which was during the blowdown

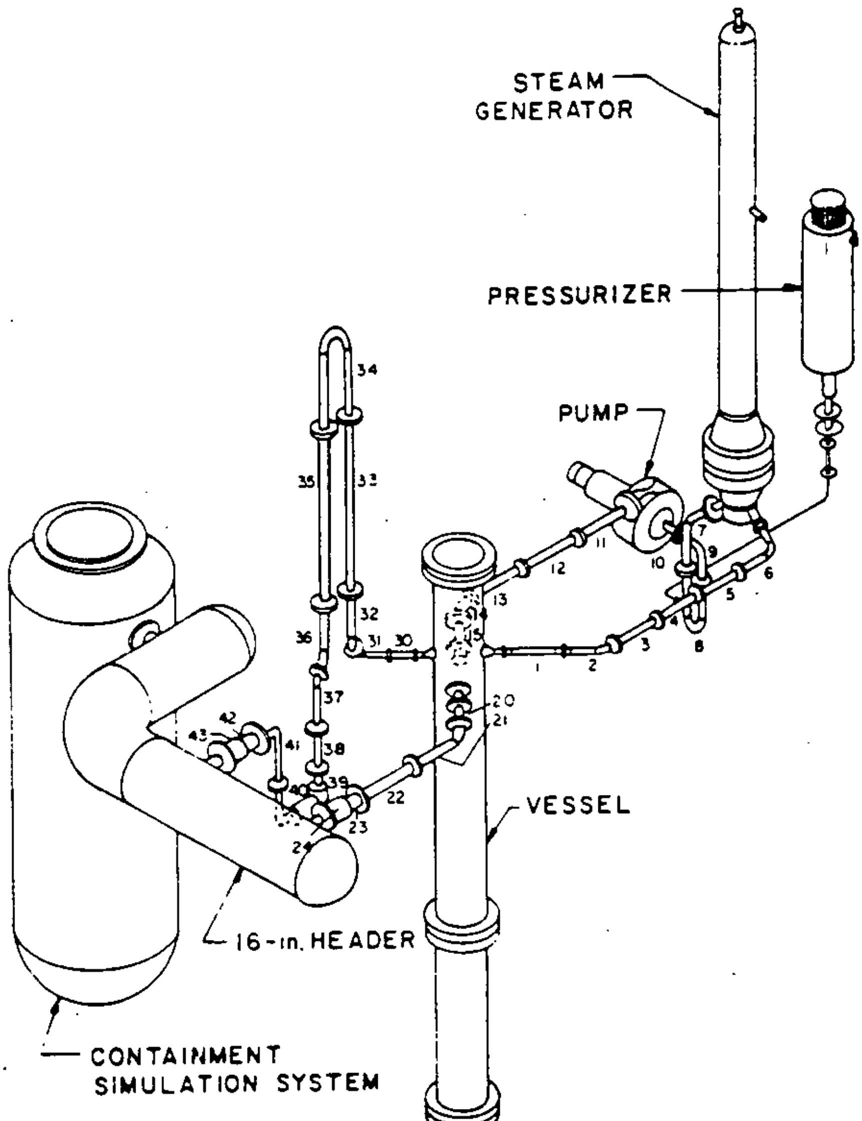


Figure 2 Isometric view of semiscale Mod-1 system

phase; and the INEL-BE calculation predicted a PCT of about 1165°K to occur at about 21 sec, also during the blowdown phase. A best estimate calculation with WRAP paralleled the INEL-BE result up to 17 sec at which point the WRAP calculation was halted. The agreement suggests that the nonconservative EM results were not due to WRAP coding errors.

Surface temperatures calculated by WRAP-EM were as much as 200°K lower than the measured data during blowdown. The conservative assumption that the core heats up adiabatically during refill resulted in a sharp rise in the calculated temperature which was not observed in the experiment. However, the rise in temperature during refill was insufficient to restore the overall conservatism of the EM calculation versus the measurement.

It should be emphasized that not all Appendix K rules were applied, particularly those applying most directly to nuclear fuels. The overpower, power distribution and decay heat rules, if applied analogously to the electrically heated Semiscale calculation, might have restored the conservatism. Only the rules concerning EM calculational controls were invoked. In addition, Appendix K requires a range of discharge coefficients to be investigated. This study used only a single value. The EM break flow model with the same discharge coefficient used in the BE calculation gave larger discharge flows early in the blowdown. This resulted in a larger core flow which improved heat transfer and lowered the cladding temperatures. In addition, the EM heat transfer coefficients after critical heat flux were higher than the BE coefficients for equal flows. These findings suggest that the effects of individual specifications in Appendix K which govern EM calculations should be examined.

4.1.3 Input

Most of the input was derived from a RELAP4/MOD5 Version 72 deck of cards prepared by INEL for a pretest calculation.[12] Conversions for an EM calculation were made in partial compliance with the 10 CFR Part 50, Appendix K rules.[1] The nodalization used for the blowdown phase is shown in Figure 3. Since Semiscale is an experimental facility, the rules concerning power level and equipment operability were not applied. Only rules pertaining to the choice of calculational controls were followed. For the blowdown phase, only the BE control choices pertaining to post-CHF heat transfer, return to nucleate boiling after CHF, and critical flow did not already satisfy the EM rules of Appendix K. Once the critical heat flux was reached on any heat transfer surface and the surface temperature was more than 300°F above the saturation temperature, return to nucleate boiling on that surface was prevented by use of the EM heat transfer model.

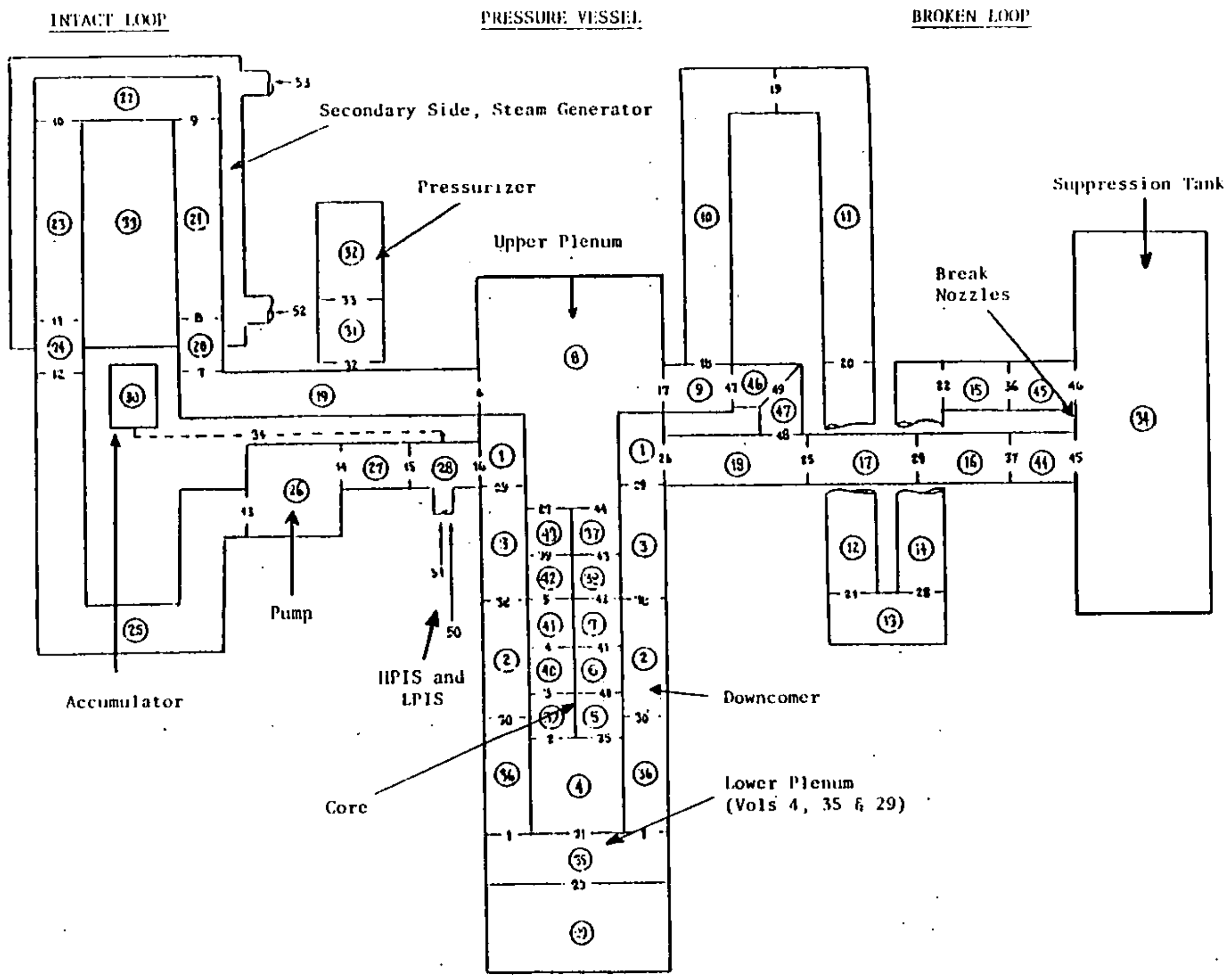


Figure 3 Semiscale blowdown nodalization (taken from Reference 12)

The Henry-Fauske subcooled critical flow model, used in the BE calculations, is satisfactory for EM calculations, but the saturated critical flow model is restricted to that of Moody by Appendix K rules. The break discharge coefficient as well as the Henry-Fauske and Moody flow multipliers were set to 1.0. The Appendix K rules require a series of calculations searching for the discharge coefficient between 0.6 and 1.0 leading to the highest peak cladding temperatures. However, the Verification Program included only a single calculation.

The WRAP-FLOOD calculations require single downcomer, single lower plenum, single core, and single upper plenum volumes. In addition, the cold leg volumes and junctions connecting the downcomer with the suppression tank through the break nozzle must be replaced by a pair of junctions, one for steam flow and the other for water flow. The secondary side of the steam generator must have two volumes, one for water and the other to act as a steam dome. The reflood nodalization developed for S-06-3 is shown in Figure 4.

The reflood input data include the subcooling of the lower plenum and the peak linear power density per rod in the core for use in the FLECHT heat transfer correlation.[13] The subcooling ordinarily would be obtained from the output of the refill calculation, while the peak linear power density per rod ordinarily would be obtained from the combined inner and outer hottest slabs at BOCREC. Unfortunately, both the subcooling and the peak linear rod power density were outside the range of the FLECHT correlation used in the reflood calculation. Therefore, the NRC approved changes in the reflood input (see Table 1). The lower plenum subcooling and the peak linear power density were set equal to the lower bounds of the acceptable ranges. Additional input specifications are detailed in Appendix A.

4.1.4 Results

The chronology of events is shown in Table 2. The WRAP calculation was run out to 51 sec, at which time quenching of the bottommost core slabs had begun, and the calculation was stopped. The Central Processing Unit (CPU) time required was 185 min for 39,667 timesteps on the IBM 360/195. Selected results are discussed in the following paragraphs.

4.1.4.1 Peak Clad Temperature

In Figure 5, the WRAP-EM calculated temperature of the highest powered rod region is compared to the INEL-BE and experimental values. The blowdown phase of the WRAP-EM calculation follows the BE and experimental results for about 7 sec when the WRAP-EM cladding temperatures show a sharp decline. The EM calculation

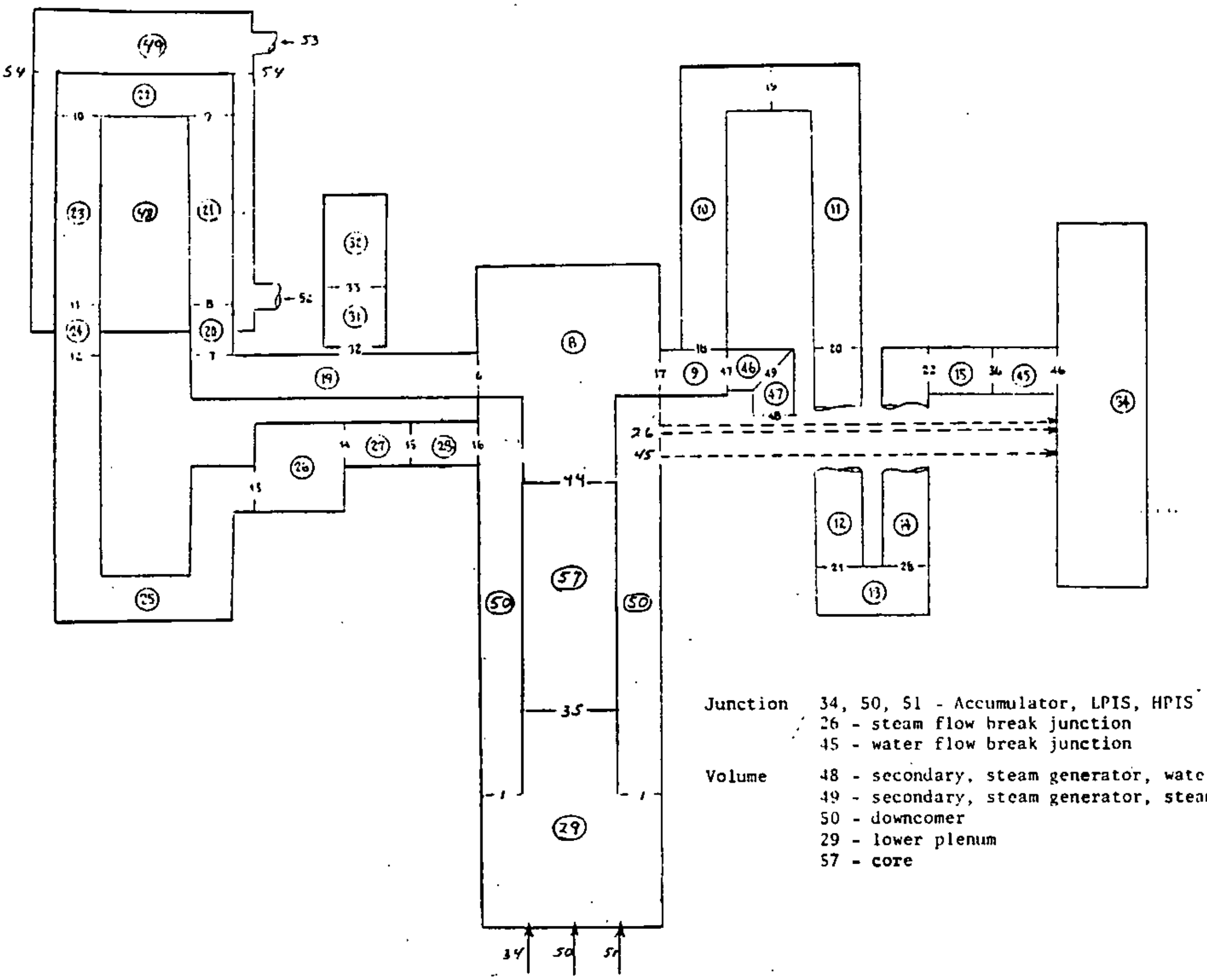


Figure 4 Semiscale reflood nodalization (modified from Reference 12)

Table 1 NRC-approved sources of reflood input data for semiscale

Item	"Proper" Source and Value	Changed Source and Value
1. Lower plenum subcooling at BOCREC.	Refill calculation 0.0°F.	Min value in range of FLECHT correlation* 16°F.
2. Max linear density per rod at BOCREC.	Sum of peak axial powers in inner and outer rods at BOCREC. 0.291 kW/ft/rod	Min. value in range of FLECHT correlation.* 0.51 kw/ft/rod*
3. Core heat slab centerline and surface temperatures at BOCREC.	Average of inner and outer rod heat slab temperatures at BOCREC defined to preserve stored energy	Inner rod heat slab centerline and surface temperatures. Conservative but does not preserve stored energy.
4. Total power at BOCREC.	Sum of power in inner and outer rod regions at BOCREC. Preserves total power.	No change.

*The range of acceptable values for use in the FLECHT correlation[13] is

Lower plenum subcooling	16 to 189°F
Peak power density	0.51 to 1.4 kw/ft

Table 2 Event chronology in Semiscale S-06-3

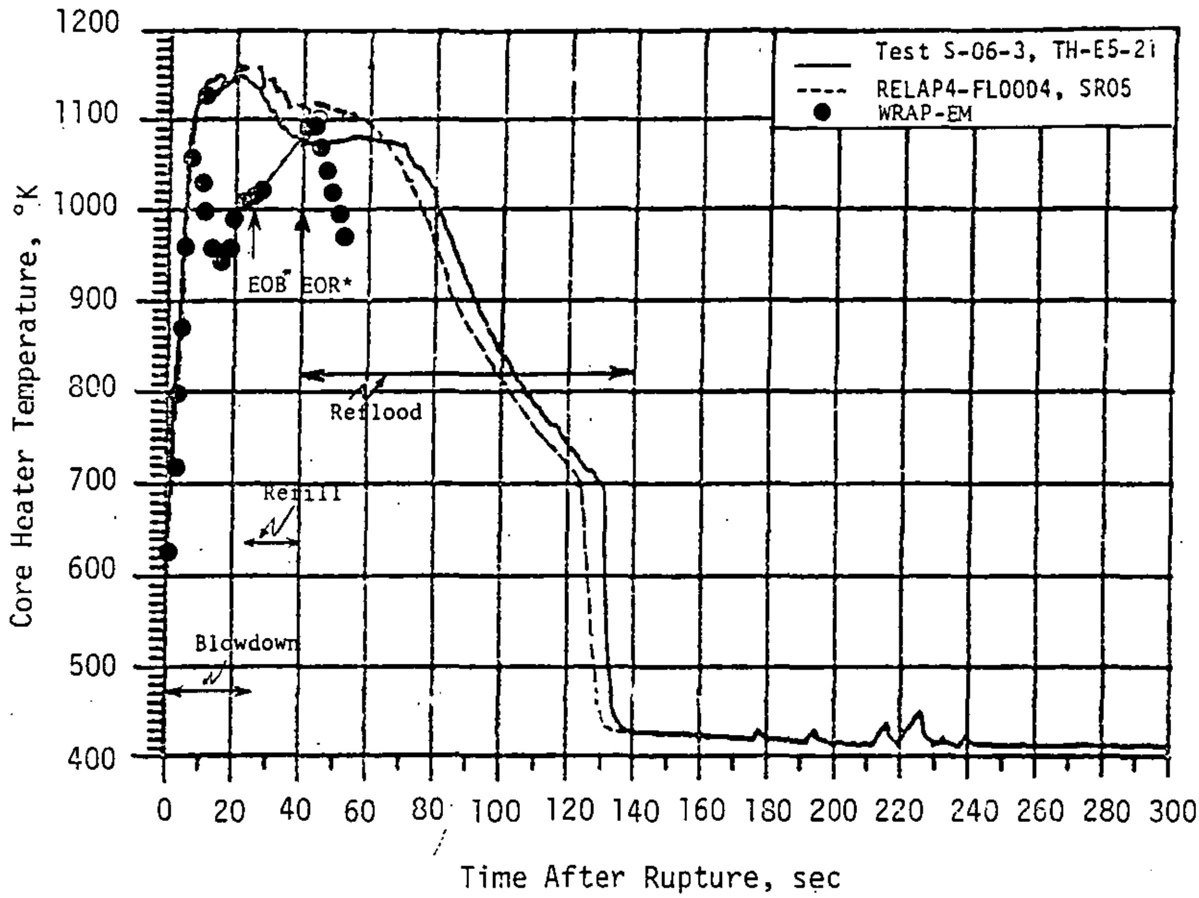
Event	WRAP-EM* Time (sec)	INEL-BE[12] Time (sec)	Experiment[12] Time (sec)
1. Begin closing water flow into secondary side of steam generator	0.0	0.0	8.0
2. Open break valves	0.0	0.0	0.0
3. Initiate power reduction	5×10^{-5}	5×10^{-5}	1.27
4. HPIS trip at 1800 psia	0.036	0.04	0.0
5. Begin closing steam flow from secondary side of steam generator	8.0	8.0	8.0
6. Accumulator trip at 600 psia	13.60	15.8	18.5
7. Steam generator water valve closed at	20.00	20.00	22.0
8. LPIS trip at 150 psia	20.76	26.0	25.5
9. Steam generator steam valve closed at	22.90	22.90	22.0
10. End of bypass	25.00	-	52
11. Cold leg 1/2 full	25.24	-	-
12. Lower plenum starting to fill (free fall delay 0.93) (hot wall delay 2.54)	28.71	-	63
13. Close valve between hot and cold legs on broken loop (Junction 49)	32.0	32.0	-

*With the Dougall-Rohsenow heat transfer correlation during blowdown.

(Table 2, continued)

Event	WRAP-EM* Time (sec)	INEL-BE[12] Time (sec)	Experiment[12] Time (sec)
14. Lower plenum filled (BOCREC) (subcooling - 0.0°F max linear power/ft/rod in core - 0.291 kW/ft/rod)	40.40	55.5	71
15. Maximum peak cladding temperature reached	43.0	20.5	20
16. Lowest core level quench	51	-	<90

*With the Dougall-Rohsenow heat transfer correlation during blowdown.



*EOR - End of Refill

*EOB - End of Bypass

Figure 5 Comparison of experimental, RELAP4-BE and WRAP-EM calculated PCT for semiscale S-06-3 through refill and reflood for the blowdown phase is nonconservative because calculated surface temperatures are as much as 200°K lower than measurements and BE calculations.

During the refill phase, the conservative assumption of adiabatic heatup caused the WRAP-EM calculated temperature to rise rapidly, and by the end of refill, the calculated temperature exceeded the measured value. During the early part of the reflood phase, the calculated temperature continued to rise and reached a maximum of 1093°K. Then the calculated temperature dropped off more rapidly than either the measurement or the INEL-BE calculation. However, this rate did not affect the PCT. The calculation was ended 12 seconds into the reflood phase (at 52 sec) following the start of quenching in the bottommost core slabs. In terms of PCT, the WRAP-EM calculation predicted a maximum of approximately 70°K less than measured.

4.1.4.2 Break and Core Inlet Flows

To make the EM cladding temperatures equal to or greater than the BE cladding temperatures will probably require that the EM core flows be somewhat smaller than the BE flows. This can be accomplished by using a smaller break discharge coefficient than the 1.0 value used thus far. Figure 6 shows WRAP results demonstrating that with a discharge coefficient of 1.0, the EM break flows are considerably larger than the BE break flows until about 17 sec. A reduction of the discharge coefficient (allowed, even required, by the Appendix K rules) will reduce the EM break flows and therefore the core flows. However, the scope of WRAP-EM verification program did not include multiple calculations.

An additional comparison is shown in Figure 7 where WRAP is compared to an independent RETRAN best-estimate calculation and experimental results. (These data are taken from Reference 11.) During the early part of the transient, the break flow calculated by WRAP-EM was larger than that measured. In Figure 8, a similar comparison is presented for the core inlet flow. Again, early in the transient, the core inlet flow calculated by WRAP-EM was larger (in the negative sense, i.e., exiting the bottom of the core) than the experimental value, therefore promoting better heat removal.

4.1.4.3 Additional Analyses

Two EM choices of the post-CHF, film boiling heat transfer correlations are offered to the user by the Appendix K rules: the Dougall-Rohsenow or the Groenveld 5.7. Two separate calculations, using one, then the other, were performed out to about 17 sec, but only the Dougall-Rohsenow calculation was continued to the end of the blowdown phase at about 25 sec because that correlation was specified by the NRC.

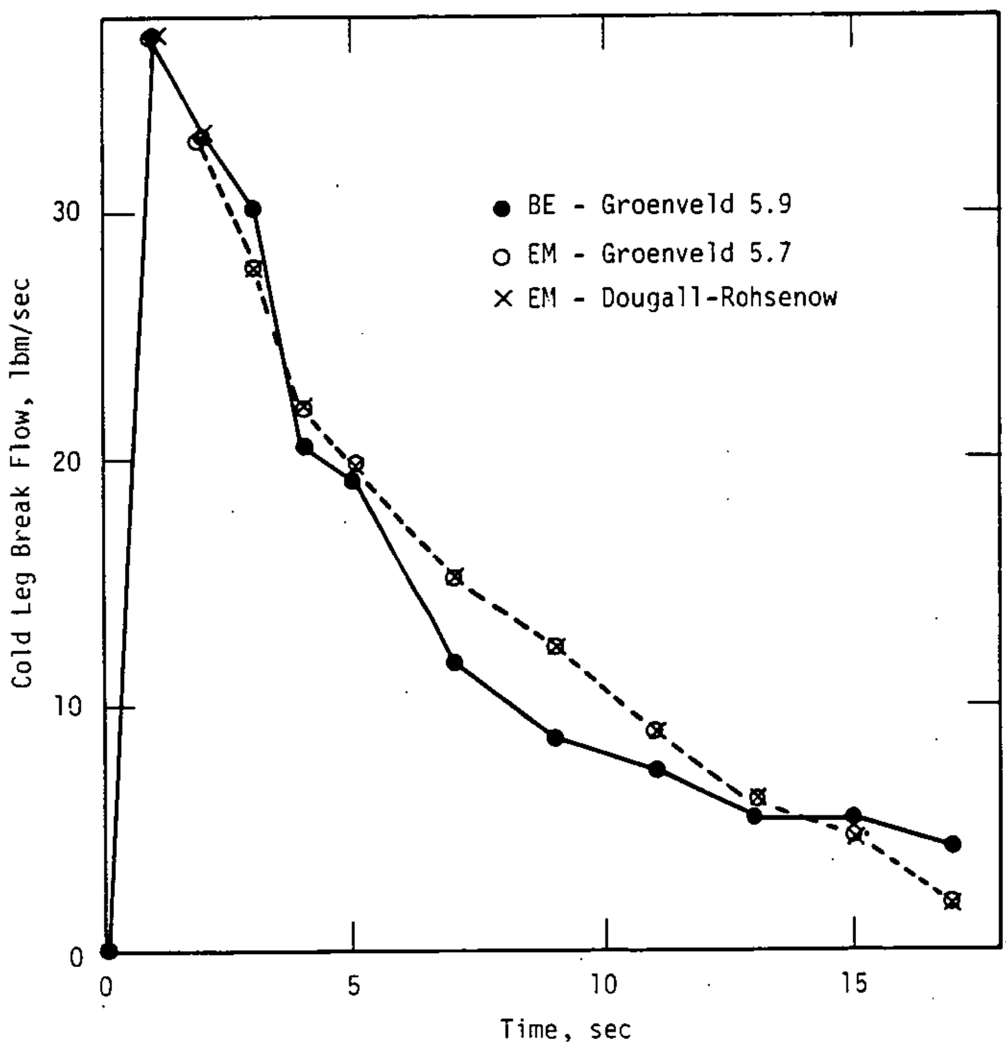


Figure 6 Cold leg break flow vs time in WRAP calculations

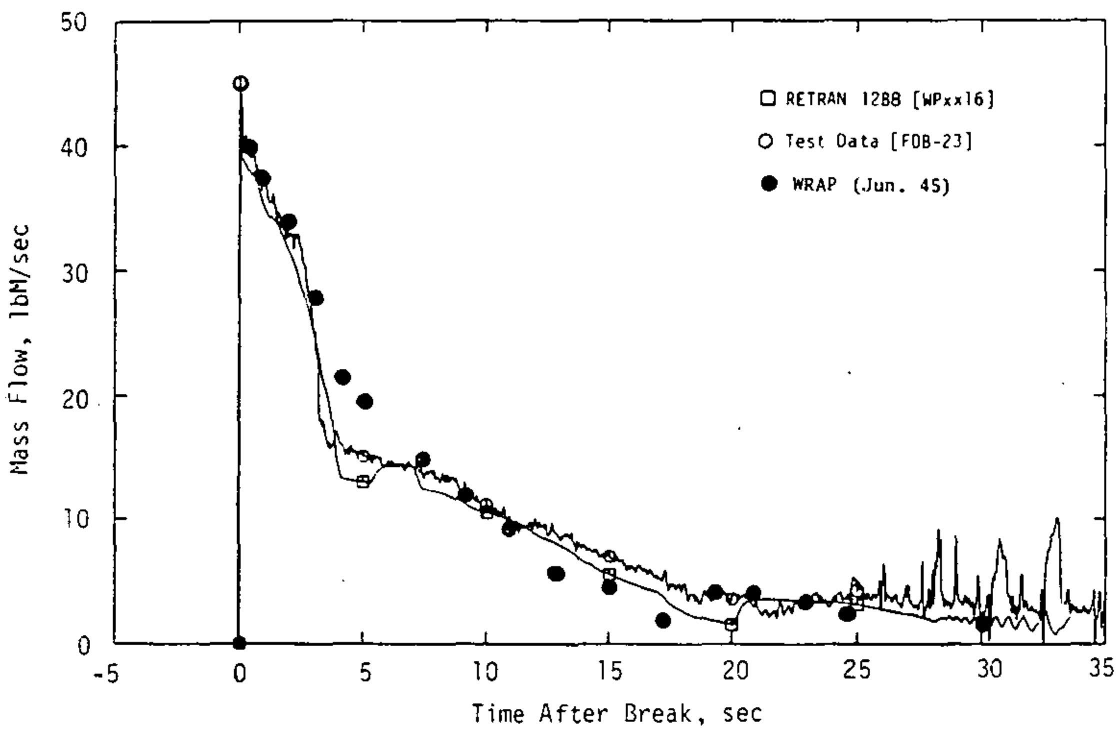


Figure 7 Flow rate from vessel side of break

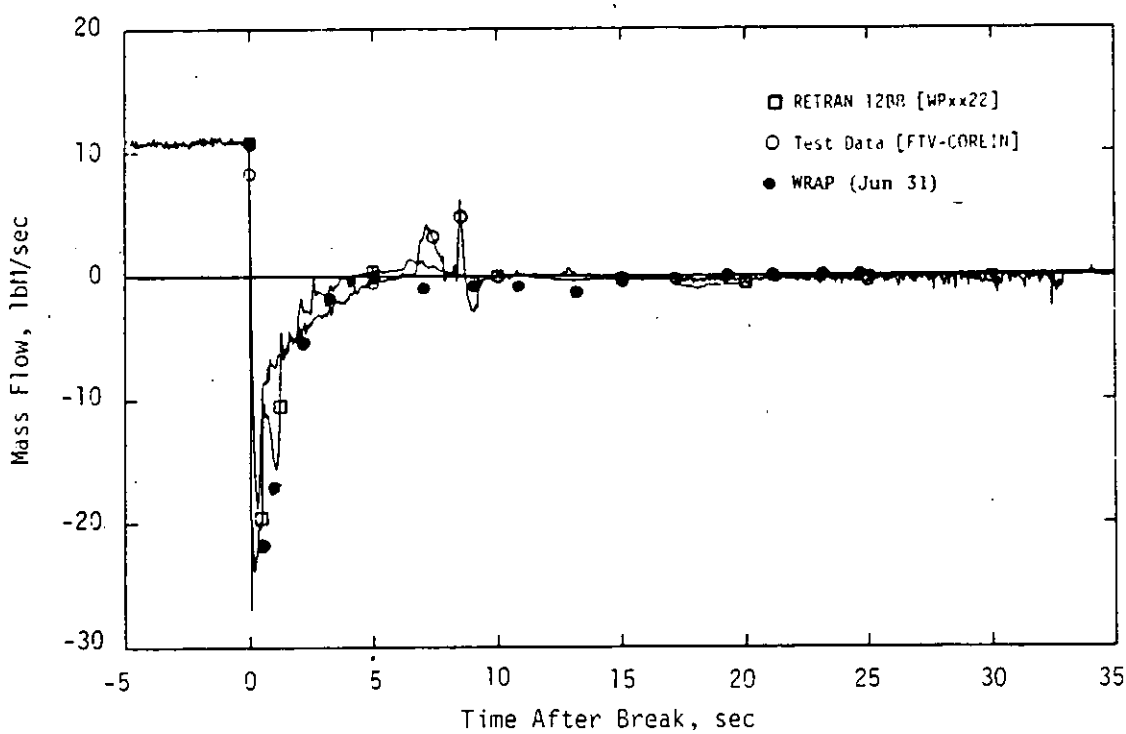


Figure 8 Flow rate at core inlet

In Table 3, the hot channel cladding surface temperature, the heat transfer mode, and the heat transfer coefficient for the two EM blowdown calculations are compared through 17 sec. The WRAP-BE results are also included in Table 3. (They closely match the INEL-BE results, with RELAP4/MOD5-Version 72, of Reference 12.)

Figures 9 through 11 show the surface temperatures, the heat transfer, and channel quality as computed by WRAP-BE and WRAP-EM with Dougall-Rohsenow and Groenveld 5.7. The results for the two EM calculations show the same general behavior, and both are less conservative, in terms of surface temperature, than the BE calculation.

4.1.4.4 Uncertainties

Because the boundary conditions for the Semiscale S-06-3 calculation with WRAP were taken from the INEL-BE pretest prediction rather than the experiment, discrepancies between input and measured data do exist. However, the boundary condition discrepancies appear to cause little error in terms of hot channel cladding temperatures as shown by the excellent agreement of the INEL-BE and experimental values. It can be argued that the 2% overpower rule, the power shape rules, and the 20% decay overheat rule of Appendix K should have been observed in the WRAP-EM study even though Semiscale is an electrically heated facility. Most likely, observation of the first rule would cause only minor increases in the WRAP-EM cladding temperatures, but the latter two rules could have very significant effects.

The major cause of discrepancy between the WRAP-EM and the experimental values is the break discharge coefficient value (1.0) used with the Moody critical flow model in the calculation. The Appendix K rules call for a search for the discharge coefficient leading to the highest PCT. However, multiple calculations were not included in this study.

Sensitivity studies for other input variables also are in order, principally, the models of emergency coolant injection and downcomer penetration during the refill and reflood stages. The sensitivity to lower plenum subcooling, core slab temperatures, and peak linear power density during the reflood phase should also be determined.

As discussed in the following sections, these verification analyses indicated that the reflood models need to be reviewed. If the models are updated, the reflood analysis for Semiscale could be repeated to try to resolve the difference in the rates of quenching between WRAP-FLOOD and measurements.

TABLE 3

SEMISCALE S-06-3

Comparison of Selected Heat Slab #5 (Hot Axial Level) Results from WRAP-BE and EM Calculations*

Time (sec)	Cladding Surface Temperature			Heat Transfer Mode			Heat Transfer Coefficient		
	SRL-WRAP** BE without HT update	SRL-WRAP EM with Doug.-Rohs.	SRL-WRAP EM with Groen. 5.7	SRL-WRAP** BE without HT update	SRL-WRAP*** EM with Doug.-Rohs.	SRL-WRAP*** EM with Groen. 5.7	SRL-WRAP** BE without HT update	SRL-WRAP EM with Doug.-Rohs.	SRL-WRAP EM with Groen. 5.7
0.0	660.2	660.2	660.2	2	2	2	5.05×10^4	5.05×10^4	5.05×10^4
1.0	831.5	812.3	831.1	4	19	14	4.48×10^2	6.65×10^2	4.38×10^2
2.0	953.5	919.8	952.2	5	29	25	1.93×10^2	3.48×10^2	2.01×10^2
3.0	1010.8	989.9	1037.9	5	29	25	1.22×10^2	1.53×10^2	9.73×10^1
4.0	1182.4	1120.9	1177.3	5	29	25	6.14×10^1	1.06×10^2	7.26×10^1
5.0	1355.4	1280.3	1337.3	5	29	25	2.01×10^1	6.69×10^1	4.85×10^1
7.0	1558.1	1451.5	1526.1	5	29	25	2.95×10^1	1.02×10^2	6.34×10^1
9.0	1605.2	1402.4	1519.1	5	29	25	3.90×10^1	1.44×10^2	8.27×10^1
11.0	1617.5	1341.5	1497.2	5	29	25	3.53×10^1	1.52×10^2	8.58×10^1
13.0	1627.0	1267.5	1458.3	5	29	25	3.48×10^1	1.61×10^2	9.49×10^1
15.0	1636.0	1257.8	1450.2	7	28	28	1.78×10^1	3.51×10^1	3.91×10^1
17.0	1640.4	1280.8	1469.8	8	28	28	2.01×10^1	2.49×10^1	2.37×10^1

* English Units: Temperature = ^{o}F, HT Coefficient = Btu/hr - ft^2 - ^{o}F.
The EM calculations do not meet all 10 CFR Part 50, Appendix K requirements.

** The BE calculations use the Groenveld 5.9 heat transfer correlation for post CHF film boiling.

*** 10 is added to the heat transfer mode indicator when the CHF is reached and 20 is added when a return to transition boiling would occur were it not precluded.

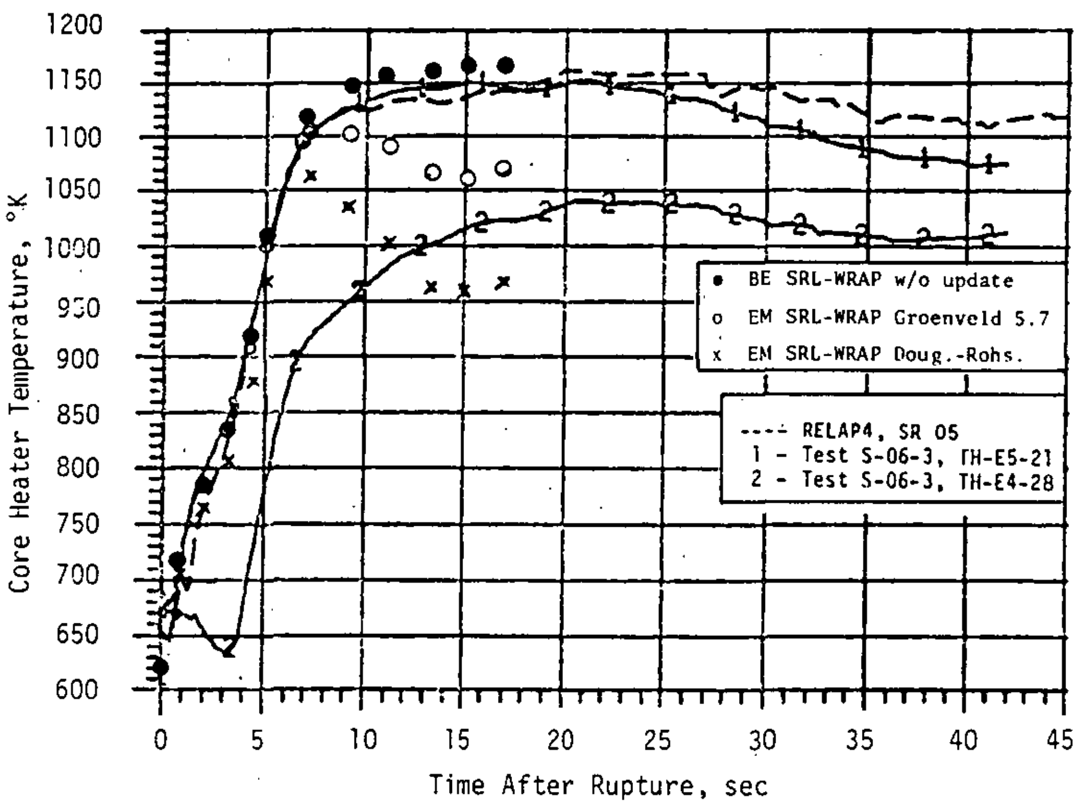


Figure 9 Comparison of SRL-BE and EM calculated hot channel cladding temperatures

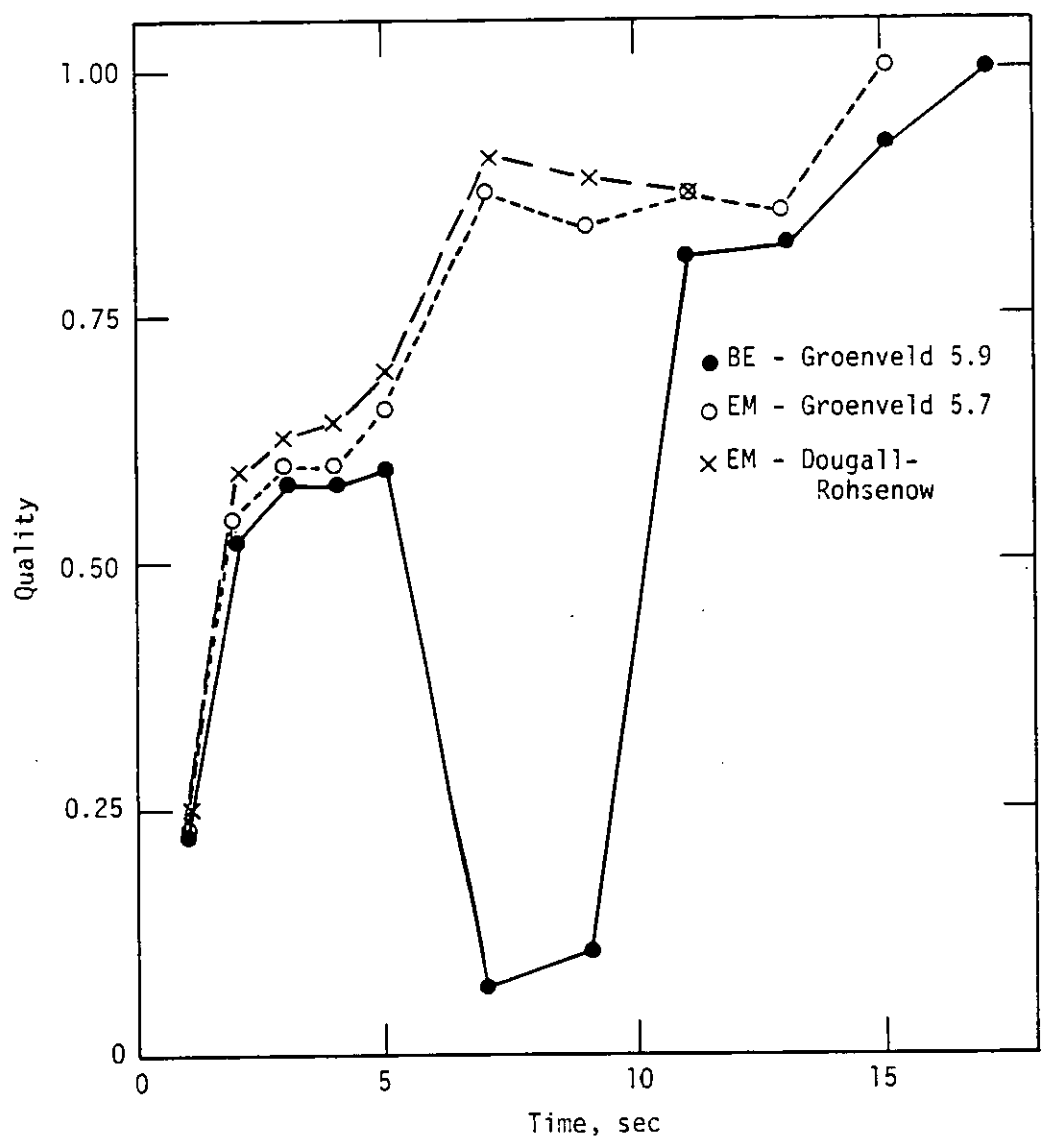


Figure 10 Predicted quality vs time in WRAP calculations at core level 5 of the hot channel

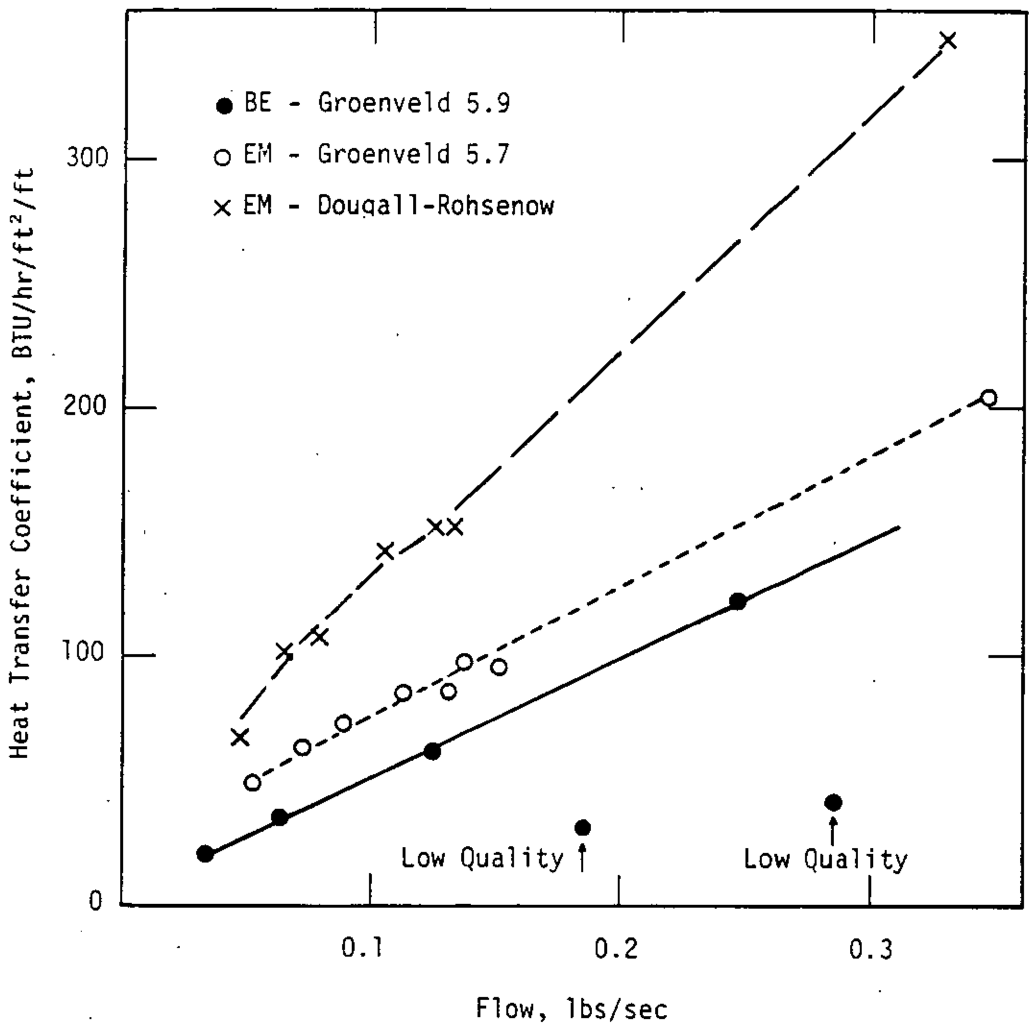


Figure 11 Predicted heat transfer coefficients vs flows in WRAP calculations at core level 5 of the hot channel

4.2 LOFT Test L2-3

4.2.1 Background

NRC selected the LOFT L2-3 test, which simulated a complete double-ended offset shear break of a large PWR inlet pipe, for one of the calculations of the WRAP-PWR-EM verification study. A LOFT test was selected because LOFT is the only complete nuclear reactor system in the Reactor Safety Research Program of the NRC designed to provide data on the behavior of commercial PWRs during postulated loss-of-coolant accidents (LOCAs) and associated transients. A schematic of the LOFT test facility[14] is shown in Figure 12.

The LOFT L2-3 model was developed and used in 1979 for NRC staff calculations. BE thermal fuel models were used to obtain closer approximations to test results than would be computed with EM models. Calculations performed by the NRC staff using the EM fuel models showed excessive conservatism in the fuel and clad temperature predictions; thus, continued use of these models in the SRL verification calculations was not considered useful by the NRC. A break model which preserved the best estimate discharge flow rate was specified by NRC. This choice was made because a key objective of this calculation was to evaluate the conservatism of the EM heat transfer models during blowdown and the WRAP models for refill and reflood.

4.2.2 Summary and Conclusions

All phases of the LOFT L2-3 test, blowdown, refill, and reflood, were calculated with WRAP-PWR-EM. WRAP results were conservative because they gave higher clad surface temperatures (1030°K at 43.5 sec after the break) than were measured in the test (at 914°K at 5 sec after the break). WRAP clad surface temperatures were also higher than those predicted in an INEL BE calculation (970°K at 4.5 sec after the break).

WRAP predicted that the peak clad temperature would be reached later in the transient than was observed in the test and was predicted in the INEL BE calculation, primarily because of the EM model assumption that the core heats up adiabatically during refill. The WRAP reflood analysis predicted that the hottest level of the core would quench much later (130 sec) than was measured (55 sec) or than was predicted by the INEL BE calculation (70 sec).

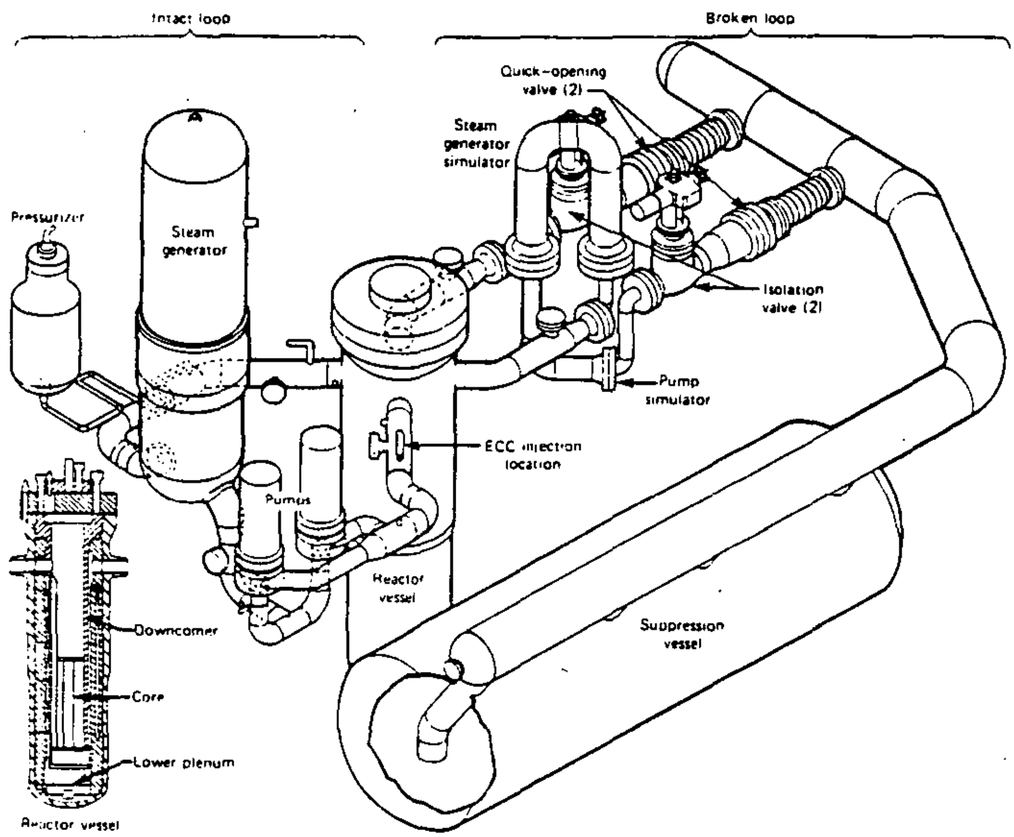


Figure 12 Schematic of LOFT test facility

4.2.3 Input

The nodalization used in the blowdown and refill calculations is shown in Figure 13, and a brief description of each control volume is given in Table 4. The nodalization was set up for a series of calculations; therefore, it contains volumes and junctions which were not involved in this calculation.

Figure 14 gives a more detailed picture of the core region, which is composed of two parallel channels, one representing the hot channel and the other representing the remainder of the core. The fuel in the core is represented by three vertical heat slab stacks. One stack of thirteen heat slabs represents the highest powered pin in the hottest assembly; the second stack of six heat slabs represents the remaining pins in the hottest assembly, and the third stack of six heat slabs represent the remainder of the core. The data for these slabs were obtained from NRC and represent BE fuel conditions.

A mixture of BE and EM models was selected for the blowdown calculation:

- The BE Henry-Fauske-HEM critical flow model was specified at the break junction.
- The EM Moody critical flow model for saturated flow was used at all other junctions.
- The B & W-2, Barnett, and Modified Barnett CHF correlations were specified.
- The Dougall-Rohsenow film boiling correlation was specified.

Bubble rise and vertical slip were used to account for two-phase flow characteristics. A Wilson bubble rise model was used for the pressurizer (control volume 39). In order to model instant phase separation in the accumulators (control volumes 41 and 63), a bubble velocity of 10^6 ft/sec with a gradient of 0.8 was used. The secondary side of the intact loop steam generator (control volume 45) used a bubble velocity of 5.77 ft/sec and a gradient of 0.8. Slip was used in all vertically oriented junctions except in the core region.

Trips control the timing of the accumulator injection system, the low pressure injection system (LPIS), and the high pressure injection system (HPIS) as well as the opening of the break. The following is a brief summary of the trips.

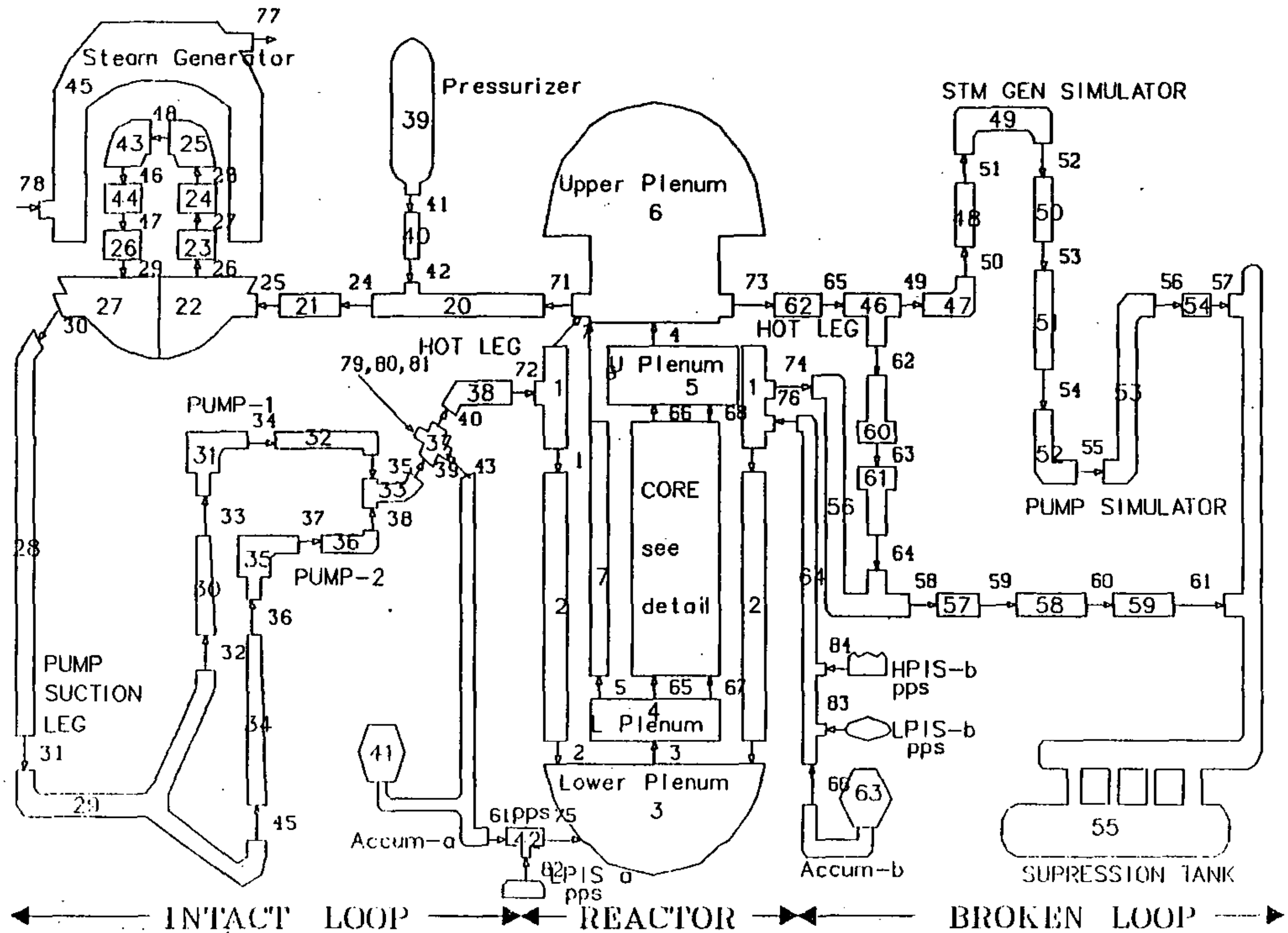


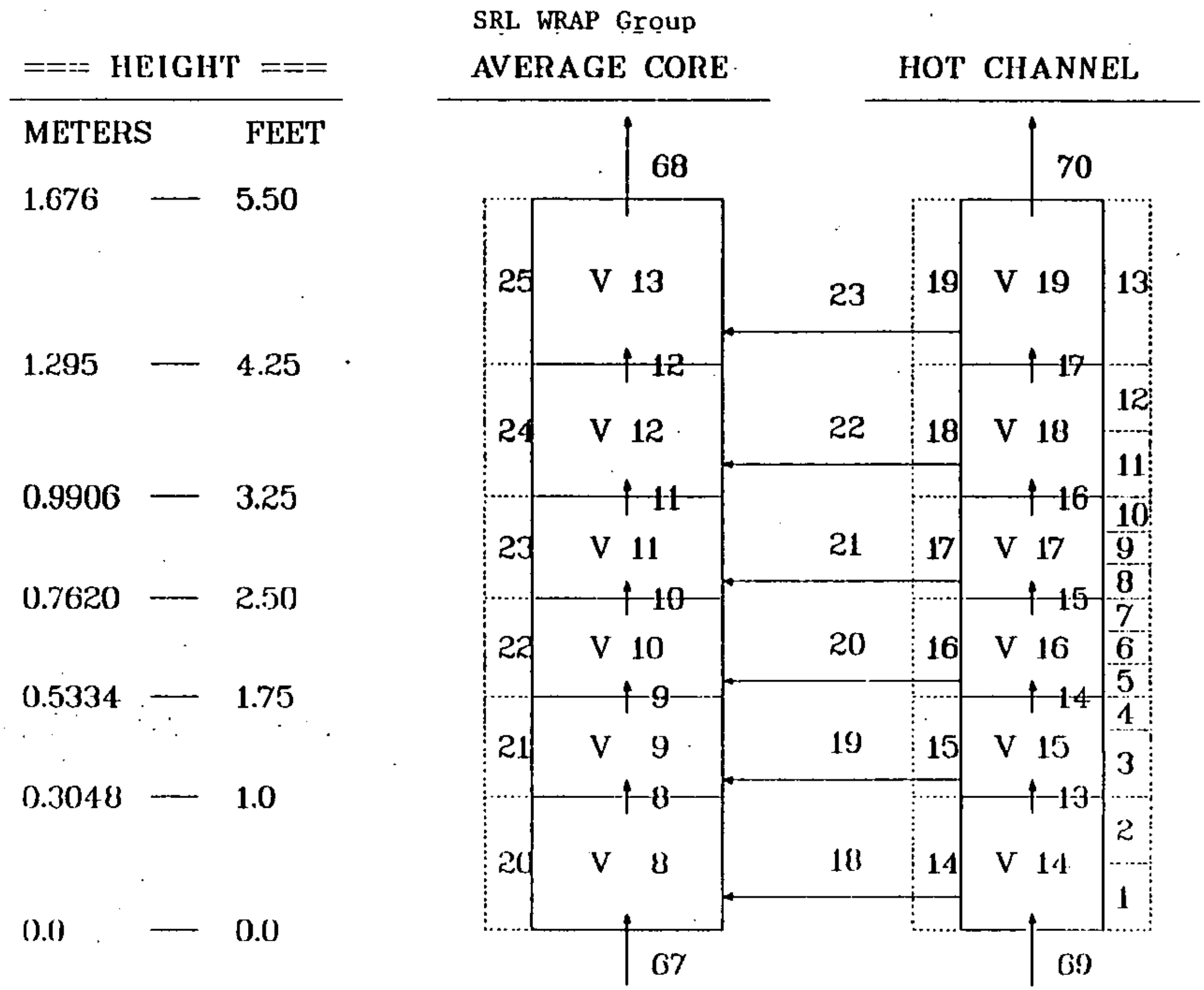
Figure 13 LOFT L2-3 blowdown nodalization

Table 4 LOFT L2-3 blowdown system model description

Control Volume	Description
1	Upper downcomer
2	Lower downcomer
3 and 4	Lower plenum
5 and 6	Upper plenum
7	Core bypass region
8 through 13	Average core (see Figure 3)
14 through 19	Hot channel (see Figure 3)
20 and 21	Intact loop hot leg
22	Steam generator inlet plenum
23,24,25,43,44,26	Steam generator primary side tubes
27	Steam generator outlet plenum
28,29,30,34	Coolant pump suction leg
31,35	Primary coolant pumps
32,33,36,37,38	Pump discharge to the reactor vessel
39	Pressurizer
40	Pressurizer surge line
41	ECCS Accumulator A
42	Accumulator A surge line
45	Steam generator secondary side
62,46	Broken loop hot leg
47 through 51	Steam generator simulator
52	Pump simulator
53	Pump side of the break node

(Table 4, continued)

Control Volume	Description
54	Lead to suppression tank (pump side)
55	Pressure suppression system and blow-down header
56	Broken loop cold leg
57	Reactor vessel side of the break node
58,59	Lead to suppression tank (reactor vessel side)
60,61	Broken loop bypass line
63	ECCS Accumulator B
64	Accumulator B surge line



The volumes are solid boxes.
The heat slabs are dotted boxes.

Figure 14 LOFT L2-3 Core nodalization

- Leak - The leak is initiated at time = 0 sec.

- Scram - The scram is initiated when the pressure in the intact loop hot leg falls below 2059.3 psi and has a delay time of 0.87 sec.

- Pumps - The pumps remain on throughout the L2-3 experiment.

- Accumulator - The accumulator trips on when the pressure in the intact loop goes below 637 psi. The accumulator is valved off when the level in the accumulator drops below 4.16 ft.

- HPIS - The high pressure injection system is initiated when the mixture level in the pressurizer (control volume 39) goes below 0.68 ft. The delay time is 6.0 sec. The HPIS flow is 3.4648 lbm/sec for the entire transient.

- LPIS - The low pressure injection system is initiated when the mixture level in the pressurizer (control volume 39) goes below 0.68 feet. The delay time is 15 sec. The LPIS flow is input from a fill table as a function of system pressure.

- Feedwater - The feedwater flow begins at 42.971 lbm/sec and is assumed to ramp to zero in 5 sec.

- Steam line - The steam line flow begins decreasing at time = 0.0 sec and is assumed to ramp to zero by 14 sec.

The reflood nodalization developed for LOFT L2-3 is shown in Figure 15. The nodalization is based upon the NRC blowdown nodalization, and the general FLOOD guidelines given in the RELAP4 manual.[6]

Although the core bypass is shown in the nodalization in Figure 15, it was removed for the reflood calculation. Severe pressure oscillations were encountered in the bypass when it was included in the calculation. The presence of bypass in the reflood phase was not important. The INEL BE nodalization also removed the bypass from its blowdown nodalization when the reflood was completed. The INEL rationale for doing this was that the exclusion of the bypass would not affect the reflood results.[15]

The emergency coolant injection system required special renoding because WRAP does not have an appropriate counter-current flow model. The treatment of the emergency coolant system outlined in the INEL-ZION report[16] was adopted for the LOFT L2-3 reflood nodalization to avoid computational difficulties associated with possible counter-current flow in the downcomer. First, the accumulator volume was replaced with a fill junction. The

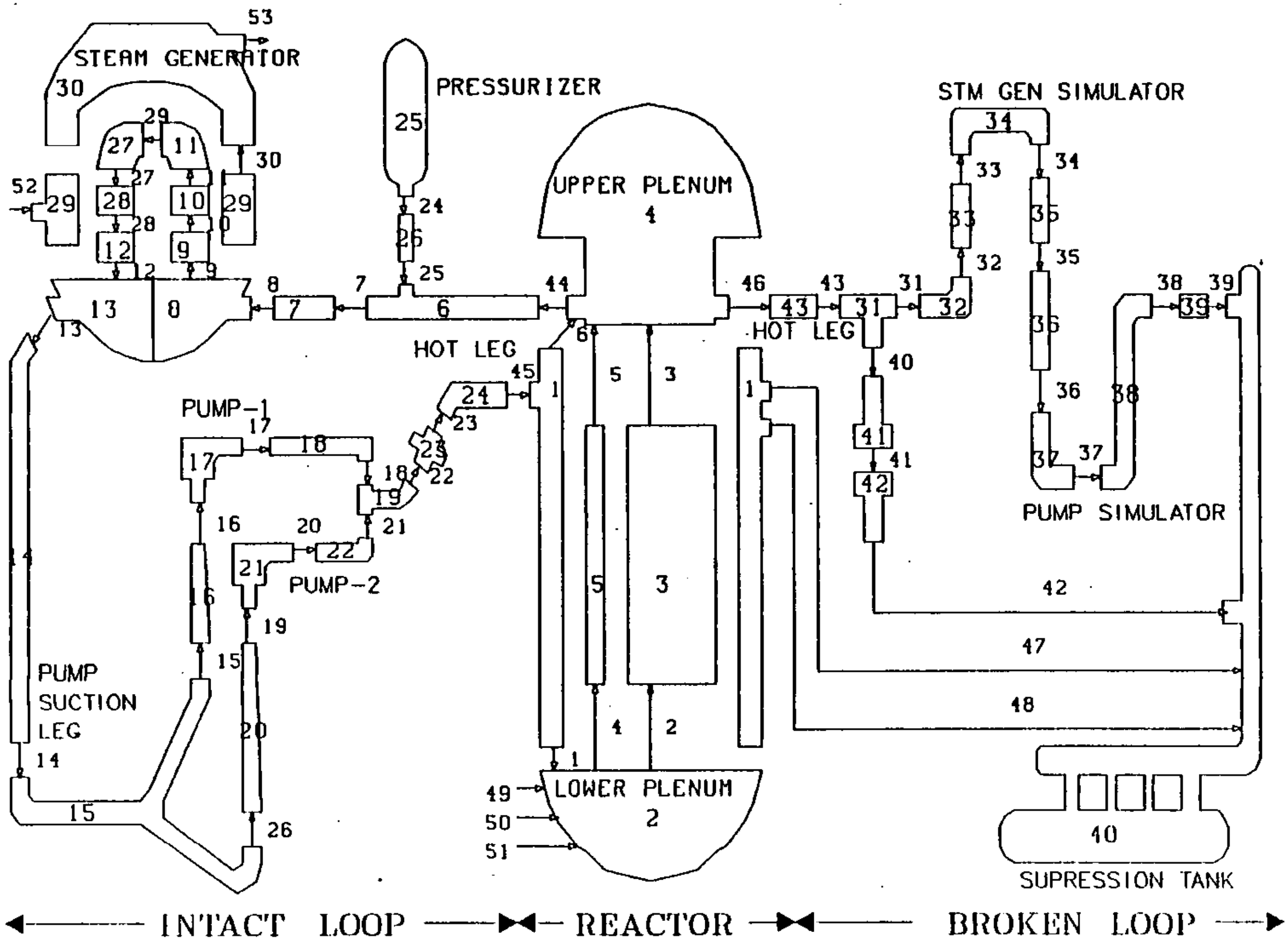


Figure 15 LOFT L2-3 reflood nodalization

time-dependent accumulator flow was input from a table generated during the refill calculation. The accumulator, HPIS, and LPIS fill junctions were removed from the cold leg of the intact loop and were placed at the bottom of the lower plenum.

The subcooling of the lower plenum and the peak linear power density per rod in the core to be used in the FLECHT heat transfer correlation were taken from the output of the refill calculation. The refill calculation gave no subcooling in the lower plenum, so the minimum value for the FLECHT correlation was used (16°F) in the LOFT L2-3 reflood analysis. (A later redefinition by NRC of the refill input--the effective portion of steam flow that completely condenses was changed from 100% to 60%--gave a subcooling of 20°F.) The power density was 0.512 kW/ft.

Based upon a recommendation by NRC, the core outlet enthalpy was held constant throughout the reflood calculation. The core enthalpy used, 1300 Btu/lb, was computed by using the enthalpy of superheated steam at the temperature of the secondary side of the steam generator at the end of bypass and the pressure of the core. Additional input details are given in Appendix B.

4.2.4 Results

WRAP-PWR-EM was run for each phase of the LOFT L2-3 LOCA and gave conservative results when comparing clad temperatures to experiment or BE calculations. In the blowdown portion of the analysis, some BE models were used to isolate the effect of the EM heat transfer options. The use of EM heat transfer options prevented the return to nucleate boiling and resulted in higher clad temperatures being computed during blowdown. In the refill portion of the transient when flow is coming down the downcomer and filling the lower plenum, the adiabatic heatup in the WRAP-EM model results in higher clad temperatures being computed than were observed in either the experiment or the BE calculations. During the reflood portion of the analysis, the WRAP-EM calculations reach their peak clad temperature.

The CPU time required for the full calculation was 221 minutes for 31,392 timesteps on the IBM 360/195. A more detailed discussion of the results is given in the following paragraphs. For reference an overall comparison of the timing of events in the test and in the WRAP and INEL BE calculations is given in Table 5.

4.2.4.1 Pressure Response

Figure 16 shows the WRAP calculated results for the core pressure as a function of time plus selected measured pressures. The

Table 5 LOFT L2-3

Comparison of Measured and Calculated Chronology of Events

	Time (seconds)		
Event	Experiment	RELAP4/MOD6	WRAP
Initiation	0.0	0.0	0.0
End of subcooled blowdown	0.05	-	0.03
First departure from nucleate boiling	0.96	0.5	0.05
End of subcooled break flow	3.0	4	4
Peak cladding temperature occurrence	4.95	4.5	43.5
First core wide rewet	8	70	>130
HPIS initiation	14.	15.2	15.1
Pressurizer empty	14	15	10.6
Accumulator initiation	16	15.6	15.2
LPIS initiation	29	28	29.1
End of blowdown/refill	-	44	41.2
Accumulator flow ended	56	59	46.2
Core reflooded	55	70	>130

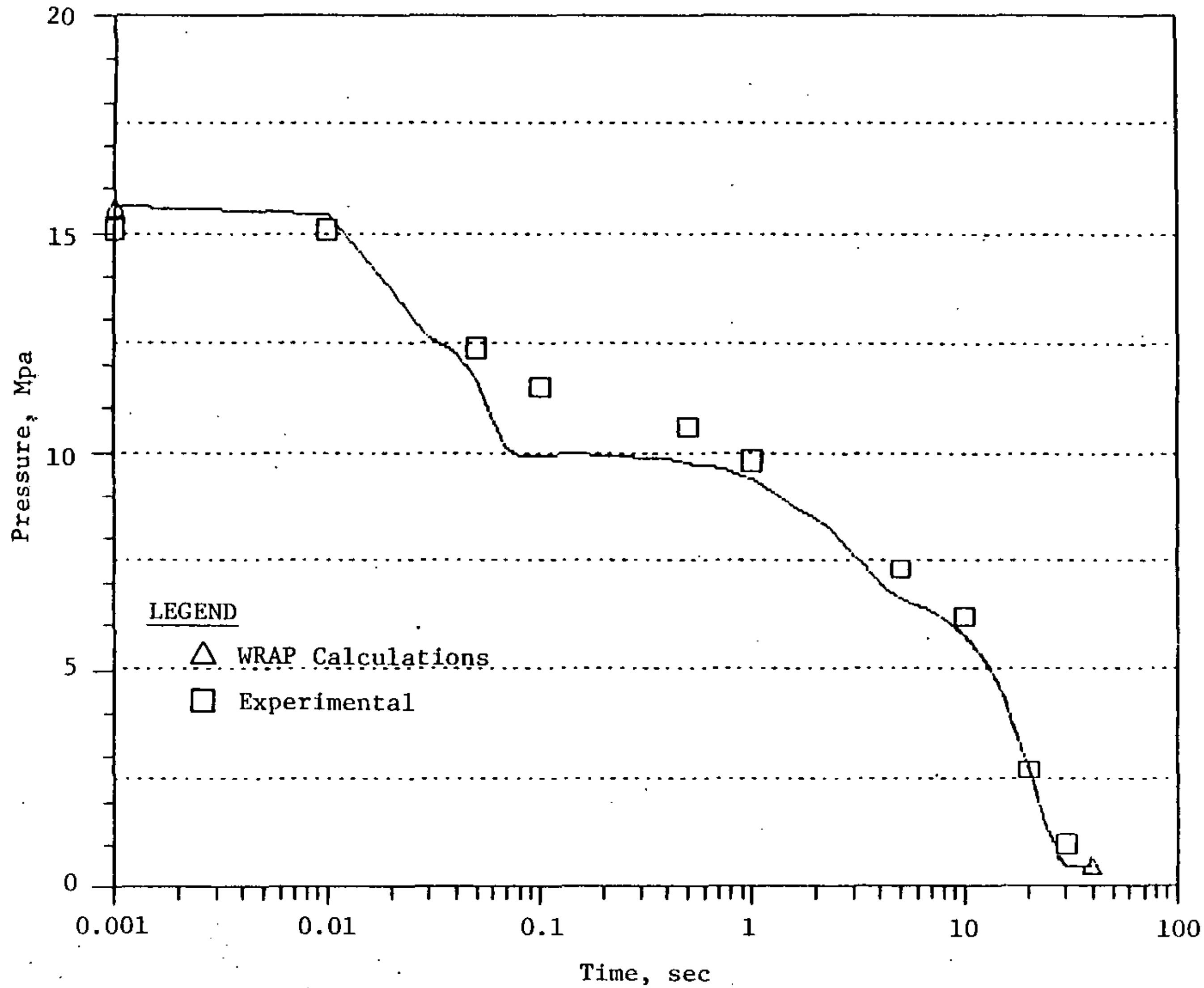


Figure 16 Core pressure LOFT L2-3 WRAP blowdown analysis

calculated results showed the same general behavior as the experiment but dropped off somewhat faster during the early stages of the blowdown. The good agreement between the WRAP results and experimental data demonstrates that the goal of calculating a BE blowdown was attained.

4.2.4.2 Mass Flow Response

Figure 17 shows the mass flow in the broken and intact loop cold legs for both WRAP and the L2-3 test (insert). An abrupt transition in the flow in the broken loop cold leg occurred as the saturation pressure was reached there (four sec), and from that time on the flow decreased rather gradually. The abrupt transition in the mass flow in the broken loop cold leg can be traced to the migration of hot fluid from the core up the reactor vessel downcomer to the broken loop cold leg. The migration of the hot fluid is illustrated in Figure 18 for both WRAP and the L2-3 test (insert) which shows the temperature response of the fluid at these locations.

4.2.4.3 Core Thermal Response

The thermal response of the fuel cladding is of considerable interest during a LOCA. Figure 19 illustrates the peak fuel rod cladding temperature measured during the L2-3 test and computed with WRAP. In the experiment the peak cladding temperature of 914°K occurred at five sec after the break; WRAP predicted a local maximum of 970°K at three sec. Thus, both the experiment and WRAP indicated a departure from nucleate boiling a few seconds after rupture.

A rewetting at around eight sec was also demonstrated in both experiment and calculation; however, the Appendix K criteria did not allow the EM calculation to return to nucleate boiling, thus the WRAP temperatures remained conservatively high. Shortly into the reflood phase (at 43.5 sec), the WRAP calculation registered its peak cladding temperature of 1030°K. During the reflood phase, the EM calculation predicted a much more gradual quenching of the clad temperatures. The behavior computed by WRAP was conservative because the WRAP-EM temperatures were higher than measured values, and quenching occurred later than in the experiment.

At 30 sec into the WRAP-EM calculation, end of bypass signals the beginning of the refill period and adiabatic heatup. At 41.2 sec into the WRAP-EM calculation, the water level reaches the

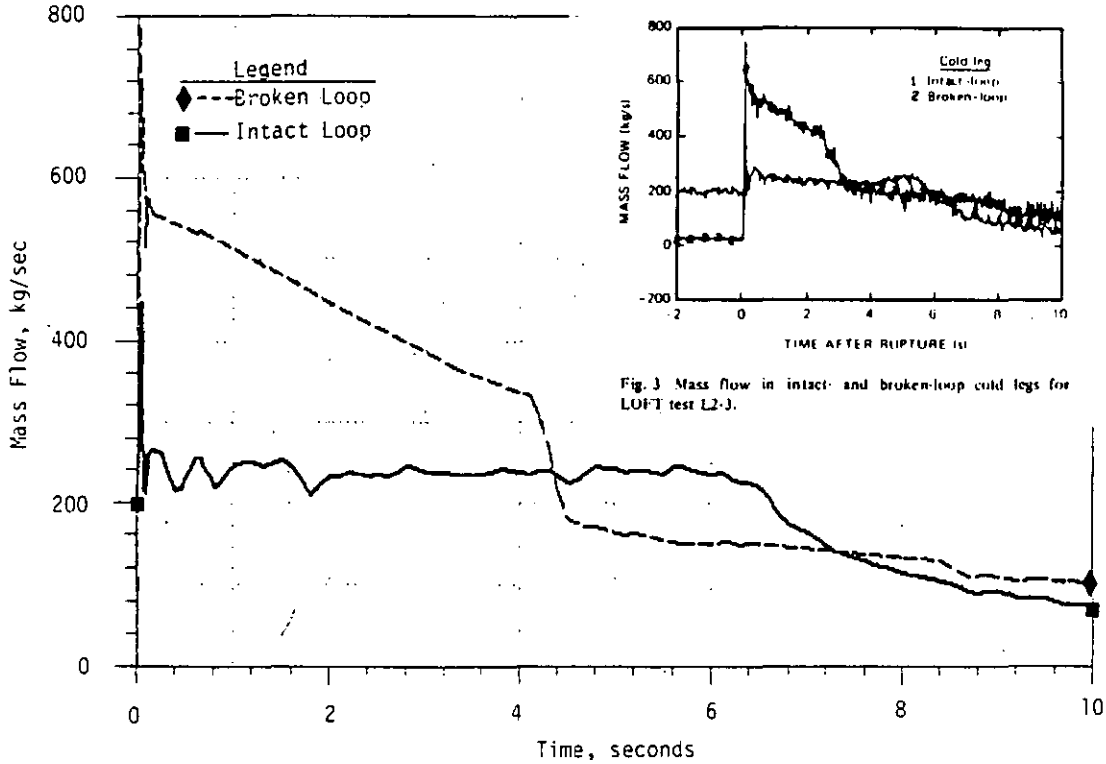


Figure 17 Intact and broken loop cold leg fluid LOFT L2-3 WRAP blowdown analysis

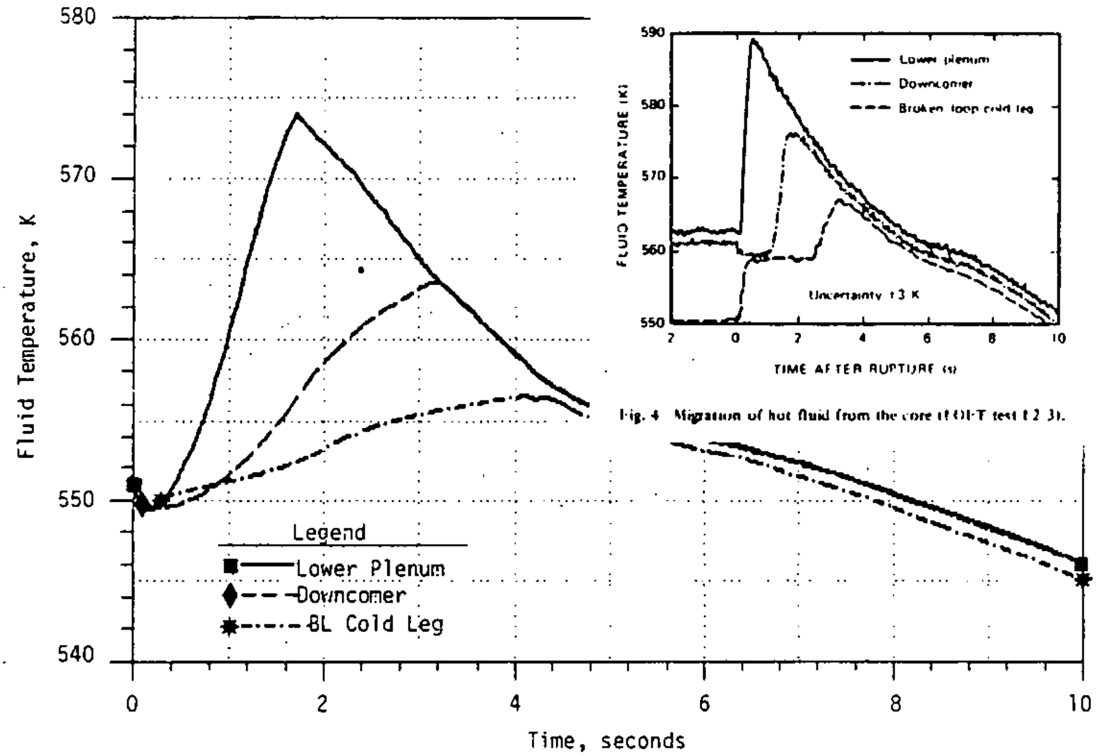


Figure 18 Migration of hot fluid from the core LOFT L2-3 WRAP blowdown analysis

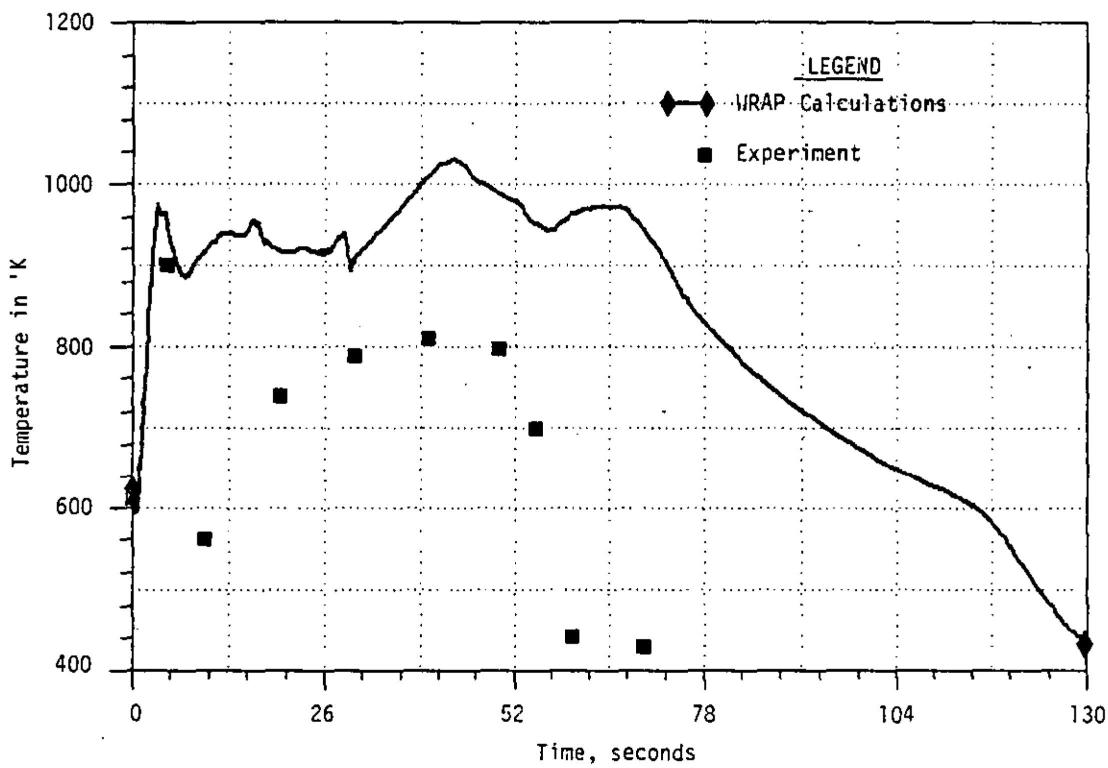


Figure 19 Fuel pin peak clad temperature (0.381 m)
LOFT L2-3 WRAP vs experiment

bottom of the core and the reflood portion of the transient begins. The duration of adiabatic heatup during the refill portion of the WRAP-EM calculation is a key factor in the conservatism of the LOFT L2-3 verification study. In the experiment the time between 30 and 40 sec is characterized by almost constant fuel clad temperatures in the core. An additional conservatism arises because WRAP-EM allows only a bottom up rewet. In the experiment the final quench occurred from the bottom up and the top down, with the hot rods in the central core region being rewet by 54 sec.

In Figure 20, WRAP and experimental results are compared to an INEL RELAP4/MOD6 BE calculation.[15] The latter has a substantially different reflood model than WRAP. The BE results fall between the WRAP and experimental results during refill and reflood.

4.2.4.4 ECCS Flows

For the emergency core cooling system (ECCS), the measured and calculated HPIS and LPIS flows agreed. The accumulator flow, however, was considerably different between WRAP and the INEL BE calculations and the experiment. The ECCS accumulator flow is shown in Figure 21. This figure shows the accumulator flow for the complete analysis (blowdown, refill, reflood) for WRAP, INEL BE, and LOFT L2-3 measurements.

For the experiment and both calculations, the accumulator initiates at about 15 sec. The effectiveness of the ECCS flow depends both upon the flow rate and the amount of water injected. The mass flow rate in the WRAP analysis is almost twice as large as either the measured rate or the INEL BE calculated flow rate.* The total mass of accumulator water which is emptied into the system in LOFT L2-3 is controlled by a trip of a valve connecting the accumulator to the system. The trip controlling the valve keys on the mixture level in the accumulator is designed to limit the total mass of accumulator water emptied into the system to 0.96 m^3. In the WRAP analysis, the trip occurs at 46.2 sec. This gives the same total amount of accumulator water injected in WRAP as in the INEL BE and the experiment, even though the flow rate is almost twice as high. In WRAP, the red water/blue water EM option[2] removes the ECCS water from the system prior to the refill calculation.

* The differences in calculated flow rate between WRAP and the INEL BE may be caused by the different nodalizations used for modeling the accumulator system in the two calculations. A more definitive answer would require additional calculations.

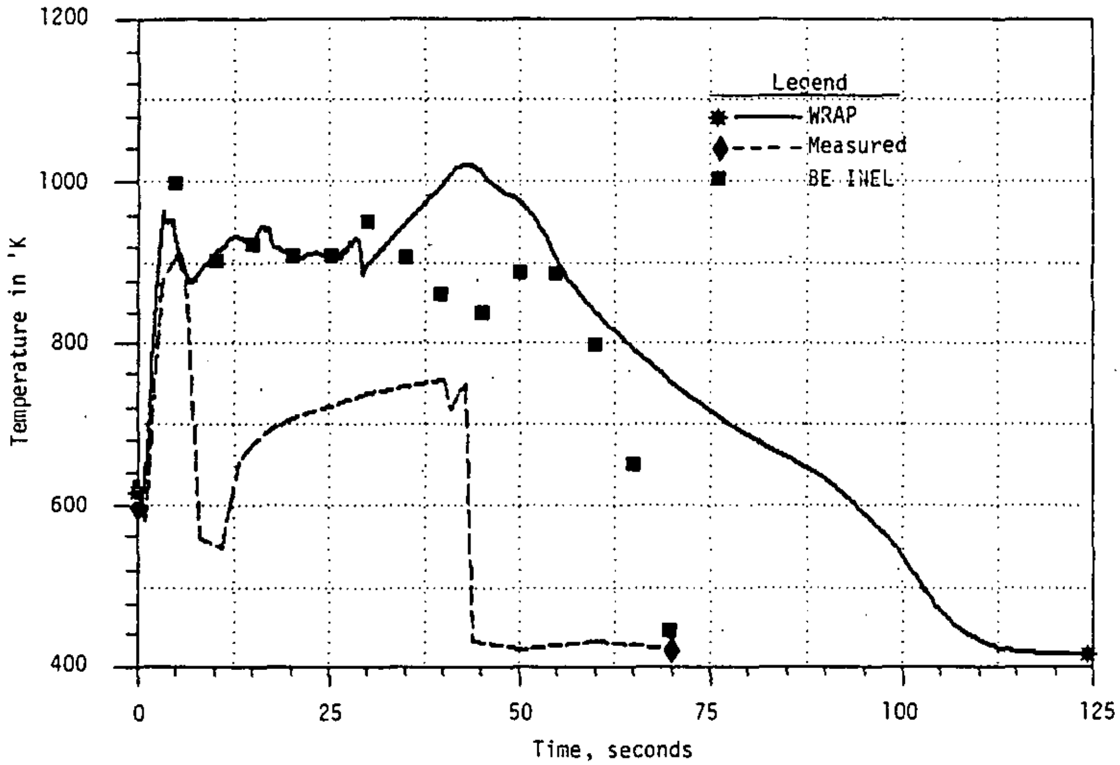


Figure 20 Fuel pin cladding temp @ 0.533 m elev
LOFT L2-3 WRAP vs experiment

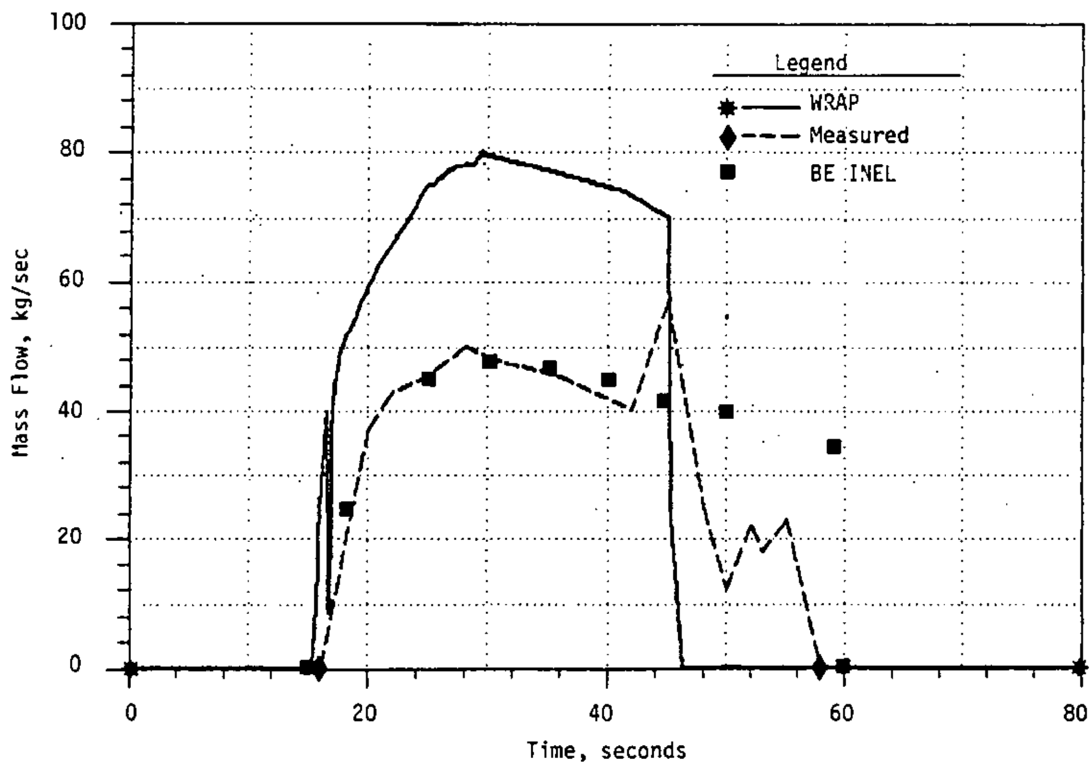


Figure 21 ECCS accumulator flow, WRAP vs measured
LOFT L2-3 WRAP blowdown analysis

The reflood analysis was terminated after the hottest heat slabs had quenched (90 sec of reflood transient time) because of the slow movement of the quench front being calculated by WRAP. The WRAP-FLOOD model allows quenching only after two requirements are met: the cladding temperature must be less than or equal to a defined quench temperature, and the quench front must be greater than halfway up the heat slab. At the time of termination of the reflood analysis, all of the core heat slabs were below the defined quench temperature, but the quench front had only reached heat Slab 4. The slow movement of the quench front is due in part to the small amount of ECCS water (only the LPIS and HPIS are still on, see Figure 22) being input to the system.

4.2.4.5 Uncertainties

The major uncertainties in this analysis are in the reflood analysis. The calculated accumulator flow, which is the dominant reflood flow source in the experiment, is cut off early in the reflood phase because the accumulator mass was depleted more quickly than in the experiment. During blowdown and refill, the calculated accumulator flow rate was much larger than the measured rate. Also a re-evaluation of the WRAP-EM reflood models is suggested because a calculation for a PWR power plant (discussed in the next subsection) has given a very conservative result. The models used in this study were updates of the original models in the INEL FLOOD code[6] with corrections for coding errors in the heat transfer correlation and modification in the FLECHT correlation to account for the shorter LOFT core.

4.3 Four-Loop PWR Parallel Calculation

4.3.1 Background

As part of the WRAP-PWR-EM verification program, a loss-of-coolant accident (LOCA) analysis for a typical Westinghouse four-loop PWR was performed in parallel at SRL and INEL. The identical input was used at both laboratories to model a double-ended, guillotine pipe break in the discharge line of a primary coolant pump. The plant was assumed to be operating at 3303 MW(th). One train of the ECCS was assumed inoperative. The SRL calculation used WRAP with its automated interfaces; the INEL calculation used a version of an established code, RELAP4, which has been modified to allow hand transfer of the same data between calculational phases.

The intent of this study was to verify that the code models and interfaces in the WRAP-PWR-EM system were functioning correctly by comparing WRAP calculations to calculations run by INEL using codes with the same models. To ensure a "clean" comparison, identical input data were to be used in both analyses. All phases

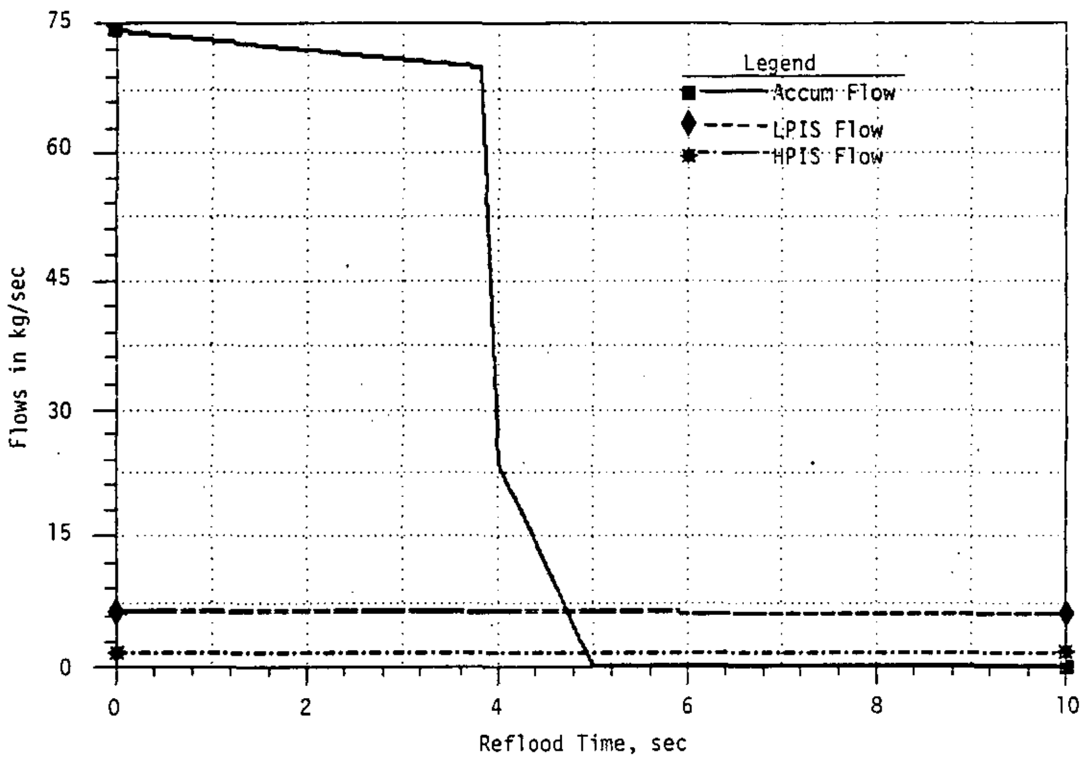


Figure 22 LOFT L2-3 WRAP reflood analysis emergency core coolant flows

of a large break LOCA (blowdown, refill, and reflood) in the four-loop PWR plant were to be analyzed. The initial steady state was computed by INEL, and the results were used by both SRL and INEL for their blowdown calculations. Most input data for the refill and reflood calculations were transferred automatically by WRAP interfaces. INEL transferred data by hand. Any input discrepancies were to be resolved by adjusting WRAP input, because the INEL calculations were run prior to the WRAP calculations.

Calculation of a hot pin (FRAP) was included in the WRAP analyses even though no INEL data were available for direct comparison.

4.3.2 Summary and Conclusions

In general, the two calculations agreed. All significant features of the PWR loss-of-coolant accident were the same in the two analyses. The minor variations observed were ascribed to differences in precision between the two computer systems (IBM and CDC) and the different timestepping strategies. The results verified that the WRAP models and interfaces are working as intended and that the WRAP-PWR-EM system can be used as an analysis tool.

4.3.3 Input

The initial input for the WRAP calculation was taken directly from Reference 16. The nodalizations for the blowdown and refill calculations are shown in Figures 23 and 24. Figure 25 shows the nodalization for the reflood calculation (FLOOD). The blowdown input was somewhat nonstandard because a dial of 0.6 was defined for the Moody critical flow model (while the discharge coefficients were set to 1.0), and no scram table as such was defined (the reactivity data were solely density and Doppler contributions).

The blowdown and refill analyses were set up as EM calculations by selecting:

- The EM heat transfer option (with the Dougall-Rohsenow correlation)
- The EM fuel pin swelling and flow blockage options
- The EM ECC bypass option
- The Henry-Fauske/Moody choked flow model for all junctions except Junctions 24, 27, 28, 29, 30, 32, 57, and 58 (see Figure 23) which were assigned the inertial flow model.

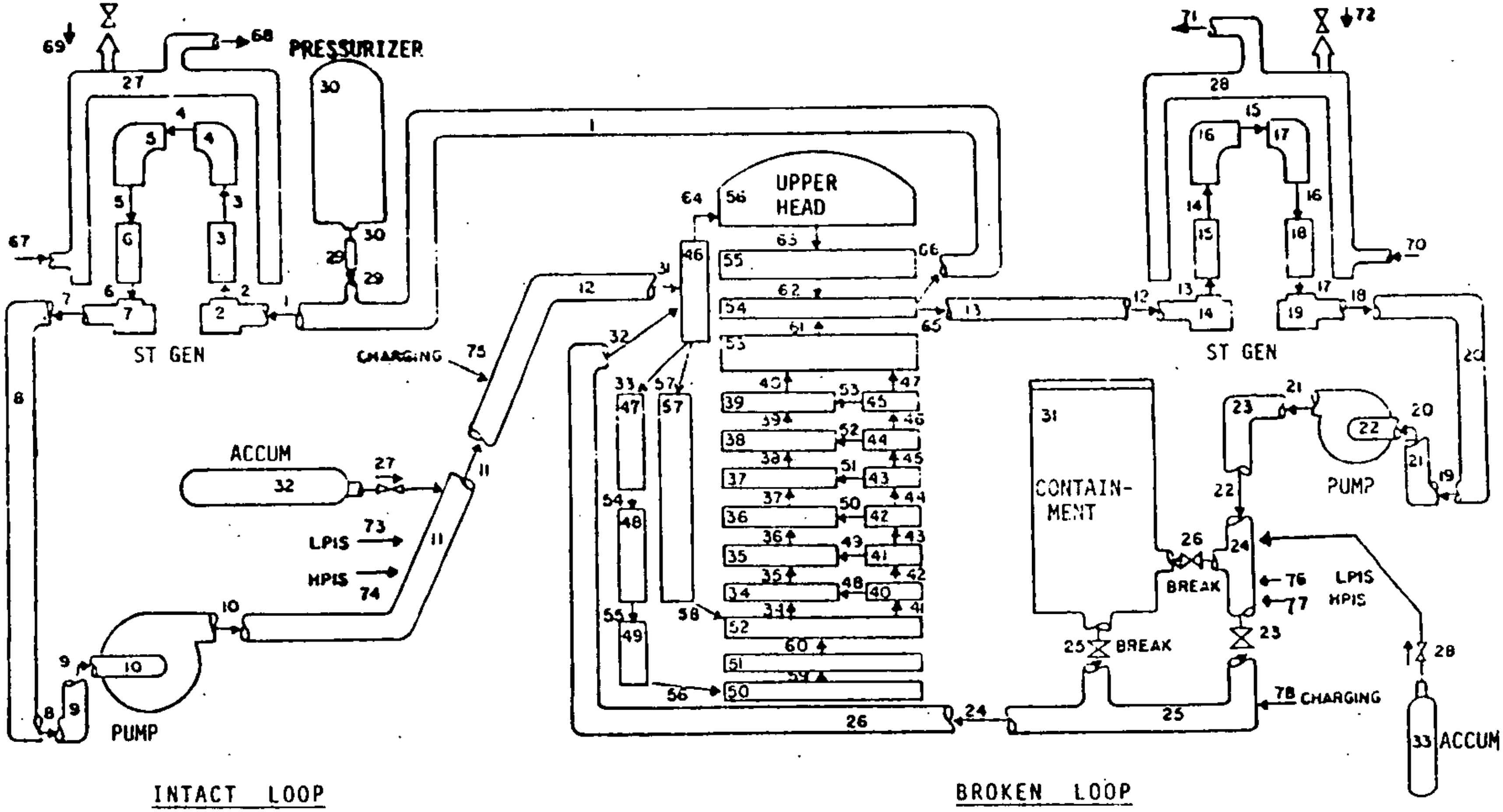


Figure 23 System nodalization diagram for Zion blowdown model

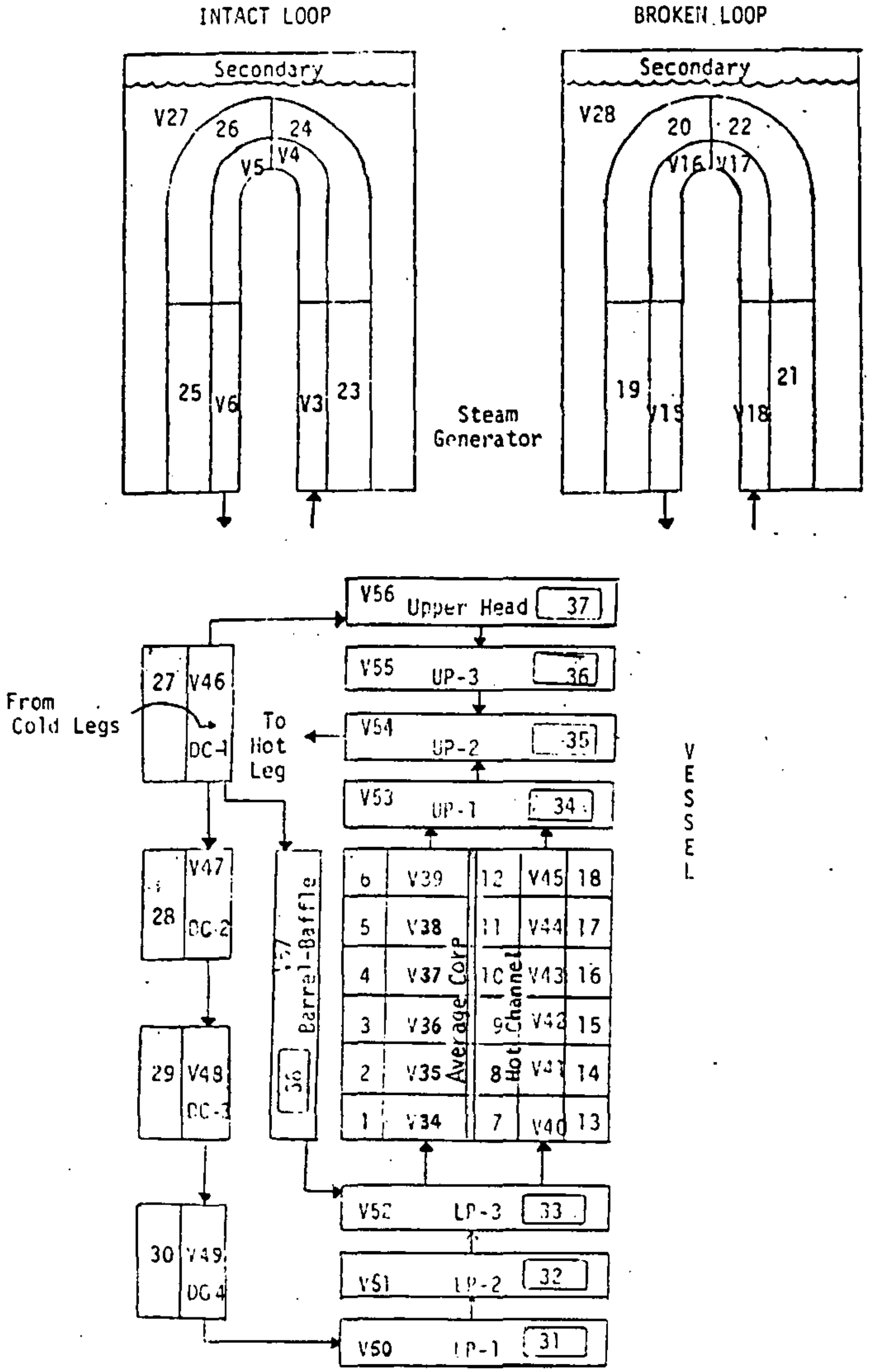


Figure 24 Heat slab nodalization diagram for Zion blowdown model

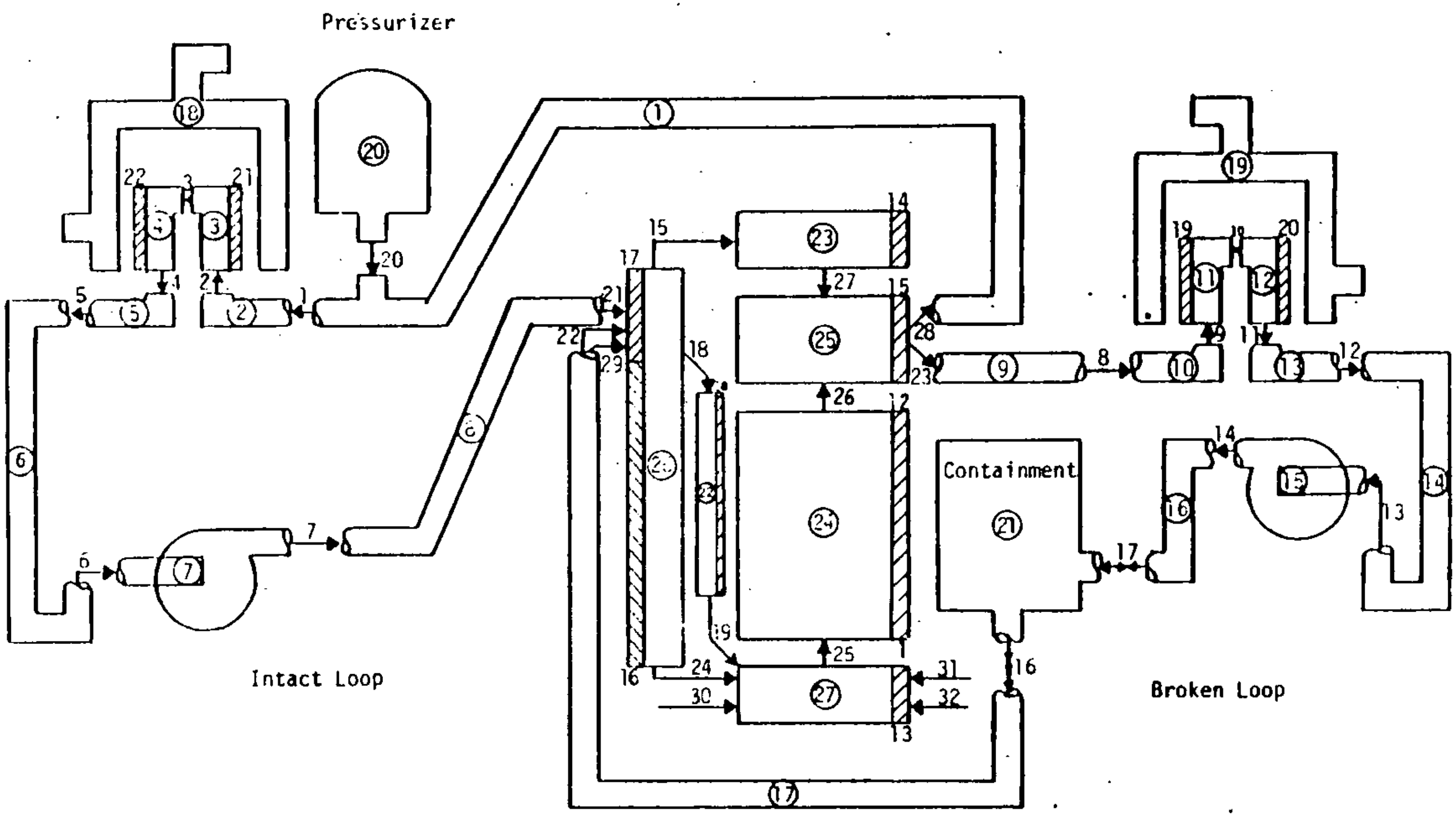


Figure 25 Nodalization diagram for Zion reflood model

The reflood analysis was set up as a standard FLOOD calculation by using the B&W entrainment correlation. As in the blowdown calculation, the Dougall-Rohsenow heat transfer correlation was selected for regions outside the core. The input used was identical to that defined by INEL (see Reference 16). Core reflood began with an initial power (from decay heat) of 163.17 MW(th). At this time, the ECC systems were injecting 4861 lbm/sec, primarily from the accumulators which had sufficient inventory for 11.65 more seconds of injection. Thereafter, the flow from the high and low pressure injection systems provided the source of reflood water.

First, the inertial flow model assigned to Junctions 24 and 32 (see Figure 23) was replaced by a critical flow model. Until this change was made instabilities caused the WRAP calculation to terminate. This change was found to have a negligible effect on the results. Second, changes were made in the correct refill and reflood data set up by the WRAP interfaces to match INEL input and thus maintain parallelism. INEL had set the enthalpy of the ECCS water to zero and had not specified adiabatic heatup during refill. As a result of these changes, there is an intentional discontinuity in the WRAP refill and reflood calculations.

The FRAP-T4-LACE code was used to analyze the thermal and mechanical behavior of the hot pin in the core. The transient data automatically transferred from TWRAM during the blowdown phase of the transient included the power history and the axially dependent temperature, pressure, enthalpy, and mass flow rate in the core. From these conditions, FRAP calculated the heat transfer coefficients at each axial level of the pin. The parameters passed from the FLOOD calculation were the flood rate, core inlet temperature, core pressure, and collapsed liquid level histories. During the refill period, adiabatic heatup is assumed so no data were passed from TWRAM.

All of the pin dimension, axial power profile, gas gap, and fuel rod plenum data were obtained from the Zion test problem created at INEL.[17] All the LACE options were selected except the power multipliers. Models used in the calculation were the Baker-Just cladding oxidation model, modified Ross-Stoute gap conductance model, the GAPCON-THERMAL-1 fuel deformation model, and the Dougall-Rohsenow post-CHF heat transfer correlation. The CHF correlations used are: 1) Hughes (Modified Barnett) for pressures less than 725 psia, 2) Barnett for pressures between 1000 and 1300 psia, and 3) Gellerstadt (B&W-2) for pressures greater than 1500 psia. At pressures intermediate to these limits, a combination of adjacent correlations is used.

For the FRAP calculation, the initial average fuel rod linear power density was 10.55 kW/ft. The peak power density was 14.76 kW/ft at an elevation of 6.0 ft. Additional input details are given in Appendix C.

4.3.4 Results

The blowdown phase lasted approximately 30 reactor seconds with a further 13 sec required to refill the lower plenum. The peak clad temperature occurred at 123 sec with quenching of the hottest slab beginning at 183 sec. The WRAP calculation was terminated at 198-reactor seconds. The CPU time required for the full calculation was 288 min for 61,722 timesteps on the IBM 360/195. Selected results are discussed in the following paragraphs and compared to INEL's results (taken directly from Reference 16) in Figures 26 through 35. The solid line represents INEL's WRAP-analogue EM calculation, the dashed line an earlier INEL EM calculation (the BE/EM study) shown to lend support to their WRAP-analogue results. The results of WRAP-EM are shown as circled data points overlaid on the INEL plots. As can be seen, the WRAP and INEL EM results agree. To illustrate the fine structure in the WRAP-EM calculation, some detailed WRAP plots are also included.

4.3.4.1 Core Inlet Flow

The core inlet flow computed by the two systems are plotted in Figures 26 through 28. It is clear that there are minor differences in the fine structure (some of which may be due simply to asynchronous plot times). In general, the flow behavior is the same in the two calculations. Both show the abrupt flow changes around five sec and eight sec. These dramatic reversals are possible initiators of the instability in the original formulation of the problem.

4.3.4.2 Cold Leg Break Flow

Comparing the WRAP results in Figure 29 to the INEL results in Figure 30 shows the two flows agree. All basic features in the behavior of the solution (e.g., occurrences of local maxima and minima) are the same in the two calculations. The barely visible mini-oscillations after five seconds in the INEL solution (in Figure 30) are noteworthy. These are probably a much reduced analogue to the oscillations in the original WRAP calculation. It is significant that these mini-oscillations are totally absent in Figure 29 (the final WRAP solution), even with the expanded y-axis in that figure. The implication is that replacing the inertial flow model for Junctions 24 and 32 with a critical flow model has removed these oscillations. The flat portion of the WRAP plot

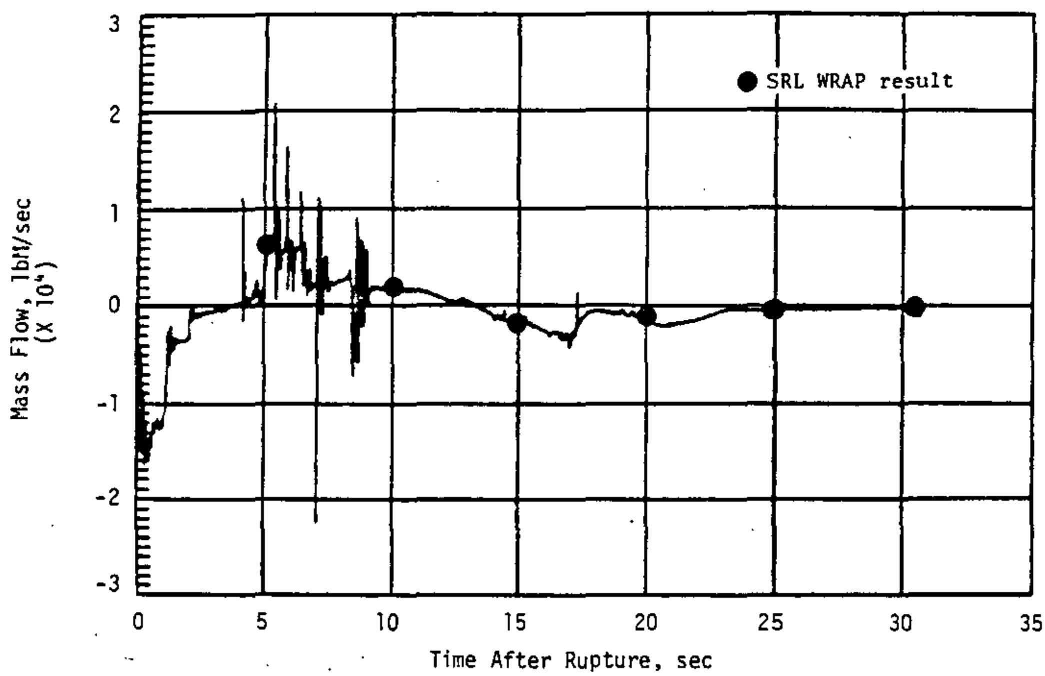


Figure 26 Core inlet flow (avg channel - Junction 34)
Zion blowdown

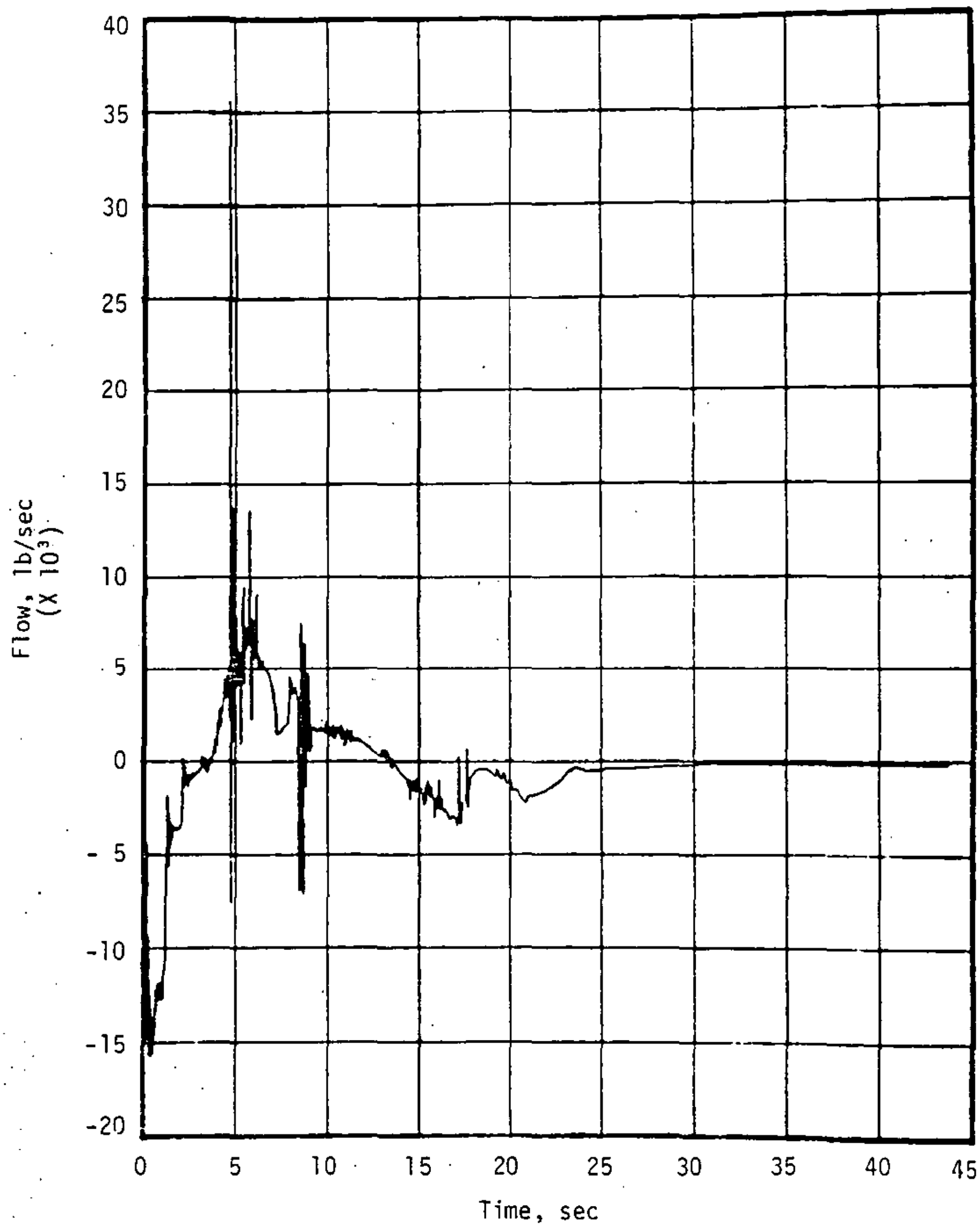


Figure 27 Core inlet flow (Junction 34)
WRAP solution

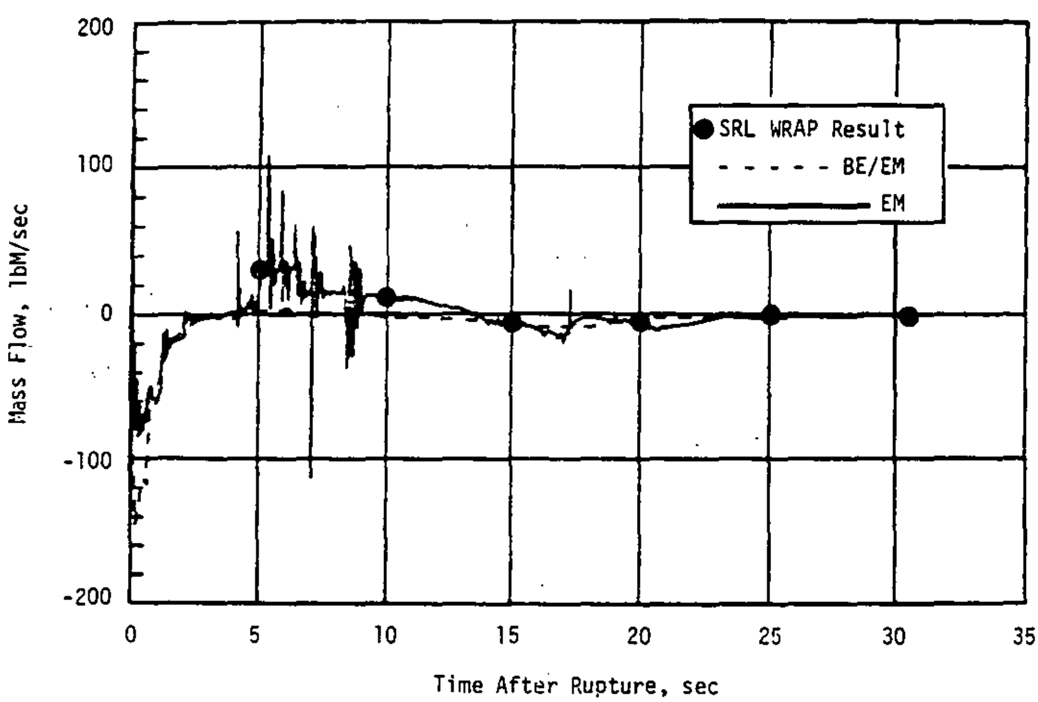


Figure 28 Core inlet flow (hot channel - Junction 41)
Zion blowdown

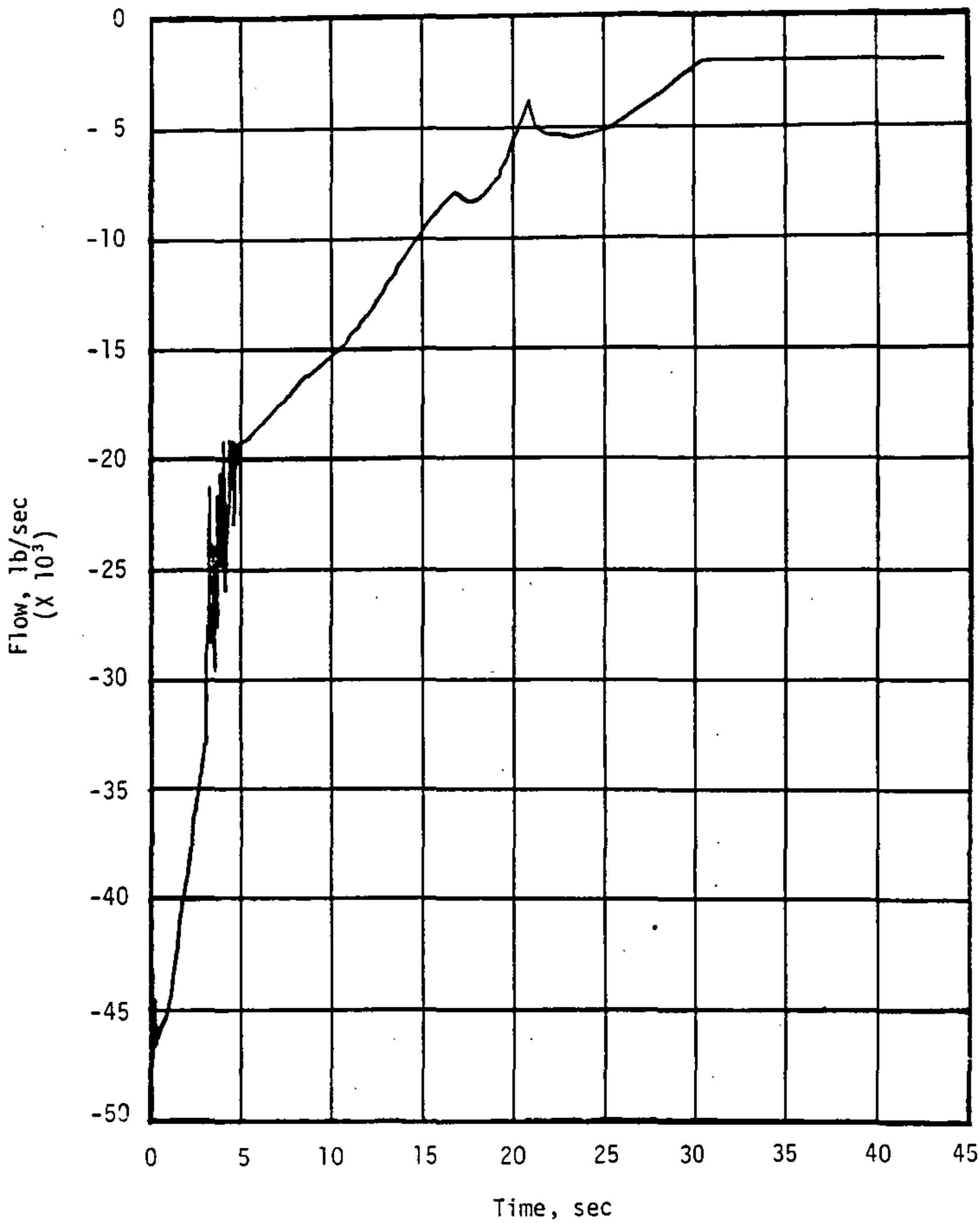


Figure 29 Cold leg break flow (Junction 25)
WRAP solution

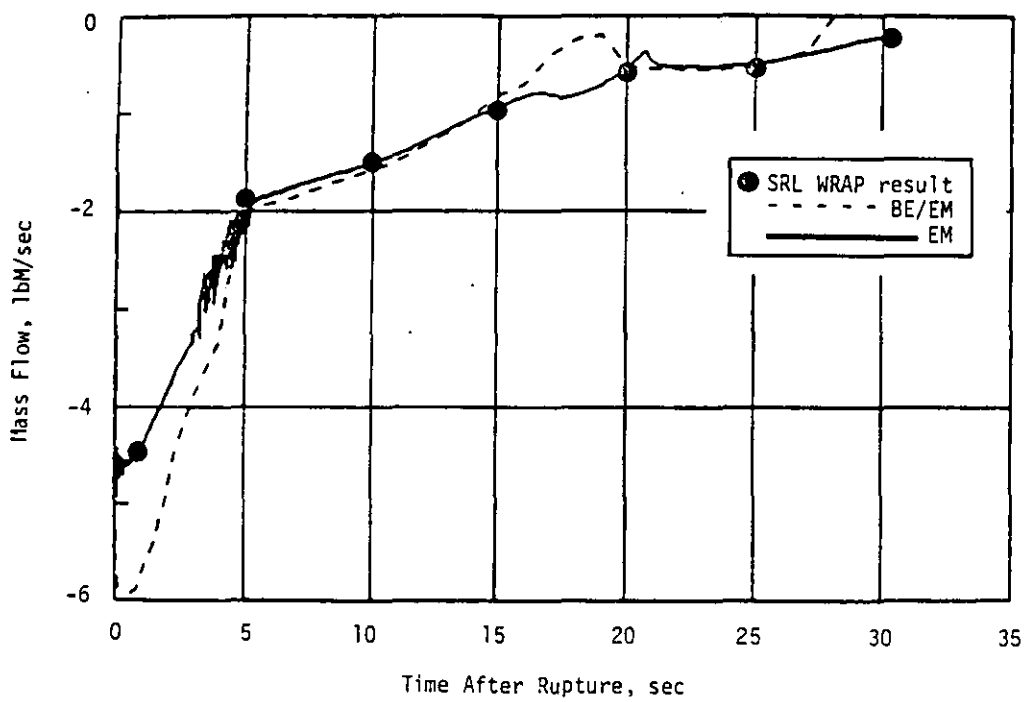


Figure 30 Cold leg break flow (Junction 25)
Zion blowdown

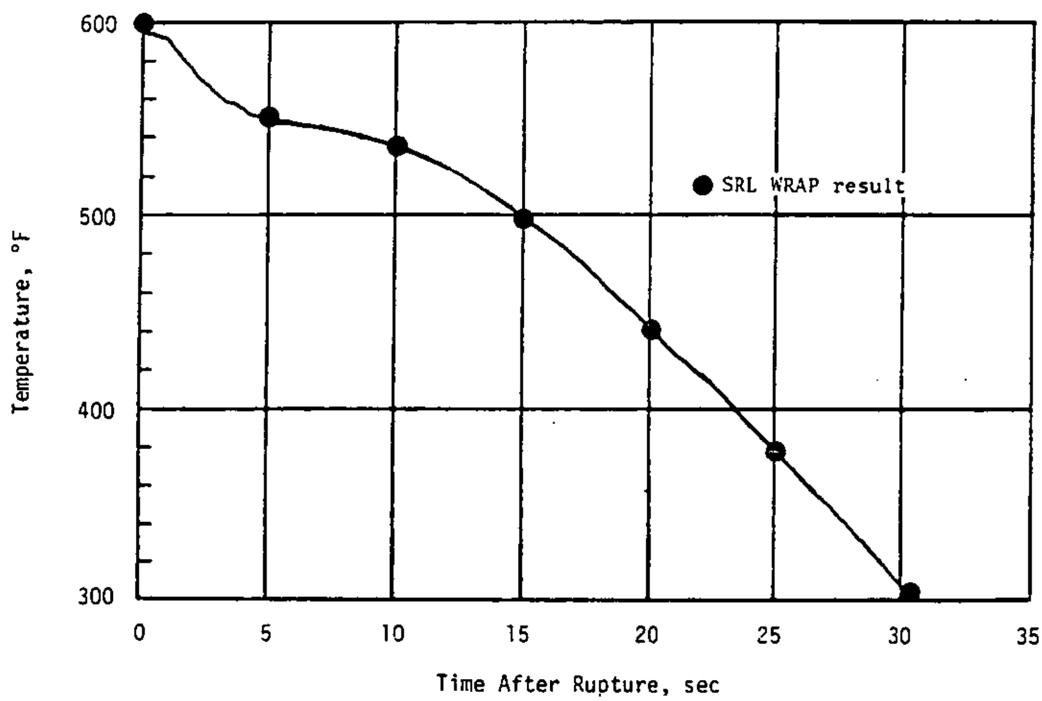


Figure 31 Upper plenum pressure (Vol 54)
Zion blowdown

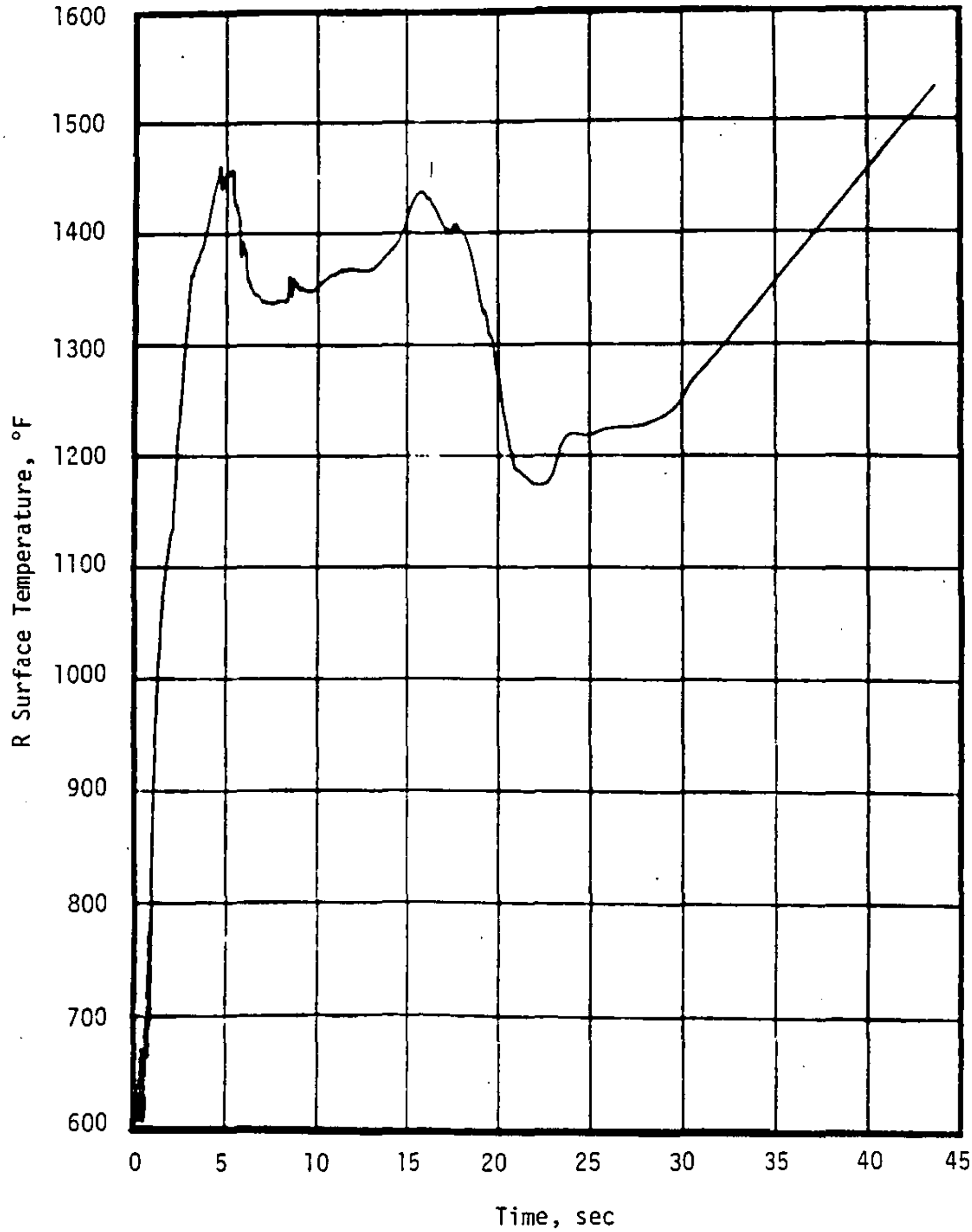


Figure 32 Core heat slab surface temperature (Slab 9)
WRAP solution

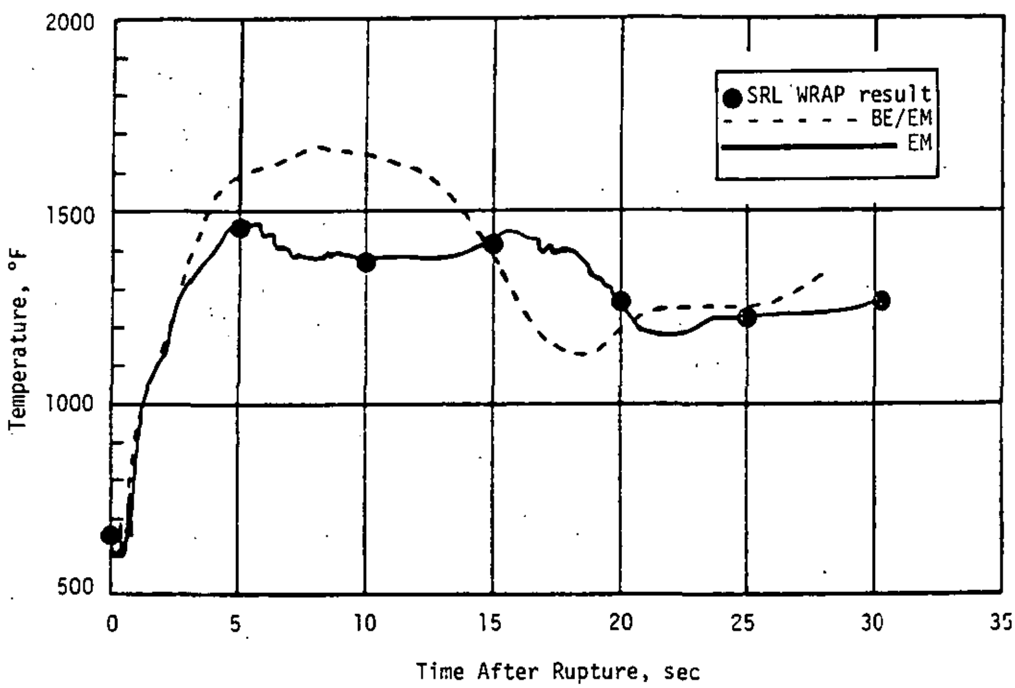


Figure 33 Hot assembly surface temperature
(Slab 9) Zion blowdown

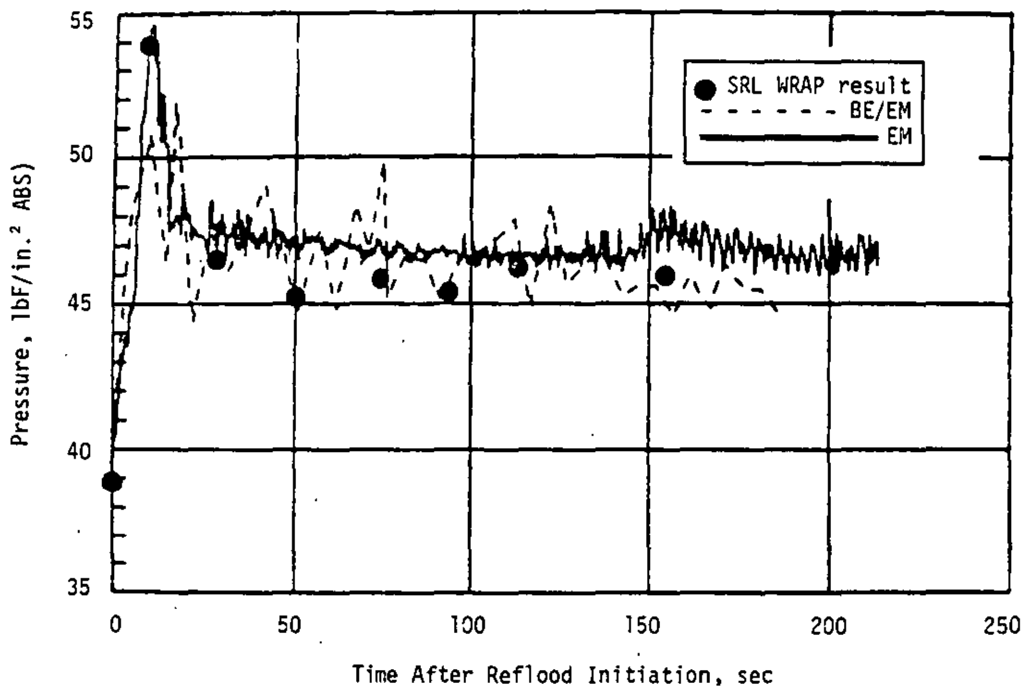


Figure 34 Upper plenum pressure (Volume 25)
Zion reflood

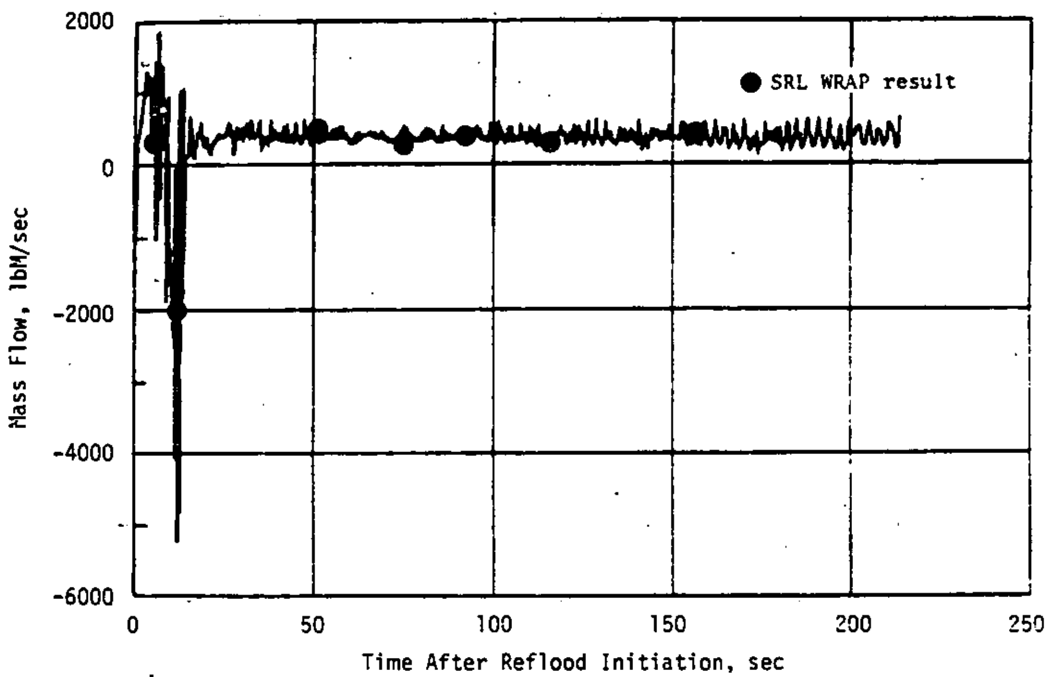


Figure 35 Core inlet flow (Junction 25)
Zion reflood

beyond 30 sec is an artifact of the system: it represents the adiabatic heatup phase of refill during which all parameters not involved in the slab heatup are held constant.

4.3.4.3 Pressure in Upper Plenum (Blowdown)

The pressure in the upper plenum as calculated by WRAP and INEL is shown in Figure 31. The initial pressure of 2262 psi is obscured by the y-axis. In general, the two blowdowns follow the same path.

4.3.4.4 Slab Surface Temperature (Blowdown)

The behavior of the slab surface temperature for a typical slab as calculated by WRAP is shown in Figure 32; the INEL results are shown in Figure 33. Prior to the end of bypass, that slab's maximum surface temperature occurs at approximately five sec. The value calculated by WRAP (about 1450°F) is approximately 20°F lower than that calculated by INEL. This difference may be ascribed to the small differences in the fine structure in the core inlet flow between the two cases (compare Figures 26 and 27). The decrease in surface temperature beyond the five sec point is consistent with the increase in core flow during this interval. Similarly, the increase in surface temperature up to 16 sec is consistent with the decrease in core flow just before that time. Unlike the other WRAP plots, the portion of Figure 32 beyond 30 sec has significance: it illustrates the adiabatic heatup of the slab surface during refill (this phase is discussed in the next section). The maximum surface temperature during the blowdown and refill phases calculated by WRAP was 1611°F, occurring for Slab 15 (see Figure 24) at the end of adiabatic heatup.

4.3.4.5 Refill Results

The WRAP refill results presented in Table 6 were computed with the purposely erroneous ECCS enthalpy. With the exception of the maximum surface temperature, the two refill calculations agree. INEL computed a much lower maximum slab surface temperature at end of refill than SRL (1481°F vs 1611°F) because they had not specified adiabatic heatup during the refill phase. This was incorrect because a basic assumption in the EM model for the refill phase is that the core slabs heat up adiabatically (no heat transfer across the surfaces) from end of bypass to the end of refill. In the WRAP calculation, the adiabatic heatup results in the monotonic temperature rise beginning at about 30 sec (see Figure 32).

Table 6 Comparison of refill quantities for PWR four-loop plant

	WRAP	INEL
End of bypass (EOB)	30.447 sec	30.265 sec
LP pressure at EOB	70.673 psi	69.612 psi
Max surf temp at EOB	1335°F	1343°F
Time to refill LP (after penetration)	11.15 sec	11.15 sec
LP penetration delay	2.10 sec	2.10 sec
ECCS mass injected	69275.6 lb	69275.8 lb
Liquid temp in LP at BOCREC	134.6°F	134.3°F
Time to empy N_2 from accum	5.32 sec	5.32 sec
Max surf temp at end of adiabatic heatup	1611°F	1481°F

Note: LP = lower plenum

The next set of figures (34 through 40) describe the reflood phase of the analysis. Although the data begin at time zero, that point is actually 43-reactor seconds into the transient.

4.3.4.6 Pressure in Upper Plenum (Reflood)

The pressure in the upper plenum is shown in Figure 34. Both WRAP and the INEL calculation show an initial pressurization of the upper plenum, due to entrained water being carried into the upper plenum, coming into contact with the hot walls there and turning to steam. The peak pressure calculated by WRAP is approximately 1.5 psi higher than the INEL result. Beyond the 20-sec point, the pressure stabilizes around 45 psia in the WRAP calculation and 47 psia in the INEL calculation.

4.3.4.7 Core Inlet Flow

The core inlet flows are shown in Figure 35. Significant flow oscillations occur in the first 20 sec. This is possibly due to the initial flow distribution not being in precise balance. During this interval of severe flow oscillations, steam table failures were encountered; it was necessary to restart the problem with reduced timesteps to continue the WRAP calculation. In addition, water packing was encountered in the lower plenum, another possible sympton of an unbalanced initial flow distribution.

4.3.4.8 Core Outlet Enthalpy

The core outlet enthalpy is shown in Figure 36. There are noticeable differences in the INEL and WRAP results. The initial behavior up to 10 sec is the same: a very rapid drop, then a steep rise. The core outlet enthalpy is quite sensitive to the core inlet flow, thus the differences in the two oscillatory flow histories before 20 sec lead to very different enthalpies.

4.3.4.9 Core Mixture Level

Plots of the core mixture level are shown in Figures 37 and 38. Again, the basic behavior of the data is consistent; however, some of the details are different. Beyond 20 sec, the mixture level calculated by WRAP recovers to an initial value about 2 ft higher than the INEL result. This is probably due to differences in the timestep selection. It was found that varying the WRAP timestep size in the oscillatory phase of the calculation could shift this recovery mixture level position approximately one ft. The timestep sizes used in the INEL calculation were unspecified in Reference 16.

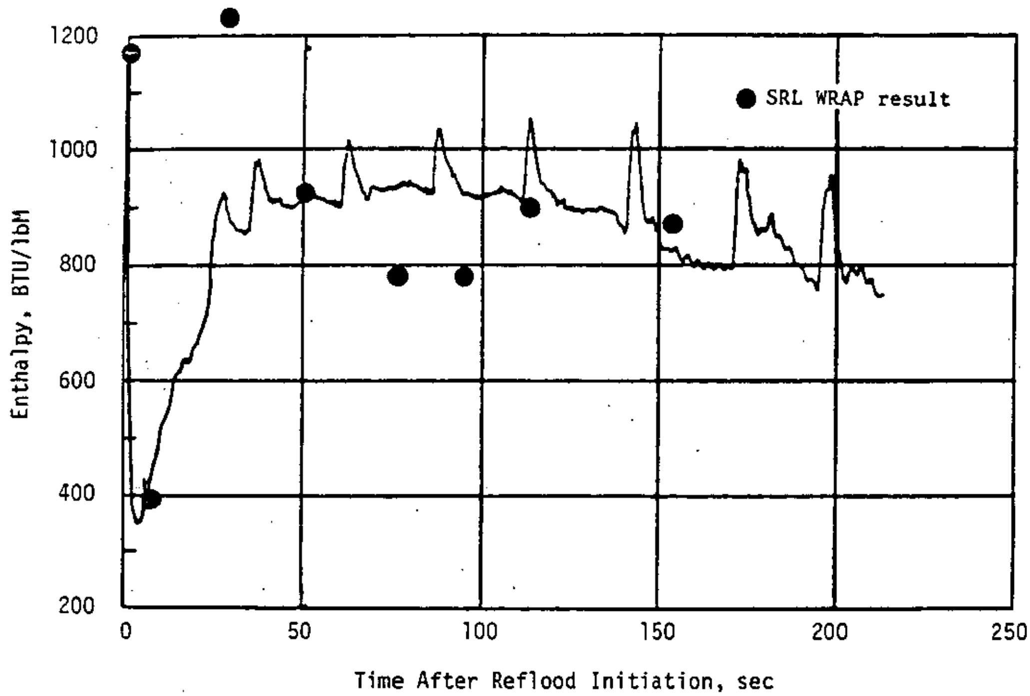


Figure 36 Core outlet enthalpy (Junction 26)
Zion reflood

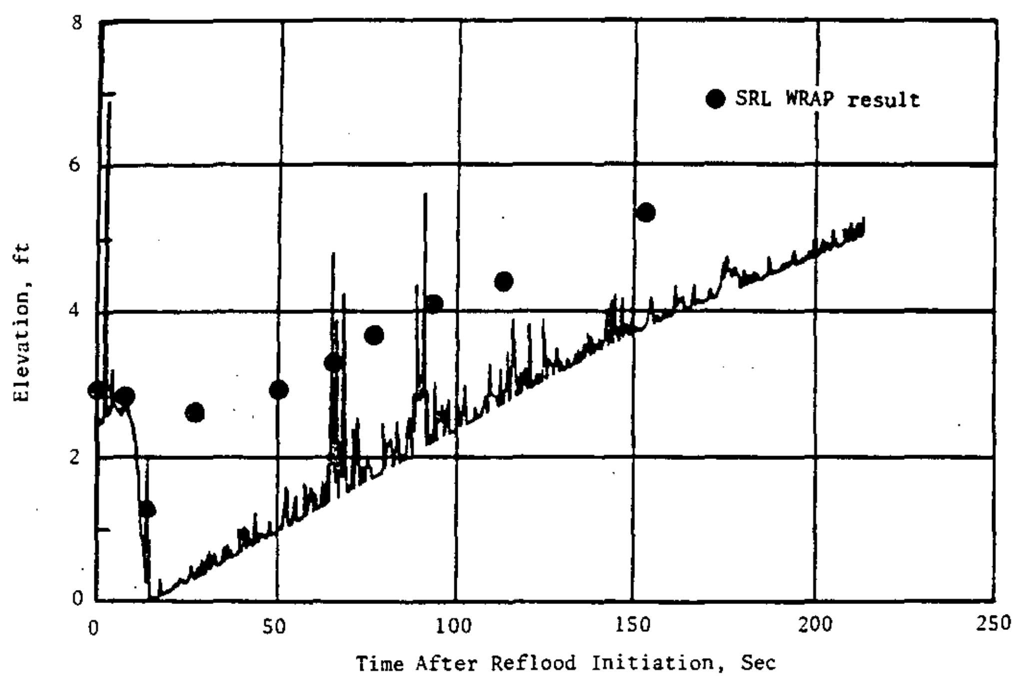


Figure 37 Core mixture level (Vol 24)
Zion reflood

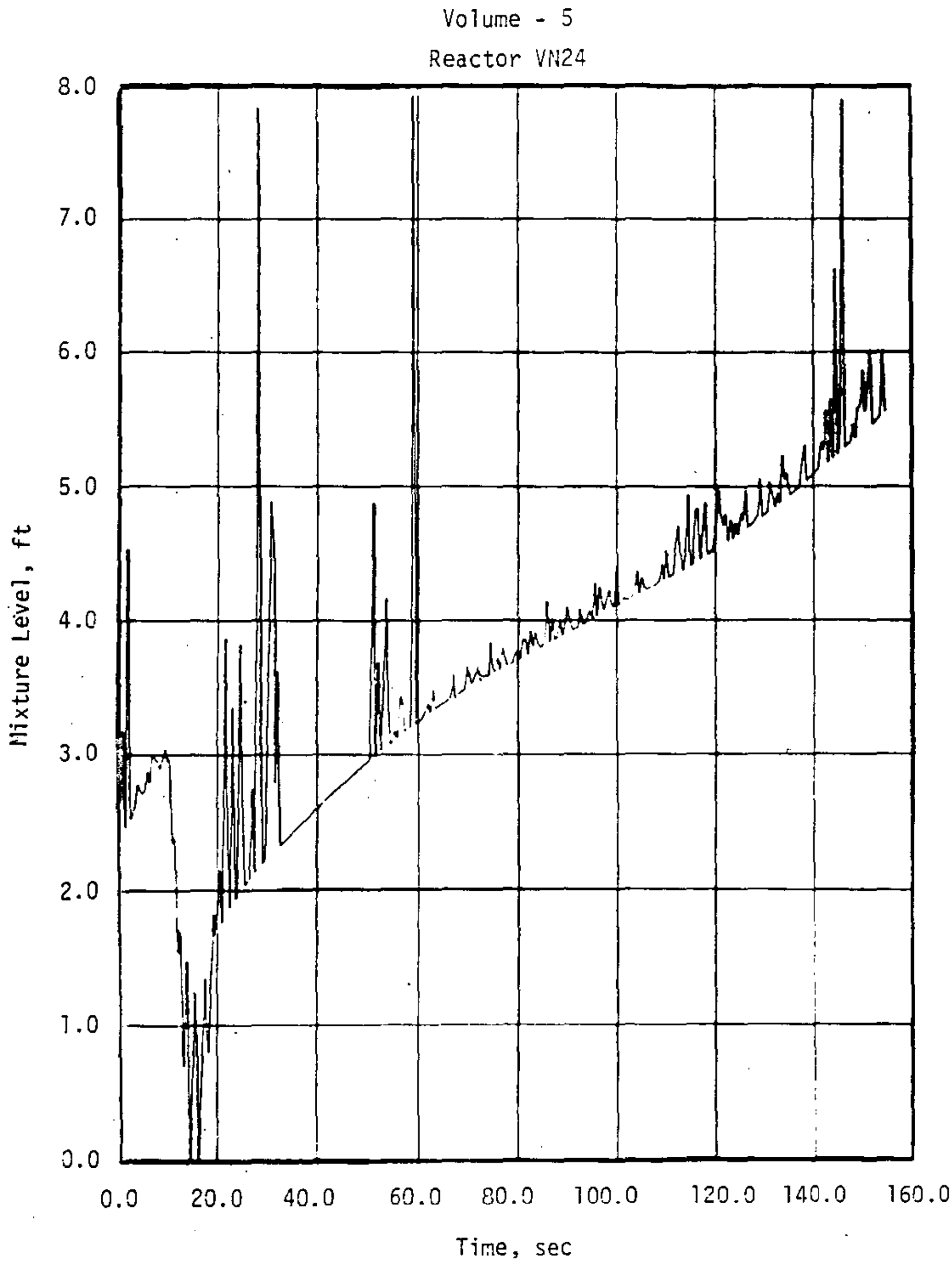


Figure 38 Core mixture level during reflood WRAP solution

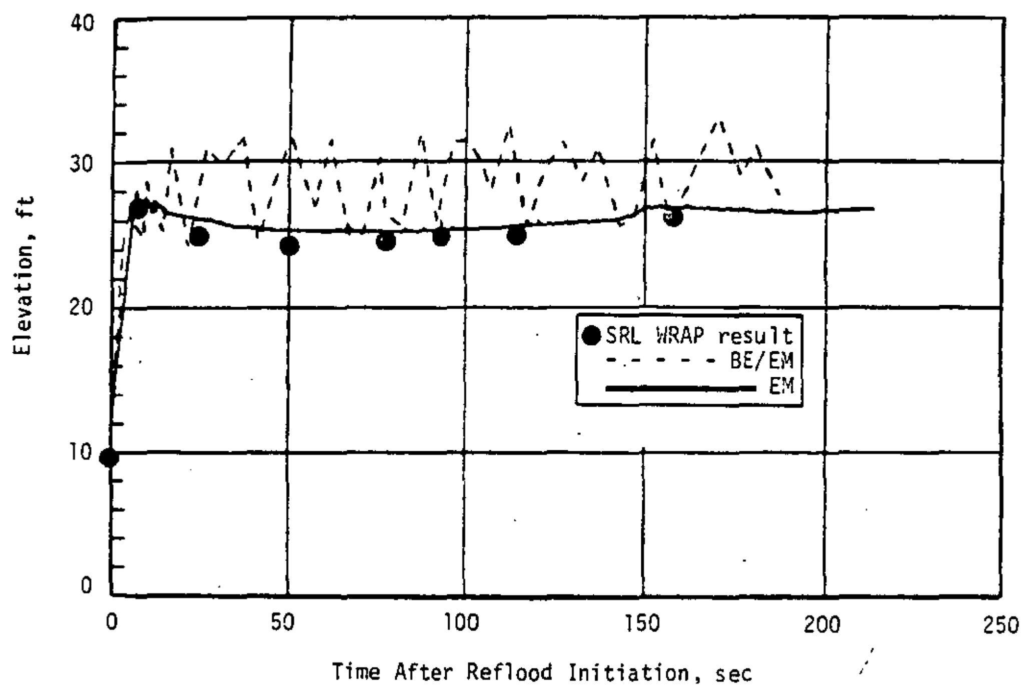


Figure 39 Downcomer mixture level (Vol 26)
Zion reflood

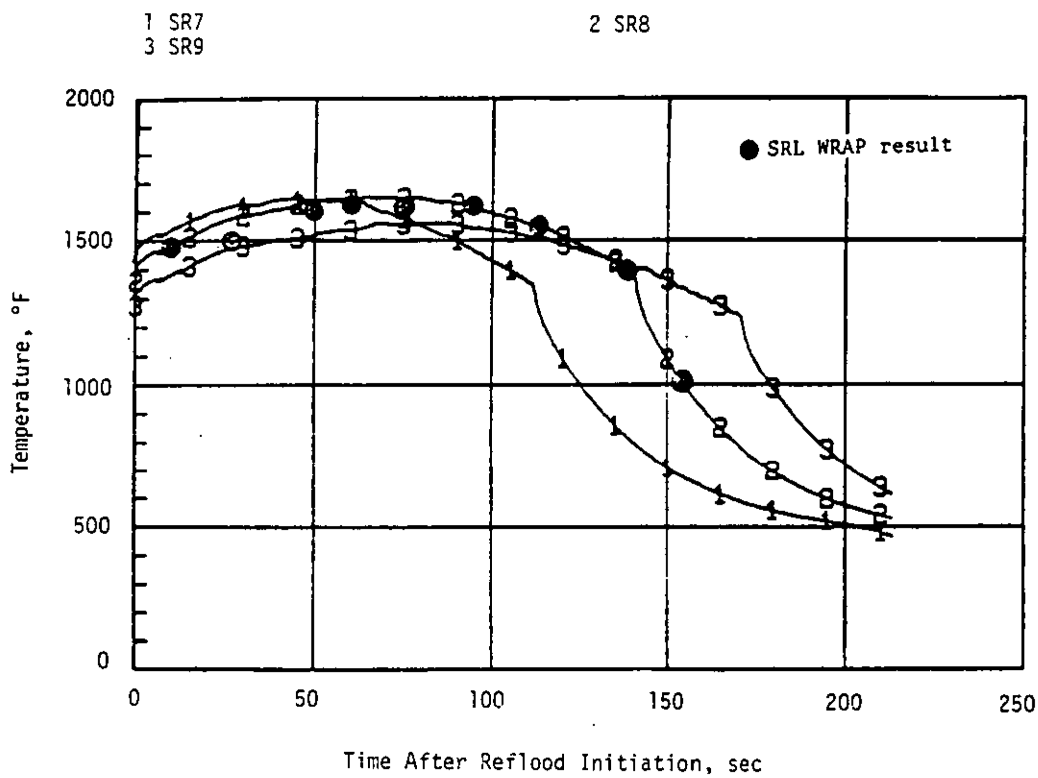


Figure 40 Rod surface temperature (Slab SR--)
Zion reflood

The spikes in the core outlet enthalpy curves coincide with spikes in the core mixture level curves. As the slug of liquid comes into contact with hotter slabs, a momentary increase in the energy content of the liquid leaving the top of the core occurs. The WRAP calculation predicts the mixture level to finally enter the heated core (without further collapses) beyond 60 sec. (The heated portion of the core extends from 3.323 to 15.323 ft.) This is consistent with the "noise" which is superimposed on all the curves beyond this time: it reflects steam generation as the coolant successively wets more slab surfaces. Each "puff" of steam causes a momentary pressure increase, which reduces the core flow rate and the steam production: hence the "noise."

4.3.4.10 Downcomer Mixture

The downcomer mixture level is plotted in Figure 39. Both calculations show very similar behavior. The WRAP mixture level in the downcomer remains slightly lower than the INEL result, while in the core, the opposite behavior is observed. This is consistent with the slightly reduced core pressure calculated by WRAP: the downcomer, lower plenum, and core can be thought of as a U-tube, with the level positions determined by the pressures above each liquid surface (the total volume of liquid and rates of liquid addition are the same in both cases).

4.3.4.11 Slab Surface Temperature (Reflood)

The behavior of the surface temperature of Slab 8 (at the elevation of 7.0-8.0 ft in the heated core) is shown in Figures 40 and 41. This is the slab whose surface attains the peak clad temperature. In the first 20 sec of the WRAP calculation, there is some extra structure in the result compared to the INEL calculation, probably due to the larger amplitude flow oscillations in the WRAP calculation. The peak clad temperature calculated by WRAP is 1610°F compared to the 1650°F calculated by INEL. This result is consistent with WRAP's prediction of a higher core mixture level (i.e., better cooling). This difference may be ascribed to the different core inlet flow history in the WRAP calculation which, in turn, was probably caused by a different timestepping strategy. Both calculations show the peak clad temperature occurring at 80 sec with quenching beginning at 140 sec. It should be noted that in both calculations the initial slab temperatures were artificially lower because of the INEL error in the adiabatic heatup phase of the refill. Thus, the peak clad temperature in the "correct" calculation would be of the order of 150°F higher than that shown in Figures 40 and 41. The WRAP calculation was terminated at 155 sec because the hot plane had already quenched.

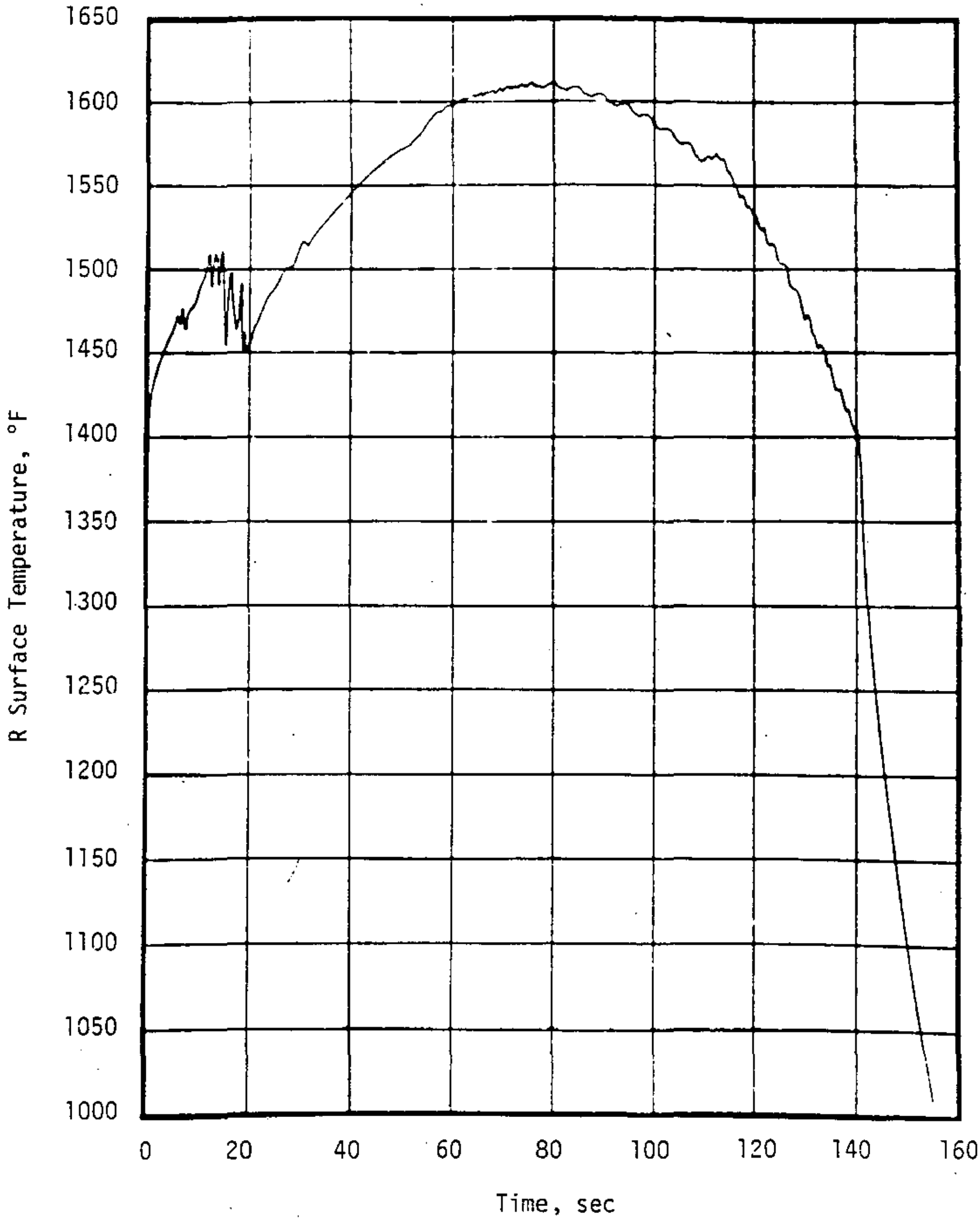


Figure 41 Surface temperature Slab 8 during reflood WRAP solution

The final set of figures (42 and 43) describe the hot pin analysis. This phase of the calculation was not performed by INEL, thus there are no parallel results to compare.

4.3.4.12 Clad Surface Temperature (Hot Pin)

Figure 42 shows the behavior of the clad surface temperature at an elevation of 8.4 ft. This node attained the highest peak clad temperature during the transient. The initial temperature response follows the coolant flow rate (Figure 27) with an immediate rise due to flow reversal and a temperature decrease as positive flow returns. The temperature again climbs since the flow is low and reaches a local maximum at 18 sec as the core quality reaches 1.0. A subsequent reduction in core quality provides better heat transfer thus reducing the temperature again until the fluid in the core is nearly depleted. At 30 sec, end of bypass signals the beginning of the refill period and adiabatic heating. At 43 sec, the water level reaches the bottom of the core, and the reflood portion of the transient initiates. The temperature continues to increase as the quench front rises through the lower regions of the core. The FRAP calculation was terminated at about 150 sec, the hot plane quenching time in the FLOOD calculation. The clad temperature is 2300°F at this time. [Note that the difference between this temperature and that computed in the FLOOD calculation (Figure 41) is primarily due to the fact that the FLOOD calculation modeled "core average" fuel conditions; whereas, the hottest pin is modeled in FRAP.] If the calculation had continued, the temperature in Figure 42 would have risen until the quench front reached the 8.4-ft elevation.

4.3.4.13 Clad Hoop Strain

As indicated by the clad hoop strain behavior shown in Figure 43 the clad ruptured at 37 sec at an elevation of 6 feet. This elevation corresponds to the highest power level in the pin.

The rather extreme behavior exhibited in Figures 42 and 43 (i.e., clad surface temperature in excess of 2300°F, clad rupture at 37 sec) has led to a review of the FLECHT correlation in WRAP/FLOOD and the planned implementation of an alternate steam cooling model in FRAP.

4.3.4.14 Uncertainties

Unlike the other two analyses, the four-loop PWR parallel calculation represents an analytical comparison of two independent calculations using essentially identical input. Thus, the main uncertainties are associated with the effect of running the codes on two different systems, an IBM 360/195 and a CDC 7600 (whose precision

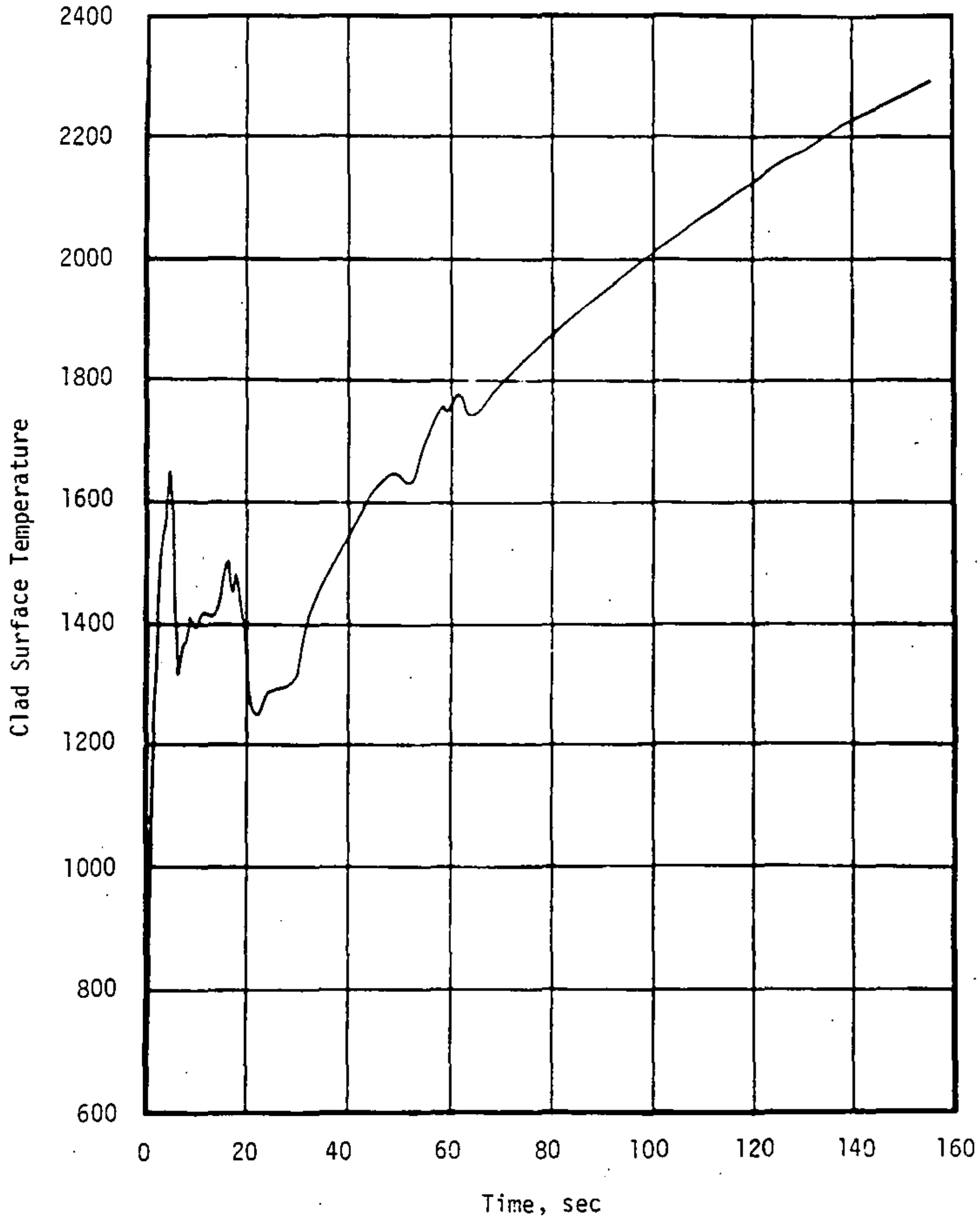


Figure 42 Hot pin clad surface temperature at 8.4 Ft elevation

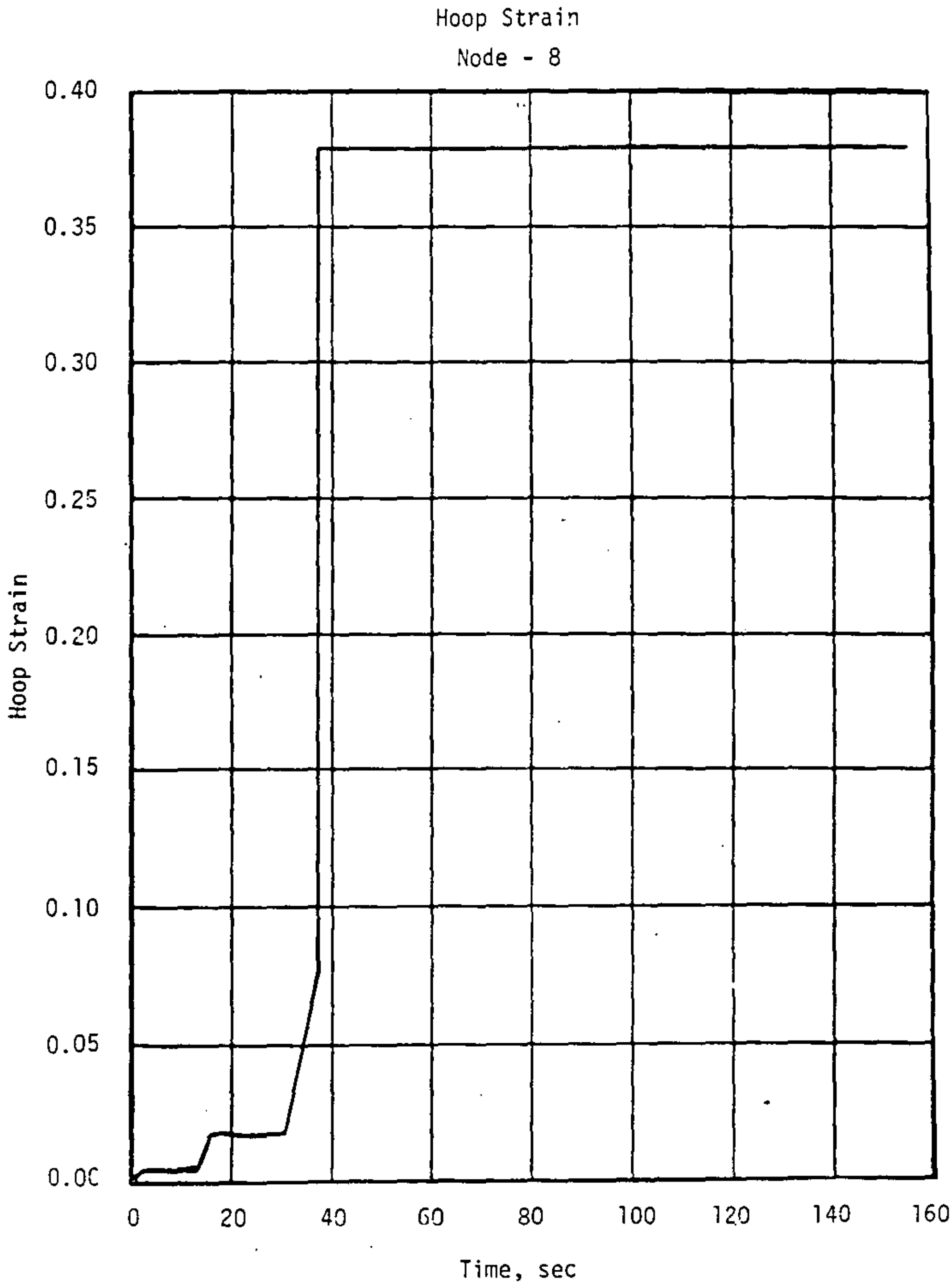


Figure 43 Hot pin clad hoop strain at 6 Ft elevation

differences can cause the automatic timestep selection algorithms to behave differently) and the effect of selecting a slightly different inertial flow model for the WRAP calculation. Otherwise, the two calculations are formally identical.

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APPENDIX A

WRAP INPUT DATA FOR CALCULATION OF SEMISCALE TEST S-06-3

The timestep selection criteria used in the Semiscale calculation are given in Table A-1. The normalized power function used in modeling the test power history is given in Table A-2, with the axial power shape shown in Figure A-1. The ECCS flow rates are given in Table A-3, and the accumulator injection rate during reflood is given in Table A-4.

Table A-1 Semiscale: timestep selection

Blowdown

Timestep size (sec) Min	Max	Time (sec) of Interval End
5×10^{-6}	5×10^{-3}	0.5
1×10^{-3}	1×10^{-2}	1.0
5×10^{-5}	5×10^{-3}	3.0
5×10^{-5}	1×10^{-3}	5.0
1×10^{-5}	1×10^{-2}	11.0
1×10^{-5}	1×10^{-3}	23.0
2×10^{-5}	5×10^{-3}	26.0
2×10^{-5}	1×10^{-3}	90.0

Reflood

Timestep Size (sec) Min	Max	Time (sec) of Interval End
1×10^{-4}	1×10^{-2}	1.5
1×10^{-5}	1×10^{-3}	2.5
1×10^{-5}	1×10^{-4}	2.6
1×10^{-5}	5×10^{-4}	2.8
1×10^{-5}	1×10^{-3}	3.1
1×10^{-5}	5×10^{-4}	6.0
1×10^{-5}	1×10^{-3}	60.0

Table A-2 Semiscale: normalized power (blowdown)

Time (sec)	Normalized Power
0.0	1.0
1.2	1.0
1.8	0.5277
2.5	0.5277
3.0	1.0
5.0	1.0
6.0	0.4398
7.2	0.1759
11.0	0.1759
11.5	0.1055
15.0	0.1055
17.0	0.00704
17.5	0.1407
19.5	0.1407
20.0	0.044
26.5	0.044
27.0	0.0176
35.0	0.0176
40.0	0.004
100.0	0.0035
Normalized Power (Reflood)	
0.0	1.0
59.6	0.87573

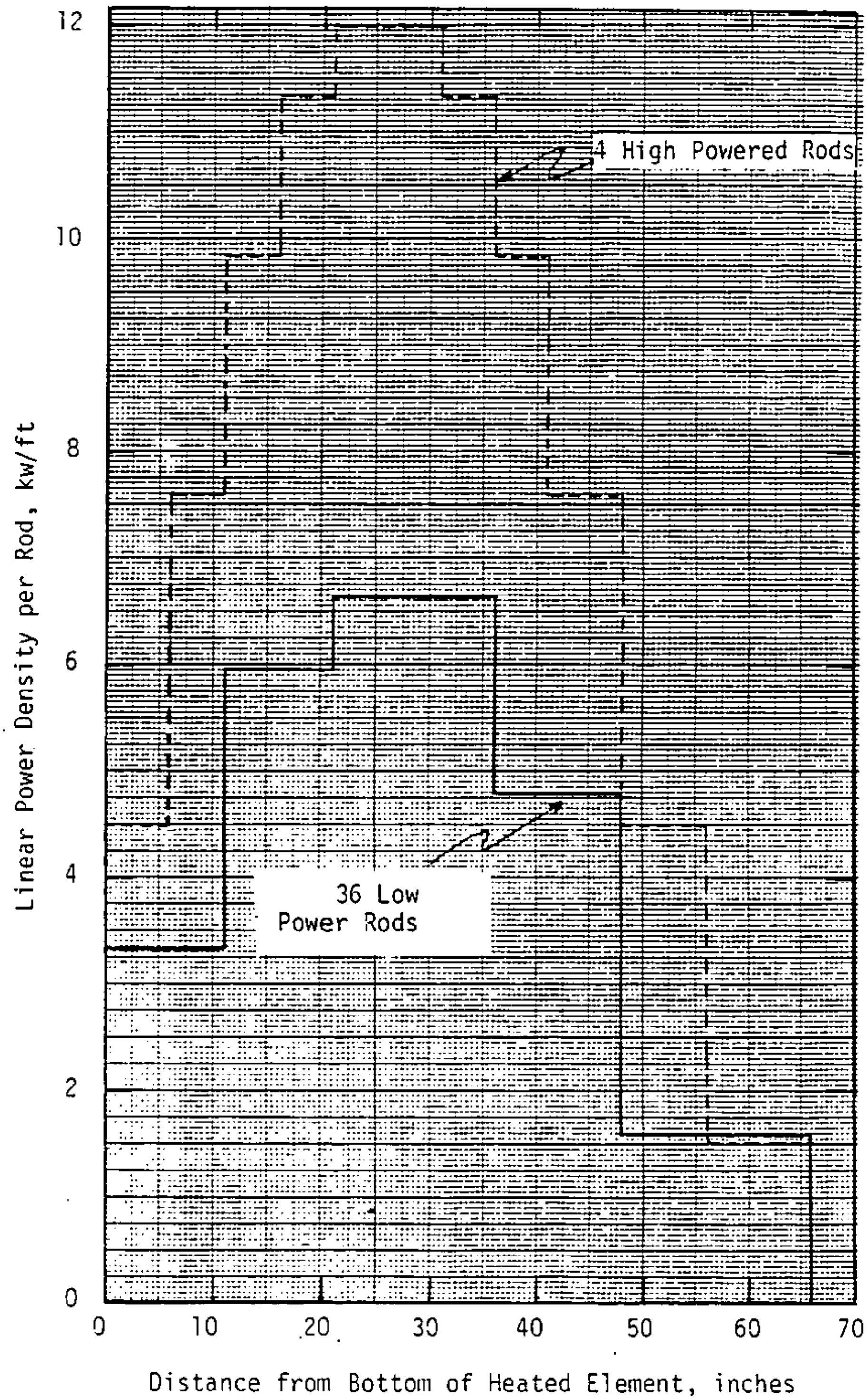

Figure A-1 Axial power distributions for core heat slabs

Table A-3 Semiscale: ECCS Flows

Low pressure injection*	High pressure injection*
941.844 gal/min - ft^2	154.308 gal/min - ft^2

*Area = 4.99×10^{-3} ft^2

Table A-4 Semiscale: accumulator injection rate (reflood)

Time (sec)	Flow (lbm/sec - ft^2)*
0.0	752.966
2.0	736.092
4.0	720.481
6.0	705.952
8.0	692.385
10.0	679.679
12.0	667.756
14.0	656.493
16.0	645.872
18.0	635.792
20.0	626.212
22.0	617.114
24.0	608.437
25.0	604.248
25.4	602.906
25.451	0.0
1000.0	0.0

*Area - 4.99×10^{-3} ft^2

APPENDIX B

WRAP INPUT DATA FOR CALCULATION OF LOFT L2-3 TEST

The timestep selection criteria used in the LOFT L2-3 calculation are given in Table B-1. The normalized power function used in the reflood calculation (the blowdown calculation relied on solution of the point kinetics equations) is given in Table B-2. The axial power distribution for both blowdown and reflood calculations are given in Table B-3. The ECCS flow rates during reflood and the accumulator flow are summarized in Figures 20 and 21. The ECCS flows during the blowdown calculation are given in Table B-4.

Table B-1 LOFT L2-3: timestep selection

Blowdown

Timestep Size (sec) Min	Max	Time (sec) of Interval End
1×10^{-5}	2×10^{-4}	0.1
1×10^{-5}	1×10^{-3}	1.0
1×10^{-4}	1×10^{-2}	12.0
5×10^{-5}	5×10^{-3}	15.0
1×10^{-5}	1×10^{-3}	16.5
2×10^{-6}	2×10^{-4}	17.0
1×10^{-5}	1×10^{-3}	18.0
1×10^{-4}	1×10^{-2}	20.0
1×10^{-4}	1×10^{-2}	40.0
1×10^{-4}	1×10^{-2}	100.0

Reflood

Timestep Size (sec) Min	Max	Time (sec) of Interval End
1×10^{-4}	5×10^{-4}	0.4
1×10^{-4}	5×10^{-3}	5.0
1×10^{-4}	5×10^{-3}	44.0
1×10^{-4}	5×10^{-2}	90.0

Table B-2 LOFT L2-3: normalized power (reflood)

Time (sec)	Normalized Power
0.0	1.0
5.0	0.97877
10.0	0.96020
15.0	0.94372
20.0	0.92893
25.0	0.91555
30.0	0.90333
35.0	0.89212
40.0	0.88176
50.0	0.86317
60.0	0.84689
70.0	0.83243
80.0	0.81946
90.0	0.80771

Table B-3 LOFT L2-3: axial power profile (blowdown)

Mid-slab Elevation (ft)	Power Fraction		
	Average Core	Hot Channel	Hot Pin
0.25	-	-	0.76888×10^{-4}
0.50	0.14002	0.03904	-
0.75	-	-	0.12947×10^{-3}
1.25	-	-	0.15536×10^{-3}
1.375	0.16096	0.04433	-
1.625	-	-	0.79150×10^{-4}
1.875	-	-	0.83107×10^{-4}
2.125	0.17044	0.04637	0.82203×10^{-4}
2.375	-	-	0.78697×10^{-4}
2.625	-	-	0.71913×10^{-4}
2.875	0.13971	0.03743	0.65694×10^{-4}
3.125	-	-	0.58458×10^{-4}
3.500	-	-	0.10346×10^{-3}
3.750	0.13093	0.03479	-
4.000	-	-	0.77680×10^{-4}
4.875	0.04194	0.01291	0.68634×10^{-4}

(Table B-3, continued)

Axial Power Profile (Reflood)	
Mid-Slab Elevation (ft)	Power Fraction
0.25	0.068
0.75	0.1145
1.25	0.1374
1.625	0.0700
1.875	0.0735
2.125	0.0727
2.375	0.0696
2.625	0.0636
2.875	0.0581
3.125	0.0517
3.750	0.1602
4.875	0.0607

Table B-4 LOFT L2-3: ECCS Flows (Blowdown)

Low Pressure Injection

Pressure (psi)	Flow Rate (lbm/sec ft^2)*
0.0	15.924
12.3	15.425
62.3	13.388
112.3	11.060
137.3	9.5905
162.3	7.7750
172.3	6.9573
182.3	5.9040
192.3	4.7398
202.3	3.3678
212.3	1.5799
220.0	0.0
3000.0	0.0

High Pressure Injection

Constant 3.4648 lbm/sec ft^2*

Feedwater

Constant 42.971 lbm/sec ft^2*

Steamline

time (sec)	flow rate (lbm/sec ft^2)*
0.0	-42.971
3.26	-22.045
13.36	-10.471
14.06	0.0
100.0	0.0

*Area = 1 ft^2

APPENDIX C

WRAP INPUT DATA FOR CALCULATION OF THE FOUR-LOOP PWR TEST PROBLEM

The timestep selection criteria used in the four-loop PWR calculation are given in Table C-1. The normalized power history used in the reflood calculation (the blowdown calculation relied solely on reactivity feedback effects) is given in Table C-2, with the axial power profiles given in Table C-3. The ECCS flows during blowdown are given in Table C-4, and those for the reflood phase are given in Table C-5. The accumulator injection rate is given in Table C-6.

Table C-1 Four-Loop PWR: timestep selection

Blowdown		
Timestep size (sec)		
Min	Max	Time (sec) of Interval End
5×10^{-5}	1×10^{-2}	1.0
1×10^{-4}	2×10^{-2}	3.0
1×10^{-4}	2.5×10^{-2}	10.0
1×10^{-4}	5×10^{-2}	40.0

Reflood		
Timestep Size (sec)		
Min	Max	Time (sec) of Interval End
1×10^{-4}	1×10^{-1}	10.0
1×10^{-4}	1×10^{-2}	14.0
1×10^{-4}	5×10^{-3}	20.0
1×10^{-4}	1×10^{-1}	60.0

Table C-2 Four-Loop PWR: normalized power (reflood)

Time (sec)	Normalized Power
0.0	1.0
5.0	0.96767
10.0	0.93941
15.0	0.91439
20.0	0.89203
25.0	0.87184
30.0	0.85348
35.0	0.83668
40.0	0.82122
50.0	0.79365
60.0	0.76964
70.0	0.74851
80.0	0.72964
90.0	0.71270
100.0	0.69731
200.0	0.59436

Table C-3 Four-Loop PWR: axial power profiles

Blowdown

Mid-Slab Elevation (ft)	Power Fraction Average Core	Hot Channel	Hot Pin
1.092	0.100327	0.729100×10^{-3}	0.3979×10^{-5}
3.2755	0.205568	0.149391×10^{-2}	0.8152×10^{-5}
5.4585	0.256148	0.186149×10^{-2}	0.10158×10^{-4}
7.6415	0.238706	0.173473×10^{-2}	0.9467×10^{-5}
9.8248	0.157817	0.114689×10^{-2}	0.6259×10^{-5}
11.458	0.034172	0.248340×10^{-3}	0.1355×10^{-5}

Reflood

Mid-Slab Elevation (ft)	Power Fraction
0.5	0.0302322
1.5	0.0570512
2.5	0.0806134
3.5	0.0995737
4.5	0.112848
6.5	0.119681
7.5	0.119681
8.5	0.112848
9.5	0.0995737
10.5	0.0806134
11.5	0.0570512
	0.0302322

Table C-4. Four-Loop PWR: ECCS flows (blowdown)

pressure (psi)	Intact Loop Low Pressure Injection flow rate (lbm/sec ft^2)*
0.0	497.30
20.0	455.86
40.0	411.65
60.0	366.07
80.0	317.72
100.0	259.70
120.0	196.85
140.0	121.56
166.0	0.0
0.0	67.688
235.0	58.018
620.0	40.198
930.0	22.517
1150.0	8.288
1250.0	0.0

Pressure (psi)	Intact Loop Charging Flow Rate (lbm/sec ft^2)*
0.0	87.718
400.0	81.778
1000.0	69.760
1260.0	62.162
1450.0	55.808
1790.0	51.802
2110.0	41.442
2490.0	20.721
2640.0	0.0

*Area = 1 ft2

(Table C-4, continued)

Broken Loop Charging Pressure (psi)	Flow Rate (lbm/sec ft^2)*
0.0	31.081
400.0	28.733
710.0	27.628
1040.0	26.937
1800.0	17.267
2110.0	13.814
2490.0	6.907
2640.0	0.0

Broken Loop High Pressure Injection Pressure (psi)	Flow Rate (lbm/sec ft^2)*
0.0	23.345
70.0	24.865
200.0	28.733
290.0	32.739
400.0	33.153
490.0	32.739
740.0	29.700
1000.0	24.174
1200.0	18.649
1340.0	13.814
1445.0	6.9069
1500.0	0.0

*Area = 1 ft^2

(Table C-4, continued)

Pressure (psi)	Broken Loop Low Pressure Injection Flow Rate (lbm/sec ft^2)*
3.0	153.33
15.0	153.33
39.0	138.14
45.0	192.01
53.0	218.26
77.0	207.21
110.0	172.67
137.0	138.14
160.0	96.70
175.0	69.07
182.0	41.44
186.0	0.0

*Area = 1 ft^2

Table C-5 Four-Loop PWR: ECCS Flows (Reflood)

Low Pressure Injection	
Pressure (psi)	Flow Rate (lbm/sec ft^2)*
0.0	467.16
20.0	428.23
40.0	386.71
60.0	343.88
80.0	298.47
100.0	243.96
120.0	184.92
140.0	114.20
166.0	0.0

High Pressure Injection	
Pressure (psi)	Flow Rate (lbm/sec ft^2)*
0.0	63.586
235.0	54.502
620.0	37.762
930.0	21.152
1150.0	7.786
1250.0	0.0

*Area = 1 ft^2

Table C-6 Four-Loop PWR: Accumulator injection rate (Reflood)

Time (sec)	Flow Rate (lbm/sec ft^2)*
0.0	4418.2
0.25	4404.5
0.75	4377.5
1.25	4351.1
1.75	4325.2
2.25	4299.8
2.75	4274.9
3.25	4250.5
3.75	4226.5
4.25	4203.0
4.75	4179.8
5.25	4157.1
5.75	4134.8
6.25	4112.9
6.75	4091.4
7.25	4070.2
7.75	4049.3
8.25	4028.8
8.75	4008.6
9.25	3988.8
9.75	3969.2
10.25	3950.0
10.45	0.0
1000.00	0.0

*Area = 1 ft^2