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WRAP-BWR VERIFICATION STUDIES

Performed by

(Dupont) E. I. duPont de Nemours & Co.
Savannah River Laboratory

for the

U. S. NUCLEAR REGULATORY COMMISSION

Please make the following correction to the title page of the report.

Delete: Prepared ~~for Probabalistic Analysis Staff~~

Replace With: Prepared for Water Reactor Safety Research Staff
NRC FIN No. B4126

DIVISION OF TECHNICAL INFORMATION
AND
DOCUMENT CONTROL

NUREG/CR-1680
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PREPARED FOR THE U. S. DEPARTMENT OF ENERGY UNDER CONTRACT DE-AC09-76SR00001

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February 1981

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E. I. du Pont de Nemours and Company
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ABSTRACT

A modular computational system known as the Water Reactor Analysis Package - Evaluation Model (WRAP-EM) has been developed for the Nuclear Regulatory Commission (NRC) to interpret and evaluate reactor vendor EM methods and computed results. A subset of the system (WRAP-BWR-EM) provides the computational tools to perform a complete analysis of loss-of-coolant accidents (LOCAs) in boiling water reactors (BWR). A series of calculations modeling tests run on the General Electric Two Loop Test Apparatus, and calculations of a large break in a BWR/4 and BWR/6 plant have verified that the WRAP-BWR-EM system is functioning as intended. Additional calculations using a BWR/6 reference design were run as part of another study to determine the sensitivity of calculated system parameters to code input and the selection of calculational options available.

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ACKNOWLEDGMENTS

The Idaho National Engineering Laboratory (INEL) and the Risϕ National Laboratory (Denmark) provided key support during the development and verification of WRAP-BWR-EM. The efforts of the members of the staffs at INEL and Risϕ are gratefully acknowledged. In addition to the authors, several other individuals in the Savannah River Laboratory contributed to the development of WRAP-BWR-EM. Their names are listed below, and their contributions gratefully acknowledged.

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1.0 INTRODUCTION

The Water Reactor Analysis Package - Evaluation Model (WRAP-EM) is a modular system of codes which performs a complete licensing type analysis of postulated loss-of-coolant accidents (LOCAs) 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) to interpret and audit reactor vendor calculational methods and computed results. Vendor safety analyses are required to conform to the regulations set forth in Appendix K of 10 CFR Part 50,[1] which were designed to ensure a conservative prediction of peak clad temperature. The evaluation models in WRAP-EM are designed to conform to the Appendix K requirements. The systems for boiling water reactors (WRAP-BWR-EM) and pressurized water reactors (WRAP-PWR-EM) are described in References 2 and 3. The final step in the BWR development program - verification of WRAP-BWR-EM - is documented in this report.

NRC specified a series of analyses to be run to verify that the WRAP-BWR-EM system was functioning properly and was capable of correctly modeling physical phenomena in different BWR systems. Input for these analyses was derived from RELAP4/MOD5[4] input data decks prepared by the Idaho National Engineering Laboratory (INEL). The transients analyzed were two large break tests in General Electric's Two Loop Test Apparatus (TLTA),[5] and double-ended large pipe breaks in a BWR/4 and BWR/6 reactor.

In general, a single analysis was run for each transient; thus sensitivity to input parameters was not determined. However, as part of the verification program, portions of selected transients were rerun with alternative models to improve agreement with measurements or reference calculations. An extensive series of calculations were planned to determine sensitivities to code input and system models for both a BWR and a PWR reactor system.

No small break analyses were included in these verification studies because the capability of WRAP to compute small break transients had been demonstrated in a series of calculations for the NRC Bulletins and Orders Task Force during 1979.[6]

2.0 SUMMARY

WRAP-BWR-EM successfully calculated transients for a BWR test facility (TLTA) and two BWR reactor designs (BWR/4 and BWR/6). The calculated behavior of all the systems was physically reasonable, and the results of the calculations were self-consistent. Thus, the WRAP-BWR-EM system was judged to be an acceptable tool to interpret and evaluate reactor vendor licensing calculations.

For the TLTA blowdown tests, the WRAP-BWR-EM calculations showed the same general behavior as the test data. Test 6007, which was run without any emergency core cooling, was calculated to depressurize more rapidly than Test 6406, which included Emergency Core Cooling System (ECCS) flow. For both tests, the calculated rates of depressurization were larger than the measured rates because the Moody[7] critical flow model, with a discharge coefficient of 1.0, gave a much larger break flow than was measured early in the transient. After about 30 seconds, the calculated clad surface temperatures, which were 200 to 300°F above the measured data, showed a steady increase similar to that observed in the experiments. However, early in the transient the calculated temperatures were very high because the very low core flows, at that time, were outside the range of the default critical heat flux correlation selected for these analyses.

For the analysis of the BWR/4 plant, parallel calculations were run at SRL and INEL. A large break in the suction side of the recirculation loop was modeled. WRAP-EM gave essentially the same results as the calculations at INEL. This demonstrated that the WRAP-EM modules, derived from established codes developed at INEL, were functioning as intended.

The large break analysis for the BWR/6 plant successfully employed all the components of the WRAP-BWR-EM system. The calculated system behavior was reasonable and showed the important physical phenomena anticipated for a BWR LOCA. To provide an additional point of reference, WRAP-BWR-EM was compared to a vendor generic analysis for the same plant. WRAP showed the same general behavior as the vendor results. However, WRAP gave much lower clad surface temperatures because WRAP did not calculate departure from nucleate boiling until much later in the transient. The difference has tentatively been ascribed to different flow models. However, to fully resolve or explain the differences will require more complete information on the vendor code input and models.

3.0 THE WRAP-BWR-EM SYSTEM

The WRAP-BWR-EM system is a major extension of the WRAP[8,9] (Water Reactor Analysis Package) system developed at SRL during 1977. WRAP is a modified version of the RELAP4*

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

Code[4] 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-BWR-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 reflood phase of the LOCA.
- Calculation of the temperature of the fuel at the hottest plane of the core.

The overall structure of the WRAP-BWR-EM system is shown in Figure 1. Initial fuel conditions are calculated as a function of burnup by the GAPCON[10] module. These conditions are passed to MOXY[11] and WRAPIT,[8] the generalized input processor for initialization of the transient fuel models. GAPCON results are also stored on magnetic tape or disk for subsequent calculations.

The WRIN module contains the WRAP BWR steady-state initialization procedure, BWRSS[12], and the RELAP4 initialization in which residual flow resistances are computed to balance the system. 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 magnetic tape. The results can be plotted by the WROP module and provide the hydraulic conditions for the hot plane analysis (MOXY module).

At the end of blowdown (EOB), system renodalization is performed by the WRAP-NORCOOL interface routine, and the reflood phase of the accident is calculated by the NORCOOL[13] module. The time to hot plane quench is passed to MOXY to determine the end time for the hot plane analysis. Other capabilities within the system include the transient restart capability provided by WRROT and MWRROT. MWRROT also has the capabilities of system renodalization and problem re-specification. 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 14.

4.0 VERIFICATION CALCULATIONS

In the following subsections each of the analyses for the WRAP-BWR-EM verification is discussed in detail. For each analysis, pertinent background information is supplied, results

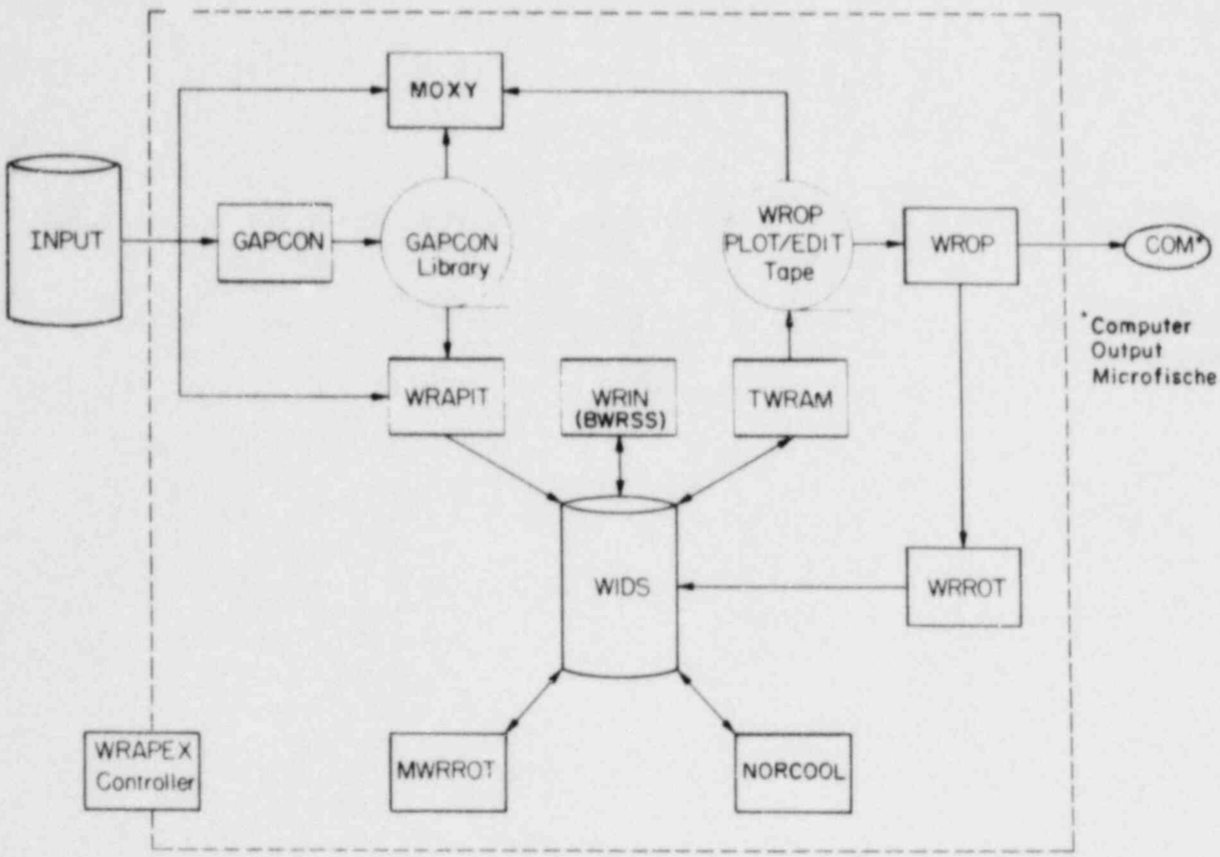

FIGURE 1. WRAP-BWR-EM System Overview

are summarized and conclusions 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 TLTA Tests 6007 and 6406

4.1.1 Background

Comparing calculations to measured data is a key step in code verification. For an evaluation model code, models are selected to ensure a conservative calculation. Thus, EM calculations are not expected to match experiments, but the results should have the same general behavior and yield higher peak surface temperatures than the experiments. As part of the verification of the WRAP-BWR-EM system, calculations were made of two tests run in General Electric's TLTA. TLTA is a volumetrically scaled version of a boiling water reactor with a single, full-size, electrically heated fuel assembly. Figure 2 is a diagram of TLTA showing the regions corresponding to specific parts of a BWR. A detailed description of TLTA is given in Reference 5.

The tests selected for these calculations were Tests 6007* and 6406*. Both tests were run at power levels representative of normal BWR operating power conditions. The major difference in the tests involved the ECCS which was available only for Test 6406. These particular tests were selected because the measured behavior was different than had been expected. Test 6007 without ECCS had depressurized more rapidly than 6406. A question to be answered in the verification was whether WRAP-BWR-EM would predict the same behavior.

All the modules in the WRAP-BWR-EM system were not exercised in the TLTA study. Fuel assemblies in TLTA were electrically heated ceramic rods, and thus the WRAP modules which calculate nuclear fuel pin behavior (GAPCON and MOXY) were not applicable. Since the focus of the experiments was on the blowdown phase of the transients, the experiments were terminated before the reflood stage obviating the need to exercise the NORCOOL** module in WRAP. A steady state analysis was not run, since the system was never brought to a true equilibrium condition before opening a valve to model the recirculation line break. Initial conditions were taken from RELAP input generated by INEL.

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* These are actually specific tests from a sequence of runs numbered 6007 and 6406.

** NORCOOL employs nuclear fuel assembly models, so it could not have been used for TLTA even if the tests had extended into the reflood phase.

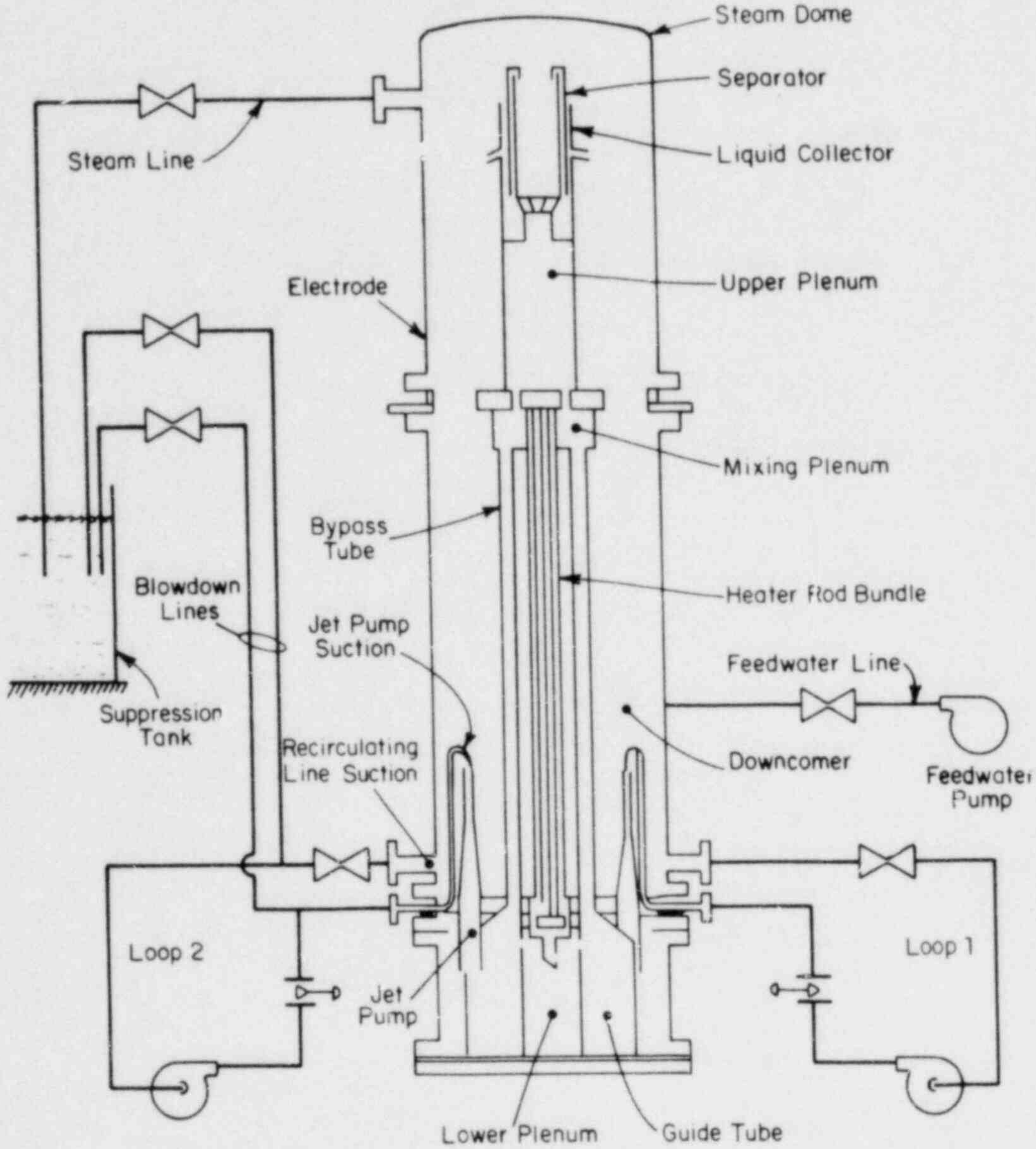

FIGURE 2. TLTA4 for 8 x 8 Blowdown Heat Transfer

4.1.2 Summary and Conclusions

WRAP-BWR-EM satisfactorily calculated the blowdown phase of two tests in the TLTA which simulated a large break LOCA with and without the ECCS. The calculated rates of depressurization showed the same relative behavior as the experiments. Test 6007 depressurized more rapidly than Test 6406. However, the calculated rates of depressurization were faster than the observed rates for both tests because the EM calculations used the Moody[7] critical flow correlation, which has a discharge coefficient of 1.0 and break flows larger than measured early in the transient. Additional calculations with a coefficient of less than 1 will be considered for a future program. For a complete licensing analysis, Appendix K of 10 CFR 50 requires calculation for discharge coefficients, ranging from 0.6 to 1.0.[1]

The calculated peak clad temperatures were larger than the measured surface temperatures. Early in the transient, the calculated surface temperatures were very high because the WRAP default critical heat flux (CHF) correlation (Barnett, and modified Barnett)[15,16] gave a departure from nucleate boiling which was not observed in the experiment. Later in the transient, the calculated and measured surface temperatures showed the same general behavior with the calculated rate of increase slightly larger. A subsequent calculation of the first few seconds of the Test 6007 blowdown using the Hench-Levy CHF correlation[17] gave surface temperatures that were closer to the measured data.

4.1.3 Input

The WRAP volume and junction nodalization for TLTA is given in Figure 3. This nodalization was taken directly from that developed for RELAP4/MOD5 calculations at INEL. Figure 4 shows the core heat slab assignments for the Test 6007 calculation. The heat producing regions of the core assembly were modeled by two stacks of heat slabs; one stack for the hottest portion and one stack for the rest of the assembly. The same model was used for Test 6406. The measured and WRAP initial conditions are given in Table 1. In general, the code input was within the uncertainty of the experimental data. The time-dependent normalized power tables used in the calculations are given in the Appendix (Table A-1). These powers are the actual measured data. The power levels were not adjusted to conform to the enhanced levels specified in 10 CRF 50, Appendix K.[1] The ECCS flows for the Test 6406 calculation were specified by flow versus time tables, which duplicated the measured flow behavior.* This differs from the

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* This is the procedure recommended at a Program Managers Group Meeting held in San Jose, California, in March of 1979.

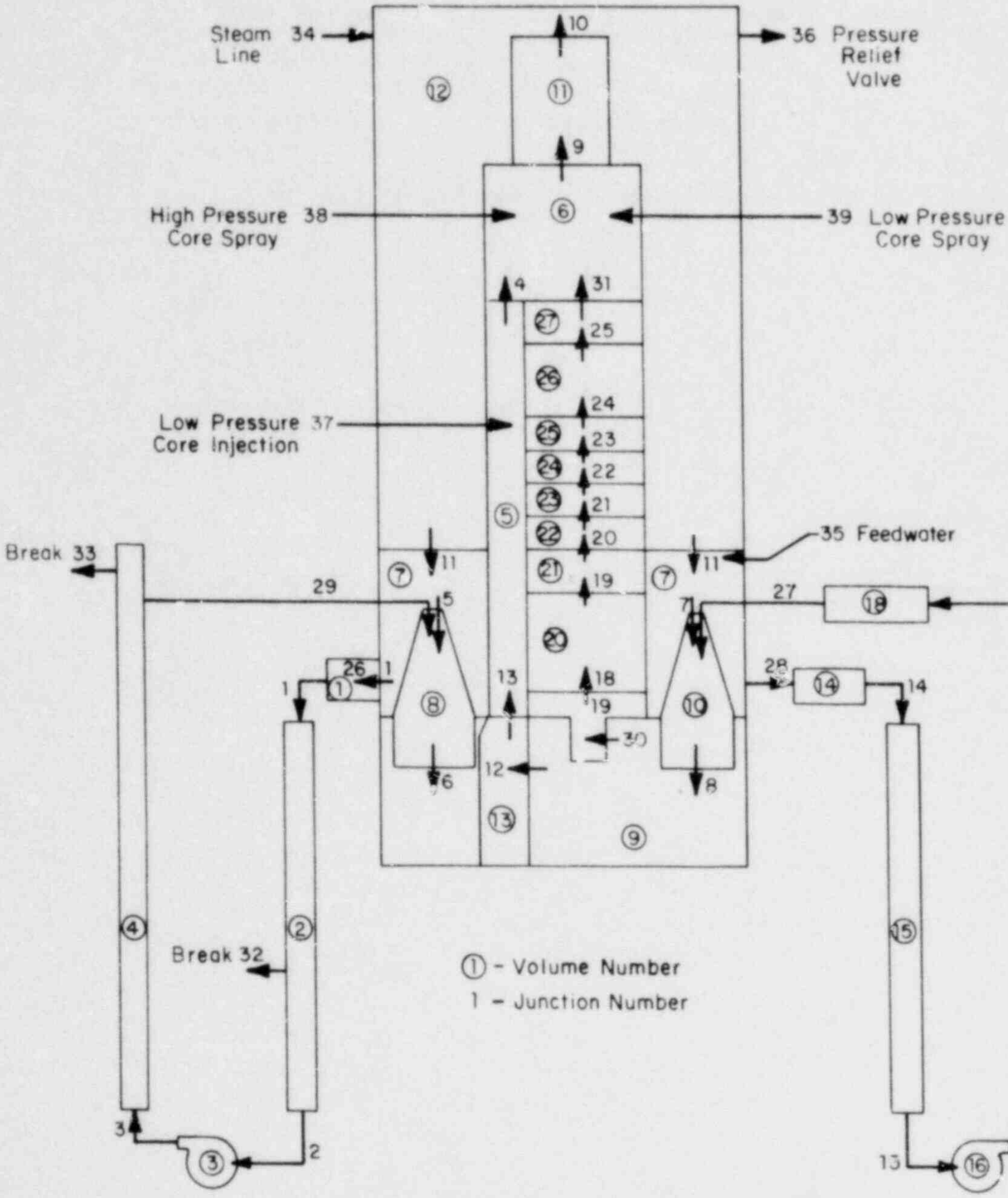

FIGURE 3. WRAP Volume and Junction Nodalization for TLTA

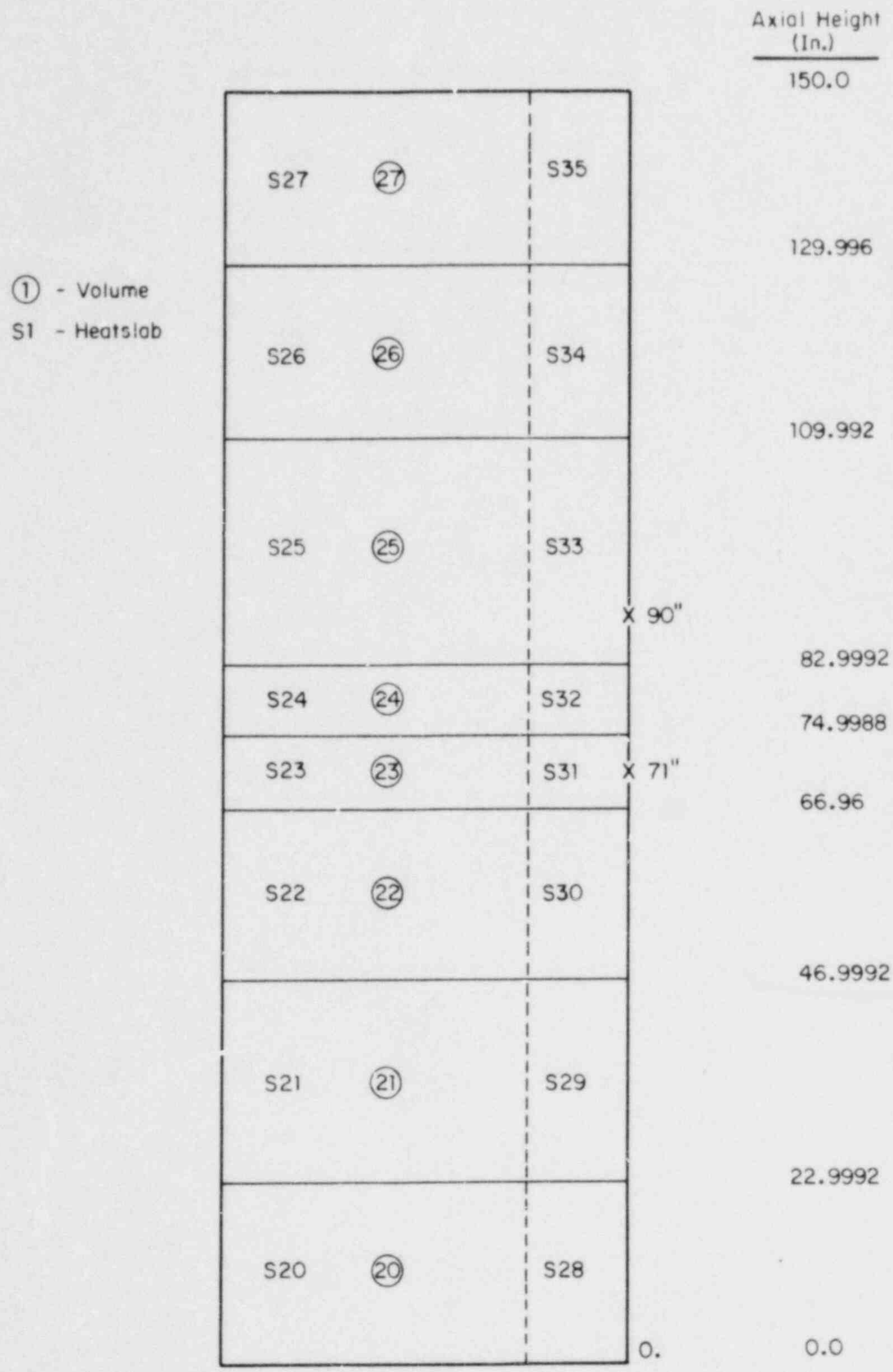

FIGURE 4. WRAP Nodalization of TLTA Heat Slabs in Core (Test 6007). Areas pictured represent relative volumes of volumes and heatslabs.

TABLE I

Initial Conditions for TLTA

	TEST 6007		TEST 6406	
	Experiment	Wrap	Experiment	Wrap
Bundle Power (MW)	5.04 ± 0.15	5.04	5.04 ± .15	5.04
Steam Dome Pressure (psia)	1050 ± 4	1051.34	1056 ± 4	1056.03
Lower Plenum Pressure (psia)	1071 ± 4	1071.18	1076 ± 4	1077.22
Lower Plenum Enthalpy (Btu/lbm)	519 ± 10	512.08	539 ± 5	539.054
Initial Water Level - Elevation (in.)	121 ± 5	121.6	126.5 ± 6	126.86
Above J. P. Suction	44.75 ± 5	45.0	50.25 ± 6	50.26
Feedwater Enthalpy (Btu/lbm)	51 ± 5	50.92	45 ± 2	45.0
Bundle Inlet to Outlet P (psia)	15.5 ± 1.5	16.6	15.0 ± 1.5	
Steam Flow (lb/sec)	5.60 ± 0.6	5.6	6.30 ± 0.6	6.374
Feedwater Flow (lb/sec)	1.34 ± 0.15	1.6158	0.35 ± 0.05	0.35
Intact Loop Drive Pump Flow (lb/sec)	8.4 ± 0.8	9.4	8.8 ± 0.8	8.8
Broken Loop Drive Pump Flow (lb/sec)	8.4 ± 0.8	9.4	9.6 ± 0.9	9.6
Intact Loop Jet Pump Flow (lb/sec)	20.2 ± 2	20.5	18.4 ± 2	18.4
Broken Loop Jet Pump Flow (lb/sec)	21.0 ± 2	20.5	20.0 ± 2	20.2
Bundle Inlet Flow (lb/sec)	36.0 ± 3.5	37.5	35.5 ± 3.5	35.5

normal calculational procedure in which flow is started by a signal based on water level or system pressure, and the flow rate is determined by a flow versus pressure fill table. The flow tables for the High Pressure Core Spray (HPCS), the Low Pressure Core Spray (LPCS), and the Low Pressure Core Injection (LPCI) are given in the Appendix (Table A-2).

Evaluation models were specified by:

- Invoking the Henry-Fauske[18] critical flow model and the Moody[7] critical flow model in the subcooled and two-phase regions, respectively.

- Invoking the EM heat transfer logic, which is designed to prevent return to nucleate boiling after the CHF is exceeded and the clad temperature exceeds the coolant saturation temperature by 300°F.

- Selecting the Barnett[15] and Modified Barnett[16] CHF correlation. (This is one of the acceptable correlations listed in 10 CFR 50, Appendix K*.)

- Selecting the Groenveld[19] 5.9 film, boiling, heat transfer correlation for all heatslabs.

The EM logic for phenomena such as metal-water reaction, fuel pin swelling, and fuel pin flow blockage was not invoked since they were not applicable to electrically heated fuel assemblies. For reference, the timestep specifications for each calculation are given in the Appendix (Table A-3).

4.1.4 Results

The WRAP calculation for Test 6007 ran to 54 sec of reactor time, ending on a low pressure (<150 psia) trip in the steam dome. All trends were well established at that time and the problem was not restarted. CPU time required was 53 min for 14,000 time steps on the SRL IBM 360/195 computing system. For the Test 6406 WRAP calculation, the low pressure trip in the steam dome was lowered to 5 psia. The calculation was stopped at 185 reactor sec, after the system pressure had remained below 20 psi for 20-30 sec. The calculation required 268 CPU min for 67,000 time steps. Selected results from the WRAP calculations and available experimental data are plotted in Figures 5-12.

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* The Hench-Levy correlation[16] used for a second calculation of Test 6007 is also acceptable for EM analyses.

The calculated total break flow* for Test 6007 (Figure 5) was significantly larger than the measured flow early in the transient. (Measured flow data was available only after about 12 sec.) The maximum flow was nearly 400 lb/sec at 0.15 sec. These very large flows are not shown in Figure 5 in which the mass flow scale was expanded to highlight the differences between the calculation and the available measured data. The calculated break flow is larger than the measured flow because the EM Moody model was used with a discharge coefficient of 1.0. The calculated total break flow for Test 6406 was similar to the calculated flow for Test 6007 but generally was lower during the first 25 sec. No measured break flow data was available for Test 6406, but the calculated flows should be larger, since the Moody model was used with a discharge coefficient of 1.0.

Figures 6 and 7 show the calculated and experimental steam dome pressure. All calculations show a faster depressurization than was observed in the experiments. WRAP-calculated Test 6007 reached 150 psia at 54 sec and Test 6406 reached 150 psia at about 61 sec. In the experiments, Test 6007 reached 150 psia at about 95 sec and Test 6406 at about 125 sec. The WRAP calculation shows Test 6007 depressurizing faster than Test 6406 in relative agreement with experiment.

The calculated steam line flows, which are a function of steam dome pressure, were lower than the measured flows because the calculations gave a more rapid depressurization than was observed in the experiments. The calculated and measured jet pump flows had the same general behavior. Early in the transient (<2 sec), the calculated core flows remained positive, while the experiments indicated negative core flows during the time from 5 to 10 sec (See Figures 8 and 9). The local peaking of the calculated core inlet flows around 5 sec is due to agreement in the core bypass and the jet pumps, which force flow into the lower plenum. The calculated local peaking at about 10 sec is due to lower plenum flashing. This peak occurs 2 to 4 sec earlier than in the measured data, because the calculated rate of depressurization is faster than the measured rate.

The lower plenum mass is underpredicted for both tests, because the lower plenum is calculated to flash earlier than was observed in the tests. Figure 10 compares the calculated and measured lower plenum mass for Test 6406. The marked difference in behavior beyond 100 sec is probably caused by deficiencies in the TWRAM models, which are not expected to give accurate results

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* Summation of flows from the recirculation loop pump drive and drive suction line (junctions 32 and 33, Figure 3).

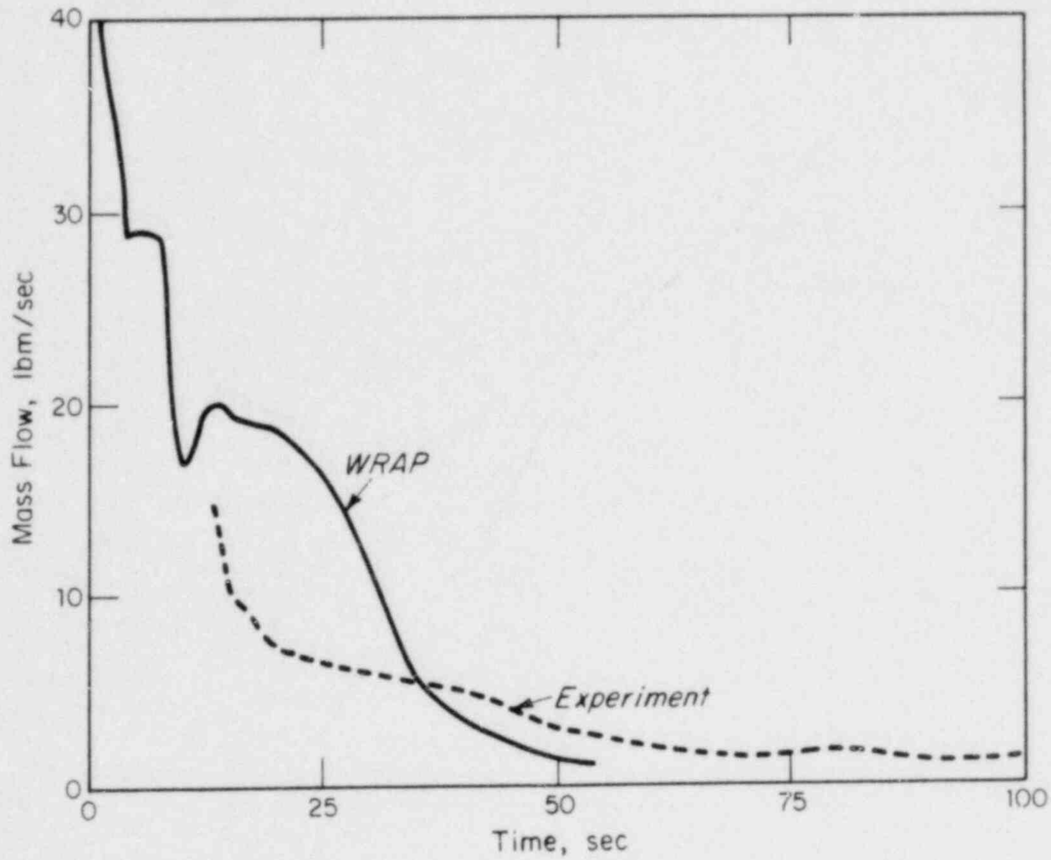

FIGURE 5. Total Break Flow for TLTA Test 6007

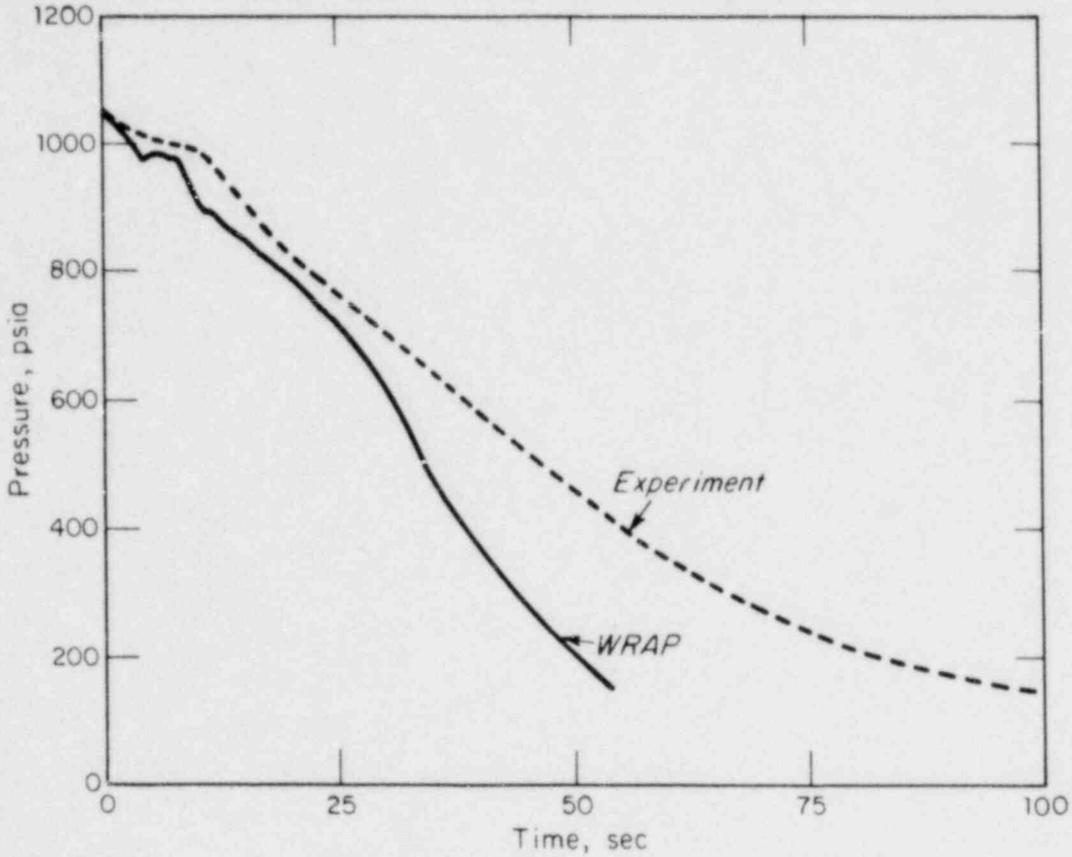

FIGURE 6. Steam Dome Pressure for TLTA Test 6007

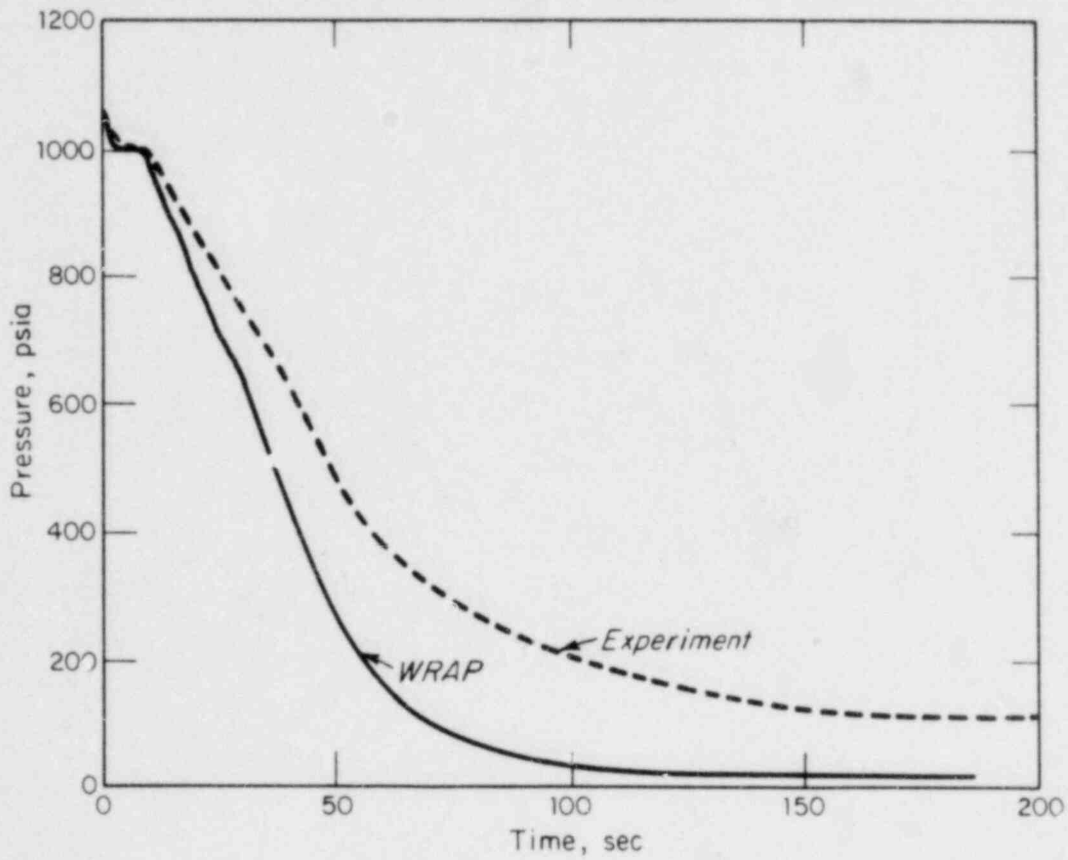

FIGURE 7. Steam Dome Pressure for TLTA Test 6406

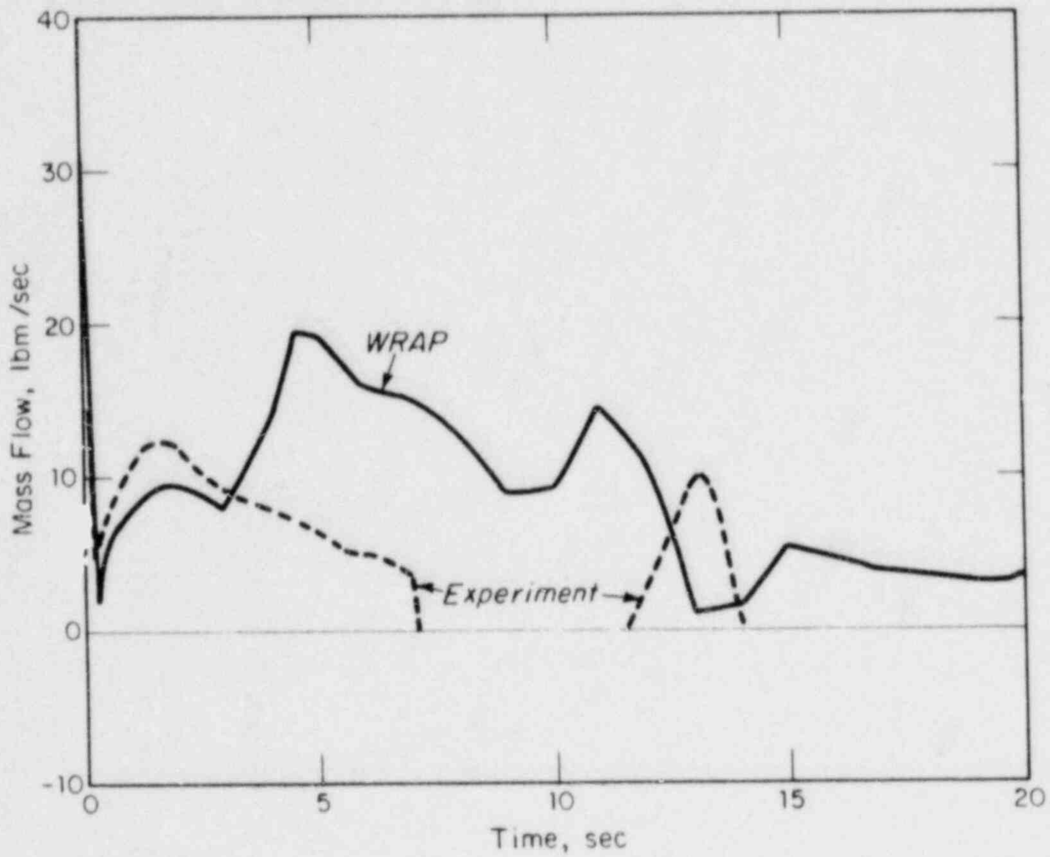

FIGURE 8. Core Inlet Flow for TLTA Test 6007

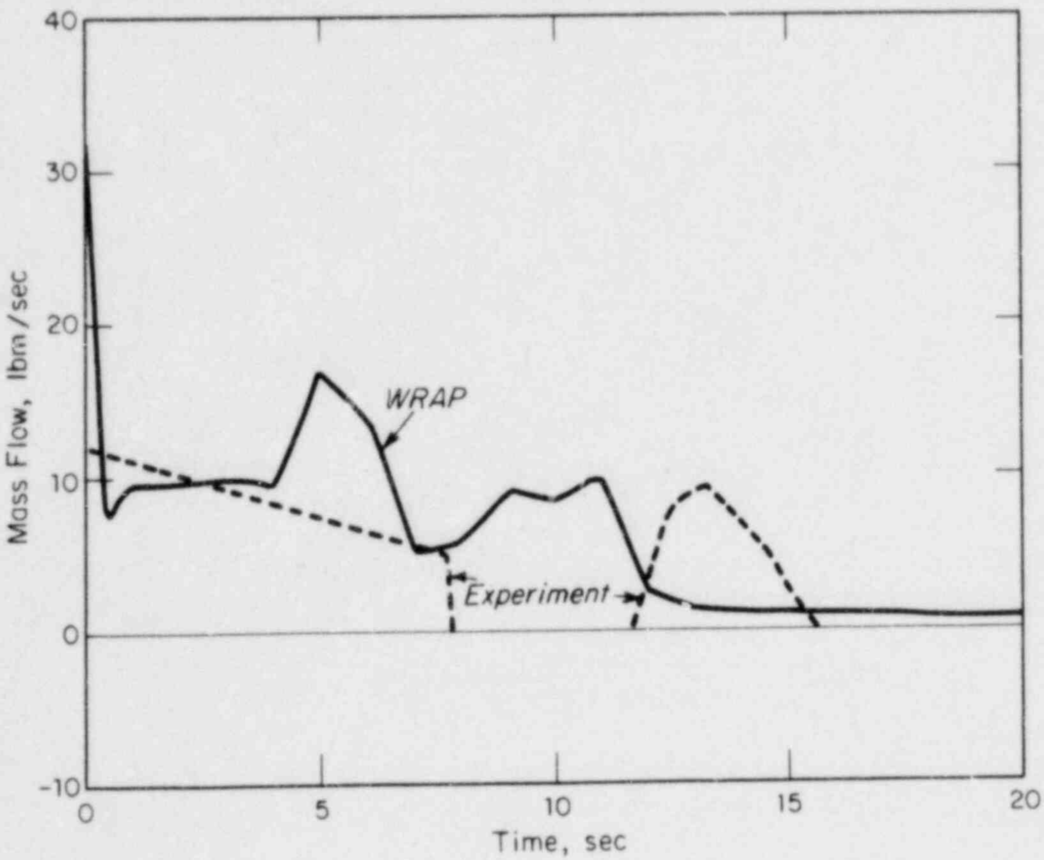

FIGURE 9. Core Inlet Flow for TLTA Test 6406

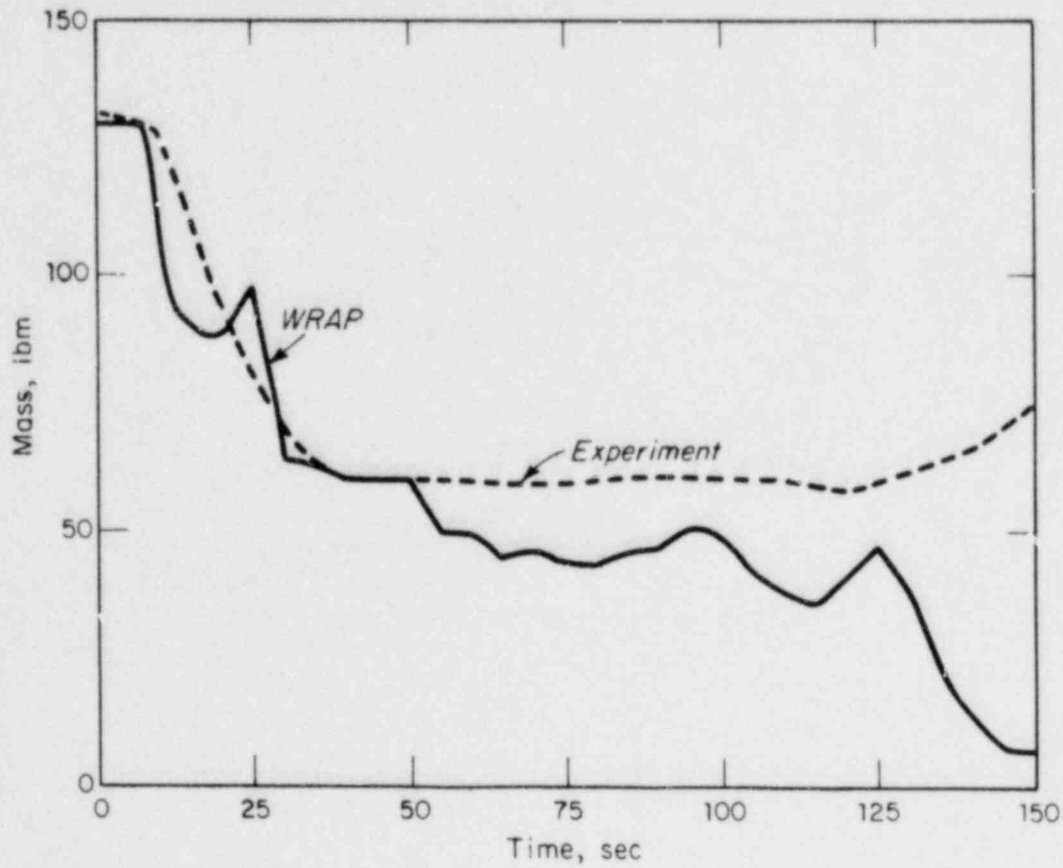

FIGURE 10. Lower Plenum Mass for TLTA Test 6406

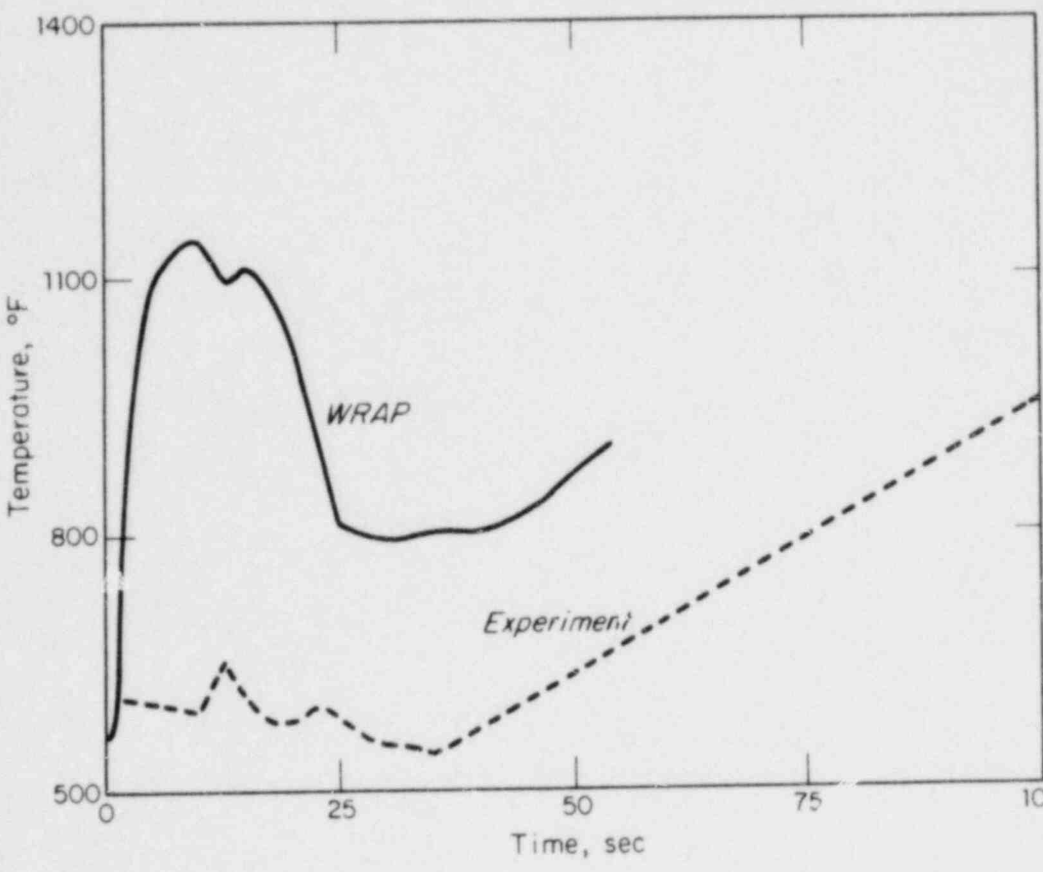

FIGURE 11. Peak Clad Temperature for TLTA Test 6007

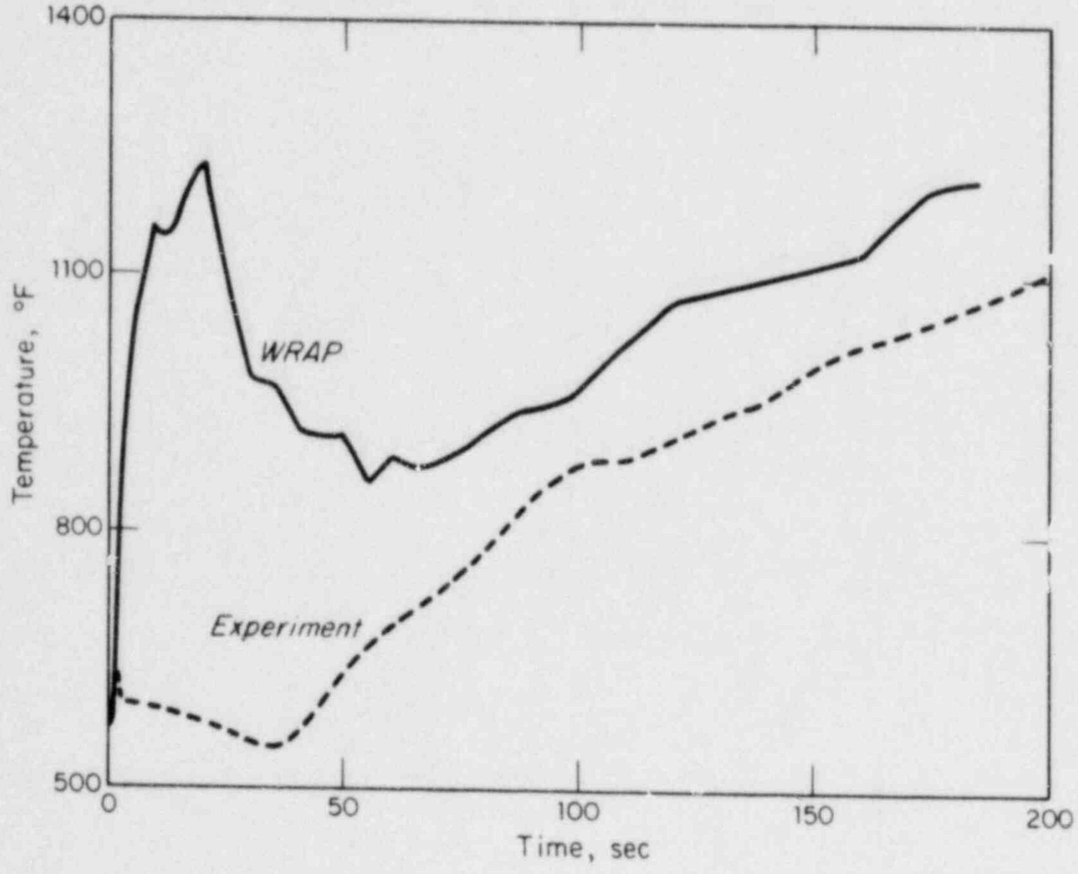

FIGURE 12. Peak Clad Temperature for TLTA Test 6406

for system pressures below 75 psi.[4] For TLTA, the calculated system pressure was about 20 psi after 100 sec. In general, for complete WRAP analyses of BWRs, the TWRAM calculation would be ended by an end-of-blowdown signal at a pressure near 75 psi, and a core reflood calculation (using NORCOOL) would be started.

Figures 11 and 12 show the peak clad temperature (PCT) for Test 6007 and 6406, respectively. The measured PCTs were taken from thermocouples located at 71, 79, and 90 in. The WRAP-calculated PCTs were for heat slabs S31, S32, and S33 which corresponded to the same region of the heat element. The WRAP PCTs are higher than the measured temperatures at all times during the blowdowns. However, early in the transient the calculated temperatures were very high. The calculations gave a very early departure from nucleate boiling which was not observed in the tests. Analysis showed the calculated core flows early in the transient were outside the recommended range for the CHF correlation selected. (A subsequent calculation with an alternate CHF correlation is discussed in the next section.)

Figure 13 shows the measured clad surface temperature at an elevation of 71 in., and the calculated surface temperature for the heat slab which includes the 71-in. elevation (Figure 4). Initially the calculated temperatures are lower than experiment at this level. However, after about 110 sec, the calculated temperatures exceed measured values. The measured drop in the temperature at about 100 sec due to core rewetting is not reproduced in WRAP-EM, which does not allow rewet once the critical heat flux has been exceeded and the temperature difference between the clad and saturated fluid has exceeded 300°F.

4.1.5 Additional Analyses

An additional calculation was run out to 8 sec of reactor time for Test 6007 using the Hench-Levy* critical heat flux correlation[17] for heat slabs in the core. This was done to verify that the anomalously high temperatures calculated early in the transients were due to use of the Barnett and Modified Barnett CHF correlations.[15,16] As is shown in Figure 14, the calculated PCTs are more in line with the experiment. Other parameters such as break flow and pressure were similar to values computed using the Barnett and Modified Barnett CHF correlations. The fact that the calculated PCT is lower than the experimental data is probably due to

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* An acceptable EM correlation as given in 10 CFR Appendix K.

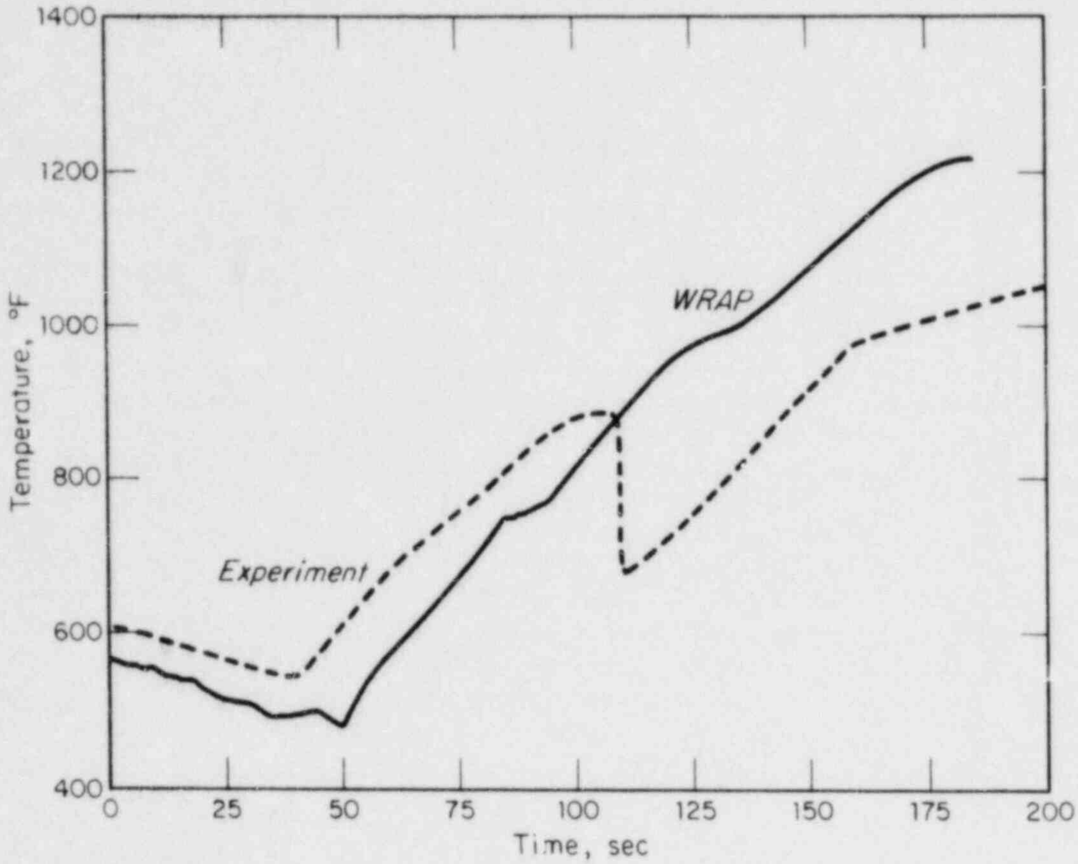

FIGURE 13. Clad Temperature at 71 in. for TLTA Test 6406

the discrepancy observed for the initial conditions. This could be due to either a poor model of the initial state of the system or the fact that the measurement is not made exactly at the fuel surface. The more rapid loss of coolant at the break and the "lockout" of return to nucleate boiling would probably cause the calculated peak clad surface temperature to exceed the measured value later in the transient.

4.1.6 Uncertainties

The additional analysis showed that calculated surface temperatures, at least early in the transient, are very sensitive to the choice of CHF correlation. Changing from Barnett and Modified Barnett to Hench-Levy reduced calculated surface temperatures of over 500°F.

The input initial conditions were different from the measured data, although the code input was generally within assigned experimental errors. The effect of these differences was probably small since no variation between calculations and experiment could be traced to the input data. Differences between the "point" at which the measurements were made and the "point" at which a parameter was computed introduced some uncertainty in comparisons.

Temperatures were measured at a point, while calculated temperatures were averaged over an axial segment. Depending on the gradients, this could cause the calculated values to be either high or low. Metal temperatures were measured using thermocouples attached to the surface of the rod, while the calculation gave metal temperatures at the true heat slab surface (in contact with the coolant).

A complete EM analysis requires a search over a range of values of the break discharge coefficient, C_D, to determine the value that yields the highest PCT. Since these calculations, run with a C_D value of 1.0, gave a much larger break flow than measured and strongly influenced the overall behavior, variations in C_D are expected to influence calculated behavior. Because of complex interactive effects, the magnitude and direction of the changes in surface temperature cannot be predicted.

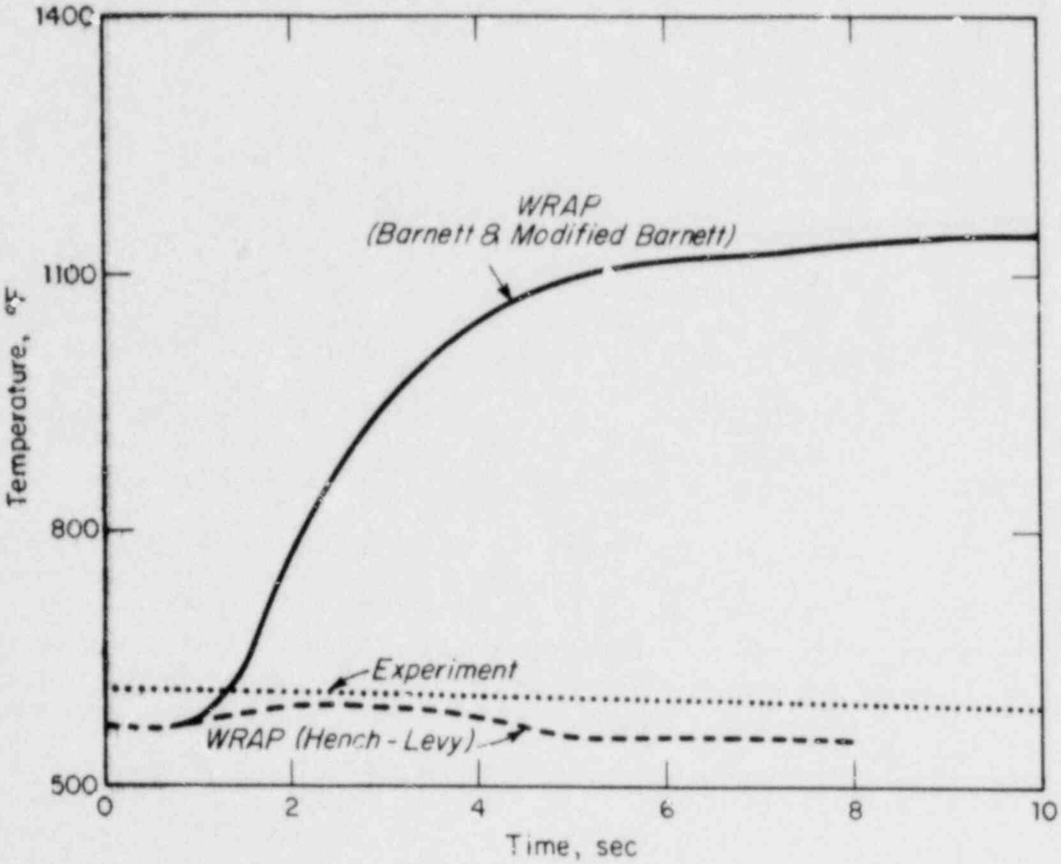

FIGURE 14. Effect of CHF Correlations on Calculated Peak Clad Temperature for TLTA Test 6007

4.2 BWR/4 LOCA

4.2.1 Background

The WRAP system is based on models developed at INEL for the blowdown calculation (RELAP4) and the hot plane analysis (MOXY). To test the WRAP-BWR-EM system, parallel calculations were run at INEL with established codes and at SRL with WRAP-BWR-EM for a suction line break in a BWR/4 plant. The initial fuel pin data were taken from GAPCON runs at SRL. INEL set up, initialized, and performed a RELAP4/MOD5 blowdown calculation using EM options and then analyzed the hot plane using their version of MOXY.[11] SRL converted the INEL input to WRAP input and ran the same calculations. This particular sequence of calculations did not test the full capability of WRAP-BWR-EM because there was no automated steady state calculation (BWRSS) or refill and reflood analysis (NORCOOL). Review of these phases of the calculation for the BWR/4 plant has already been documented in References 2 and 12.

4.2.2 Summary and Conclusions

There was excellent agreement between WRAP results and results of calculations with established codes at INEL. The results for the blowdown calculation were essentially the same. Minor differences in the MOXY calculations for the hot fuel pin were due to known differences in the SRL and INEL versions of MOXY. These calculations established that the WRAP-BWR-EM system was functioning as intended and gave physically reasonable results.

4.2.3 Input

Selected input are discussed in the following paragraphs. The WRAP nodalization showing system volumes and junctions is given in Figure 15. Figure 16 shows the volume and junction nodalization for the reactor core regions. A power-producing heat slab is associated with each core volume. The hot channel represents the hottest fuel bundle in the core which generates 40% more power than the average powered bundle. The nodalizations are identical to those developed by INEL.*

The initial power, power peaking, and system flows are given in Table 2. The axial power profile, ECCS flows, and time step specifications for the blowdown analysis are given in the Appendix.

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* Data transmitted to SRL in a letter from P. North, E.G. & G. to R. E. Tiller, Idaho Operations Office - DOE, "Completion of WRAP-BWR Tasks - PN-178-79," October 19, 1979.

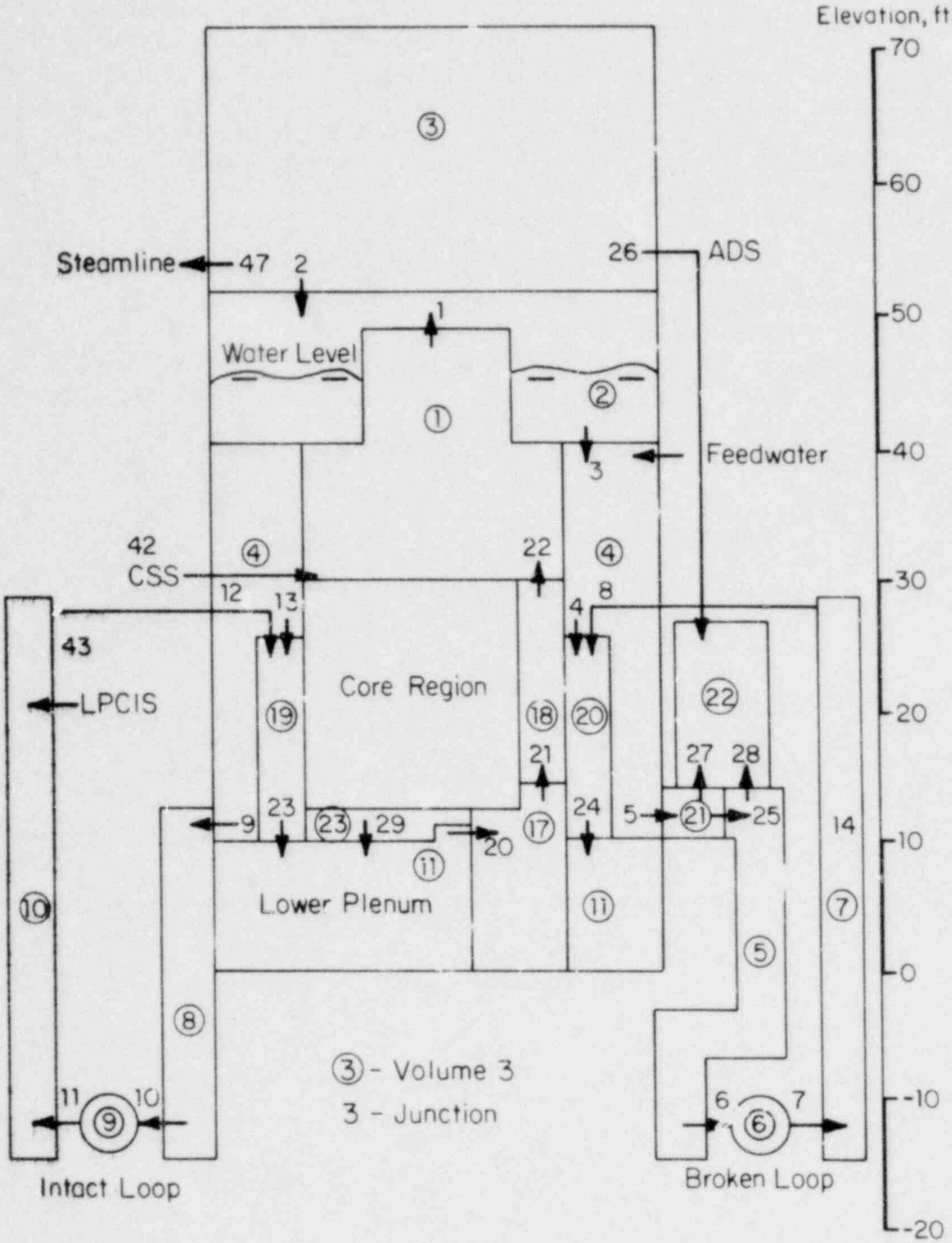

FIGURE 15. BWR/4 Nodalization for WRAP

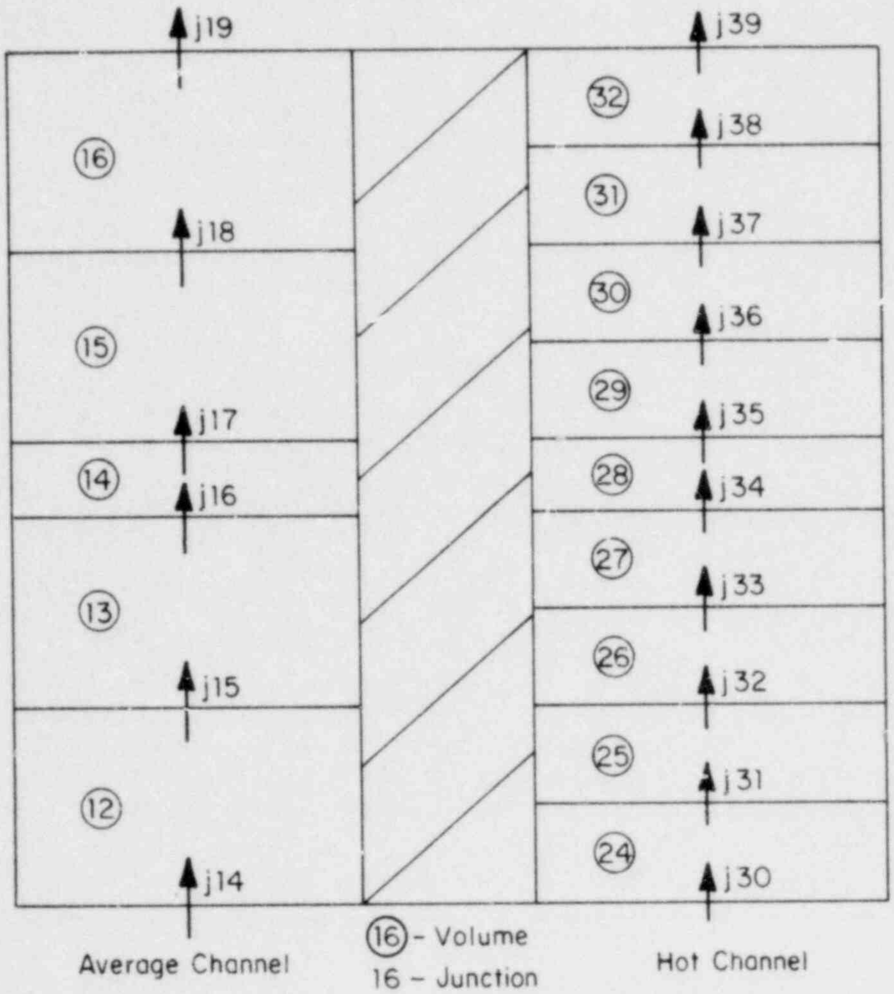

FIGURE 16. BWR/4 Core Model

Table 2

BWR/4 Initial Conditions

Condition	Value
Initial Reactor Power	3388 MW(t)
Hot Assembly Peak Linear Power	8.93 kW/ft
Hot Pin Peak Linear Power	9.74 kW/ft
Steamline Flow	4065.59* lbm/sec
Feedwater Flow	4074.31* lbm/sec
Steam Dome Pressure	1025 psi

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* Values specified in INEL input. No attempt was made to correct this imbalance, since the objective was to run the identical calculation.

The TWRAM blowdown analysis was run as an EM calculation by selecting:

- A fission product decay rate 1.2 times the ANS standard (heat from radioactive decay of actinides was included).
- The Henry-Fauske[18] (subcooled) and Moody[7] (saturated) critical flow models.
- The Hench-Levy CHF correlation[17].
- The Groenveld 5.7[19] post - CHF heat transfer correlation.
- Experimental data for the two-phase pump head degredation.
- Models for fuel rod rupture and flow blockage as a function of ΔP across the cladding.[20]

These models were taken directly from the INEL input.

The hot plane analysis code, MOXY, normally uses fuel input data from GAPCON calculations. The automatic GAPCON/MOXY interface was not used for this verification. The INEL input data for MOXY was used directly in the SRL analysis. The blowdown data from TWRAM was passed directly to MOXY via the TWRAM/MOXY interface. The MOXY calculation was run using the EM options given in Table II of Reference 2.

4.2.4 Results

The WRAP blowdown calculation (TWRAM) for the BWR/4 was run to 38 sec of reactor time. The calculation was stopped at this point since there was excellent agreement between WRAP and the INEL RELAP run. A summary of major events is given in Table 3. TWRAM required 90.0 CPU min on the SRL IBM 360/195 computing system. The MOXY hot plane calculation required 4.0 CPU min. Selected output data is discussed in the following paragraphs.

Figure 17 shows the steam dome pressure calculated by INEL with SRL WRAP results overlaid. The WRAP results coincide with the INEL plots. The initial rate of depressurization is low because pressure control valves tend to counteract the drop by reducing the steam line flow. The steam line flow is cut off at 4.2 sec causing a temporary rise in pressure. At 7.5 sec, pressure begins to drop when the jet pumps uncover; lower plenum flashing causes a brief rise in pressure at 8 sec. The recirculation line uncovers at 9.9 sec, and the resulting large steam flow coming out of the break causes a rapid depressurization.

Table 3

Major Event Summary for the BWR/4 Analysis

Time (Sec)	Event
0.001 to 0.002	Break opened, pump power off, feedwater tripped
0.7	SCRAM initiated on overpressurization of the containment
1.0 to 4.0	Feedwater stopped
1.5 to 4.2	Steam line closed
4.3	Upper downcomer emptied
7.1	Jet pumps uncover
9.9	Recirculation lines uncover
13.6	Dryout predicted at hottest axial core location (end of blowdown for MOXY-EM calculation)
38.0	Calculation terminated

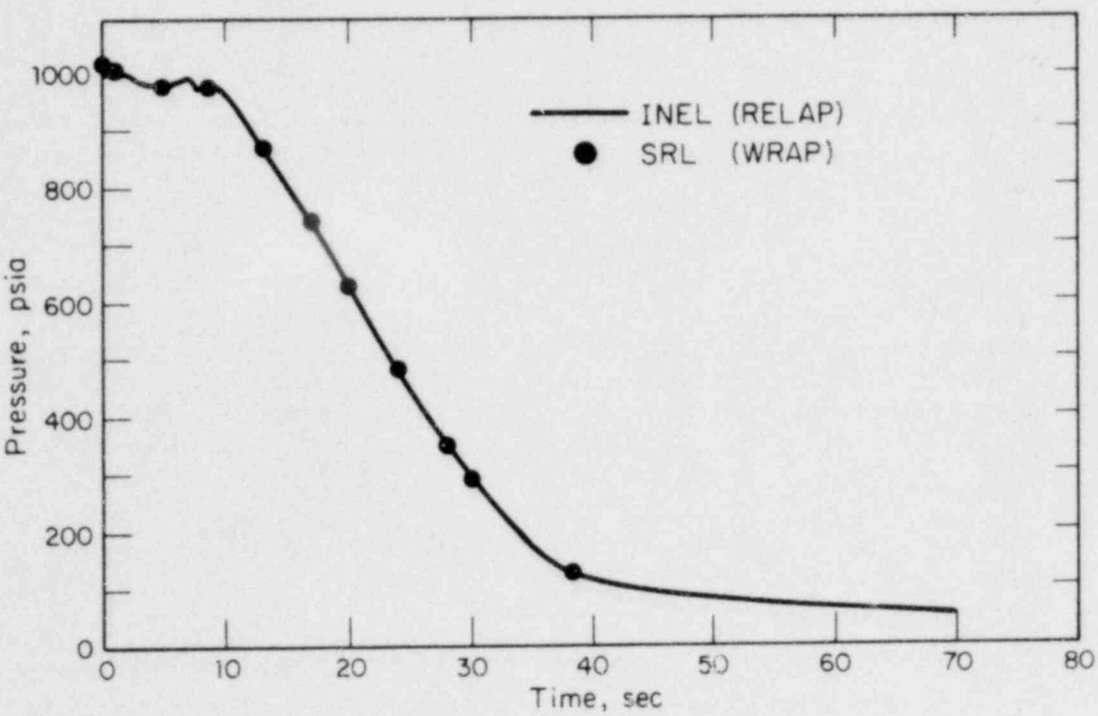

FIGURE 17. BWR/4 Calculated Steam Dome Pressure

The vessel side break flow calculated by INEL with WRAP results overlaid is shown in Figure 18. Again, WRAP results fall on the plot of the INEL output. At the time of the break, the flow increases to about 26,500 lbm/sec and then remains greater than 24,000 lbm/sec until the recirculation line uncovers at 9.9 sec. Then, the flow at the break becomes a high quality two-phase fluid, and the mass flow rate drops sharply.

The WRAP-EM and INEL calculations for jet pump flows were essentially the same. As shown in Figure 19, there are some minor differences in the inlet core flows shortly after 8 sec when the lower plenum flashes. Flashing causes rapid oscillations in the core flow, and the differences in the two calculations are probably caused by minor time step differences. The differences in core flow did not affect the overall agreement between the two calculations, as is evident from the comparisons of calculations for other system variables.

The INEL calculated slab surface temperature for the hot channel rod at an elevation about 100 in. from the core bottom, with WRAP results overlaid, is shown in Figure 20. WRAP results matched the INEL output. Although the WRAP calculation was stopped at 38 sec, just slightly before RELAP computed the peak temperature, the subsequent MOXY analysis was not affected, since it only used results from the blowdown analysis out to 13 sec when the hot plane dried out.

The INEL calculated hot pin surface temperature at the hot axial plane, with SRL-MOXY results overlaid, is given in Figure 21. The SRL results coincided with the INEL output plot for the first 100 sec. Beyond that point, SRL-MOXY surface temperatures are 4 to 5°K higher. The difference is probably associated with the onset of cladding plastic strain or onset of cladding rupture in a nearby pin. The abrupt change in the slope of the surface temperature at about 135 sec occurs one time step (2.5 sec) later in the SRL calculation. In this interval, the surface temperature rises 4 to 5°K, as calculated by WRAP-MOXY.

4.3 BWR/6 LOCA

4.3.1 Background

NRC selected a LOCA calculation for the current boiling water reactor design designated BWR/6 as part of the WRAP-BWR-EM verification study. The LOCA selected was a double-ended guillotine break in the suction line of a recirculation loop. The General Electric Company (GE) had analyzed the same LOCA for a generic analysis and NRC supplied SRL with some GE results[21] to provide a frame of reference for evaluating the WRAP-BWR-EM calculations.

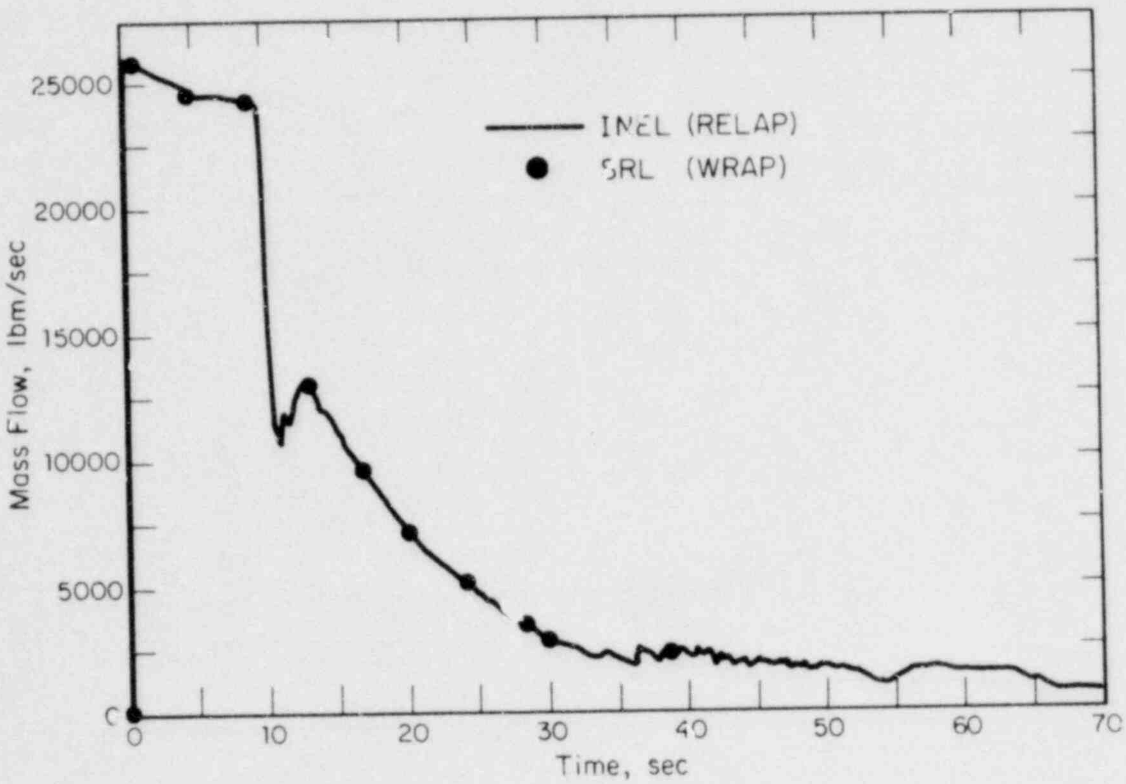

FIGURE 18. BWR/4 Calculated Vessel Side Break Flow

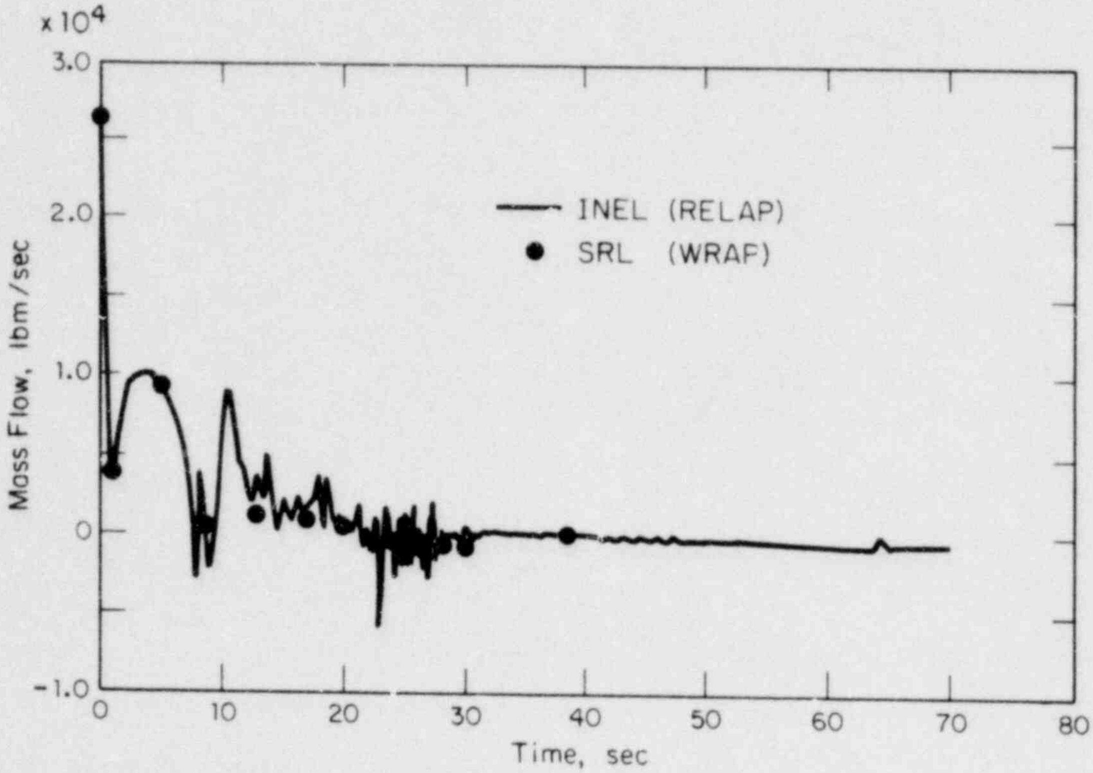

FIGURE 19. BWR/4 Calculated Core Inlet Flow

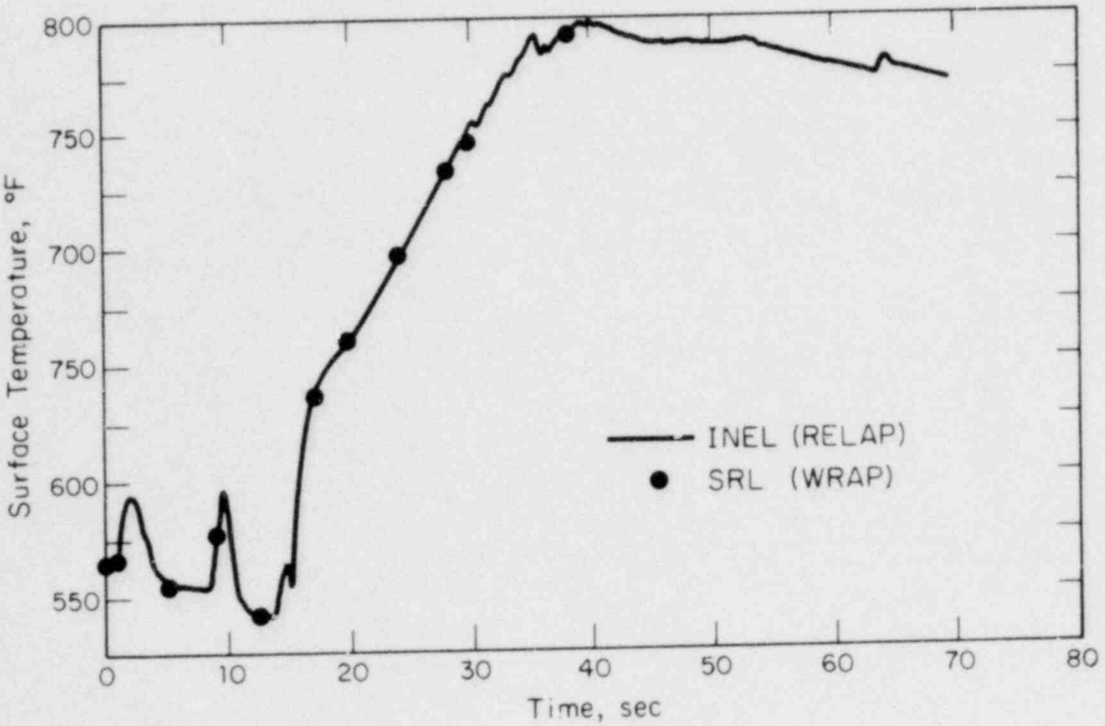

FIGURE 2C. BWR/4 Calculated Hot Channel Surface Temperature

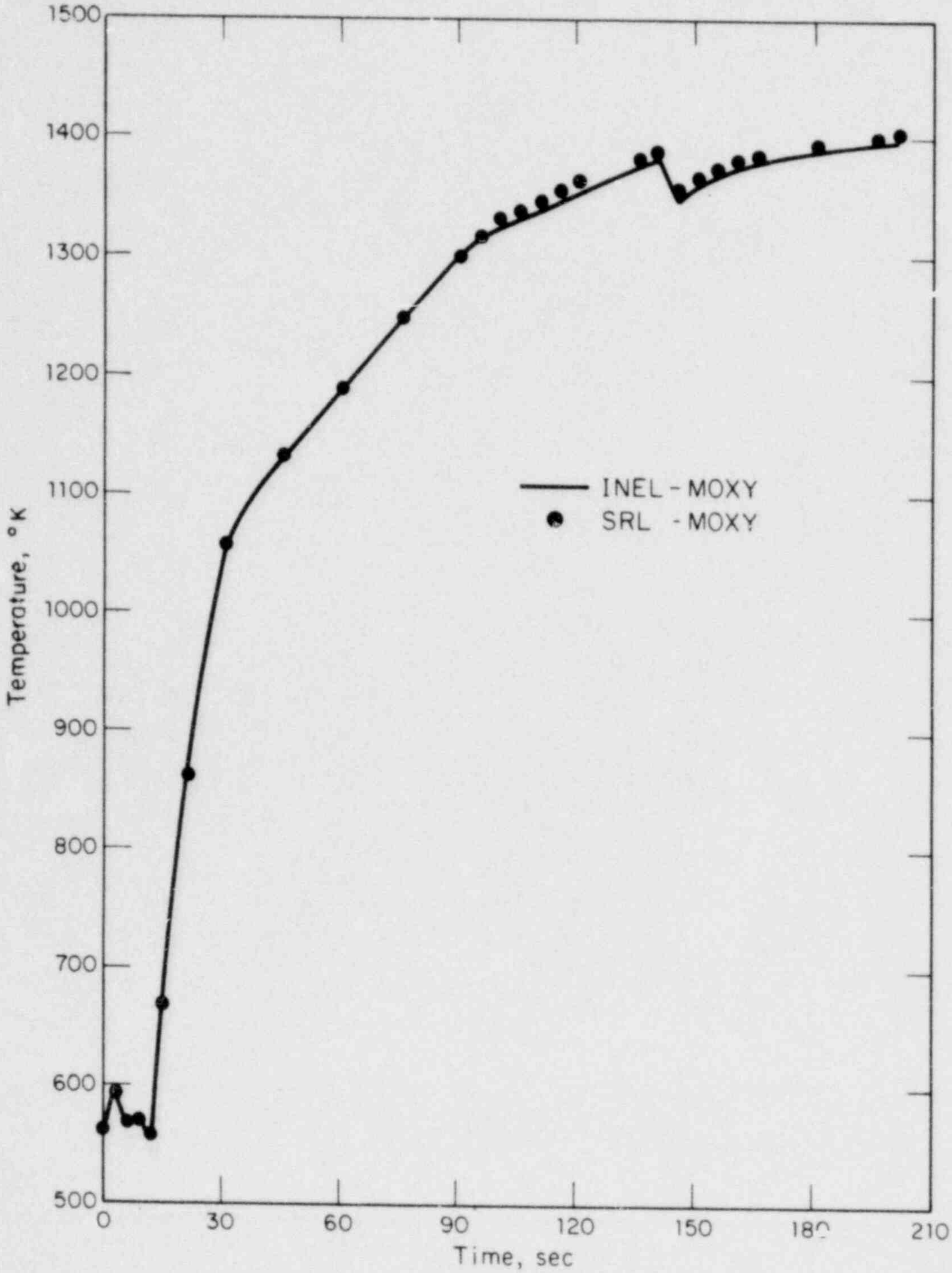

FIGURE 21. Computed Surface Temperature of the BWR/4 Hot Rod

Detailed information on all of the GE input and code models was not available to SRL at the time of the calculation, so it was anticipated that there would be some differences in the results of the two analyses.

This analysis exercised the full transient analysis capability of the WRAP-BWR-EM system. Initial fuel conditions were calculated by CAPCON; WRAPIT and WRIN were used to set up the transient calculation; the blowdown was calculated using TWRAM; and, the behavior during refill and reflood was calculated using NORCOOL.[13] MOXY[11] was used for the hot plane analysis. A steady state was not computed using the BWR steady state module in WRAP since INEL had already established steady state input parameters. However, a supplementary calculation with WRAP yielded a steady state similar to the INEL input.

The nodalization for the blowdown calculation and most of the code input were taken from a RELAP4/MOD5 Best Estimate (BE) deck developed by INEL.[22] The fuel burnup and power shapes were supplied by NRC.

4.3.2 Summary and Conclusions

WRAP successfully computed the BWR/6 large break LOCA from fuel parameter initialization during normal reactor operation to the end of reflood signalled by hot plane quench. The results were reasonable, and the behavior in the various parts of the system was consistent.

The blowdown portion of the transient (calculated using TWRAM, a reprogrammed version of RELAP4/MOD5[4]) ended at 90 sec; the subsequent NORCOOL calculation predicted a hot plane quench time of 216 sec. The MOXY module calculated a PCT of 1765°F just prior to quenching of the hot plane.

WRAP and analogous GE calculations gave similar results for system pressures and flows. However, the GE calculation gave a much larger PCT, because departure from nucleate boiling (DNB) was computed to occur much earlier in the transient. The reason for the difference in the two calculations has not yet been determined because all the input data and code output for the GE analyses are not available to SRL at this time. The difference in time of DNB is thought to be due to a combination of two effects: 1) different calculated core flows just prior to the time of DNB in the GE analysis, and 2) the form of the critical heat flux correlations for low or reverse core flow used in the GE calculation.

4.3.3 Input

The nodalization used for this study (Figure 22 and Table 4) was developed at INEL for a best-estimate study of the BWR/6.[22] No modifications were required to meet EM criteria. The core region is composed of two parallel channels, one representing the hot assembly flow path and the other representing the remainder of the core. Two stacks of heat slabs are adjacent to the hot channel to better model the detailed thermal behavior of the fuel. One stack is associated with the fourteen hottest pins in the hot bundle, while the other stack represents the 48 remaining active pins (Figure 23). The initial system hydraulic parameters were taken from Reference 22. The initial power, steam dome pressure, and system flows are given in Table 5.

To account for two-phase flow, bubble rise models were used in the steam dome, downcomer, core bypass, lower plenum, guide tube regions, and in containment. The bubble velocity was set to 3.0 ft/sec in all regions except the lower plenum which was set at 0.0 ft/sec. In addition, vertical slip was used at the internal core junctions.

Trips, based primarily on the state of the system, were used to initiate scram, cut off normal operating flows, and trigger the emergency core cooling and safety systems. The systems controlled by trips were:

- <u>Scram</u> - A high drywell pressure signal* is assumed at the time of the accident so a scram is initiated immediately.

- <u>Feedwater</u> - The feedwater is assumed to ramp to zero in one second.

- <u>Steam Line</u> - The steam line flow is dependent on the steam dome pressure and is controlled by a pressure-dependent fill table. The main steam line isolation valve (MSIV) begins closing when the mixture level in the shroud drops to 10 feet above the top of the core and takes three seconds to close.

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* A high drywell pressure (2 psig) is the initiating signal for several of the safety systems. Since GE does not model containment, and the WRAP nodalization will not accurately treat the large, inhomogeneous volume correctly, a high pressure signal is assumed to occur at the time of accident initiation. This is a nonconservative assumption (time) and sensitivity analyses should be conducted to determine the quantitative effect.

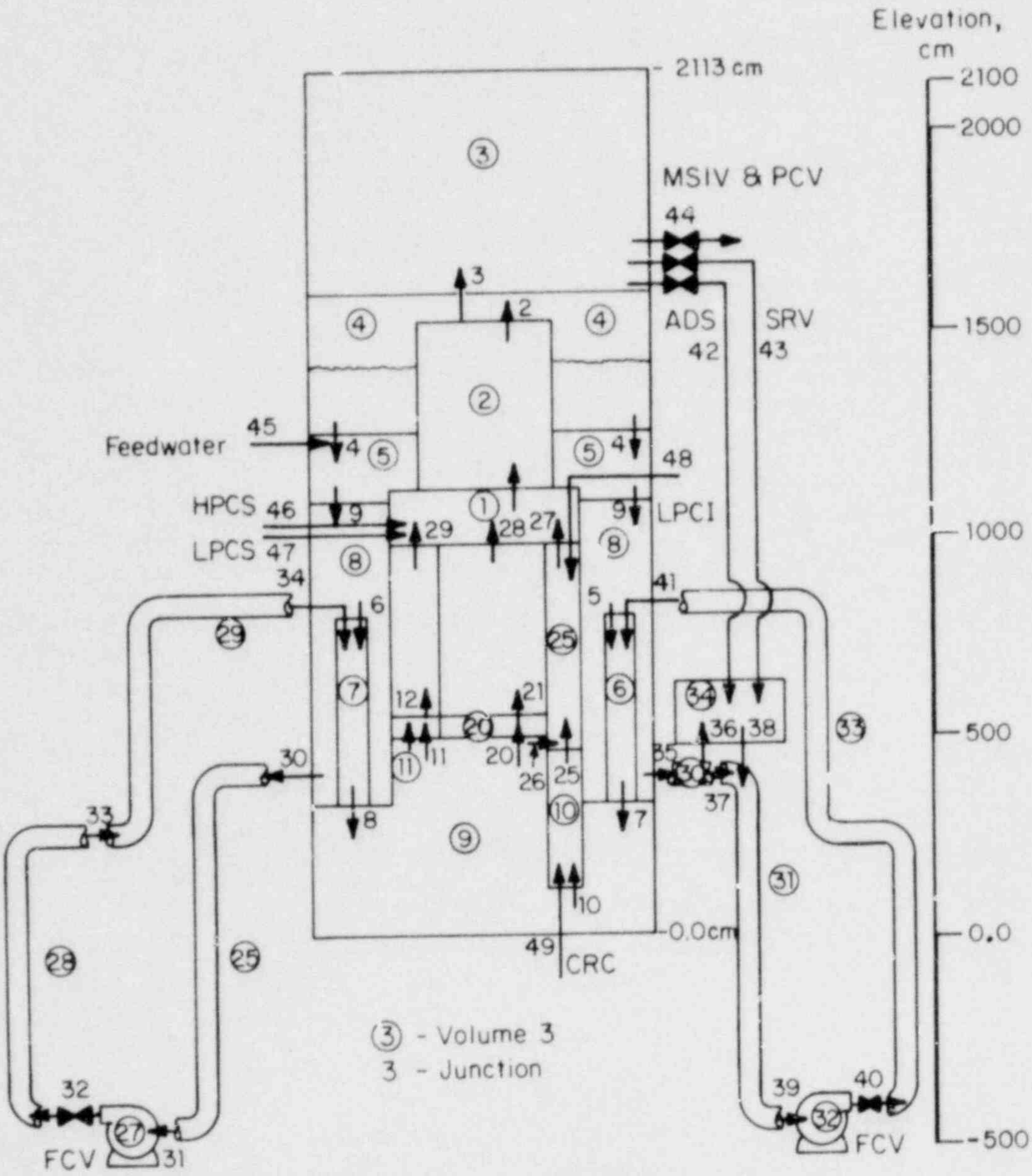
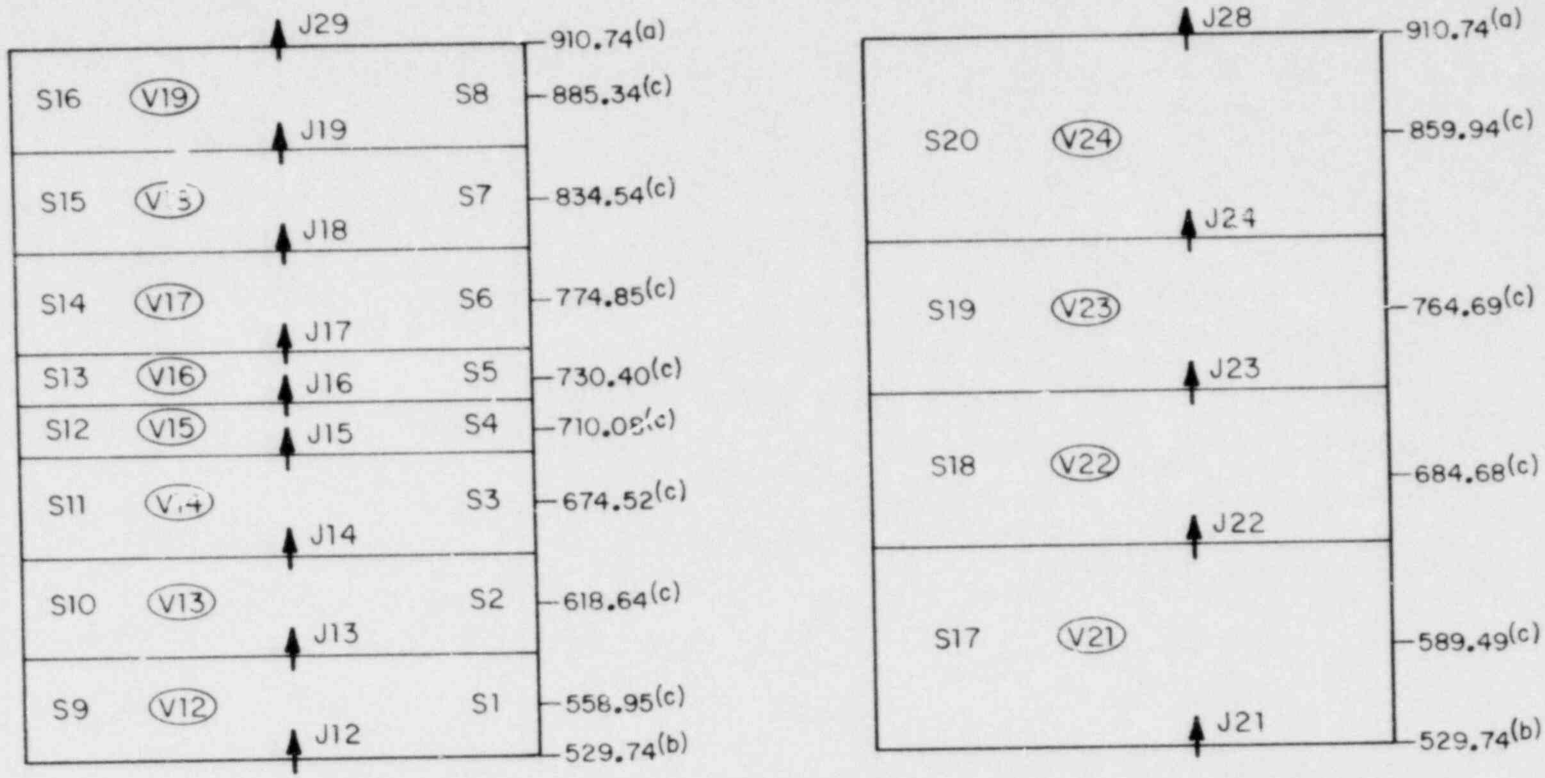


FIGURE 22. BWR/6 System Nodalization for WRAP

Table 4

BWR/6 Large Break Model

Volume Number	Volume Description
1	Upper plenum
2	Steam separator
3	Steam dome
4	Upper part of downcomer region
5	Middle part of downcomer region
6	Broken loop jet pump
7	Intact loop jet pump
8	Lower part of downcomer region
9	Lower plenum
10	Guide tubes
11	Nonheated portion of hot core channel
12	Nonheated portion of average core channel
13-20	Hot channel core volumes
21-24	Average channel core volumes
25	Core bypass
26	Intact loop recirculation suction line
27	Intact loop recirculation pump
28	Lower section of intact loop recirculation discharge line
29	Upper section of intact loop recirculation discharge line
30	Break node-volume containing vessel side break junction
31	Broken loop recirculation suction line - contains pumpside break junction
32	Broken loop recirculation pump
33	Broken loop recirculation discharge line
34	Containment



(a) Top of active core (cm) relative to reactor system

(b) Bottom of active core (cm) relative to reactor system

(c) Volume and slab midpoint elevations (cm)

(d) V = Volume, S = Slab, J = Junction

FIGURE 23. BWR/6 Active Core Volumes and Heat Slab Nodalization[d]

Table 5

BWR/6 Initial Conditions

Condition	Value
Initial Reactor Power	2952 MWth
Hot assembly peak linear power	8.55 kW/ft
Hot pin peak linear power	9.03 kW/ft
Steamline Flow	3459.2 lbm/sec
Feedwater Flow*	3451.7 lbm/sec
Steam Dome Pressure	1040 psia

* The total system input flow is 7.5 lbm/sec higher. The additional input flow is the control rod drive flow.

- Control Rod Drive (CRD) - The CRD flow is 7.5 lbm/sec at the initiation of the accident and ramps to zero flow at 5.0 sec.

- Recirculation System - Recirculation pumps trip off at accident initiation. The recirculation loops also have flow restrictors which reduce the flow area to 78% of the original area over a period of 120 sec starting at the time at which the temperature difference between the steamdome and the respective loop suction line becomes 8.5°F.

- High Pressure Core Spray (HPCS) - Assumption of a high drywell pressure signal at the time of the accident causes the HPCS to actuate at that time. In order to model pump start-up time, there is a 27 sec delay before actual injection begins.

- Low Pressure Coolant Injection (LPCI) - As with the spray systems, the LPCI is actuated at the time of the accident and has an accompanying 27 sec delay time. Injection will not begin until system pressure is less than 187 psi. Only 2/3 normal flow is allowed in this problem since one LPCI pump (out of the three available) is assumed to fail.

- Automatic Depressurization System (ADS) - The ADS actuates when the mixture level outside the shroud drops to about 1.5 ft above the core. The ADS has a 2.0 min delay time.

The ECCS flows as a function of system pressure are given in the Appendix. Representative fuel data for use in GAPCON was obtained from NRC. The criterion used for exposure was that the reactor operates for a time which produces a burnup of 2000 MWD/MT at the hottest node of the lowest power pin in the hot assembly. At a power of 2952 MWT (102% of nominal power), an operating time of 51 days satisfied the given criterion.

A chopped cosine axial power profile with an axial peaking factor of 1.4 was used. Average linear power densities were 6.10 kW/ft (core average excluding the hot assembly), 8.41 kW/ft (48 lowest powered active pins in the hot assembly), and 9.03 kW/ft (14 highest powered pins in hot assembly). An EM calculation was specified by selecting the following models:

- Critical Flow - The recommended EM critical flow models of Henry-Fauske[18] for subcooled flow and Moody[7] model for saturated flow were used. The multiplier for these models was 1.0.

- Decay Heat - The ANS standard with a multiplier of 1.2 was used.

- Heat Transfer - The Groenveld 5.9[19] film boiling correlation was used for core slabs. The Hench-Levy[17] GE critical heat flux correlation was used for the two heat slab stacks representing the hot fuel bundles.

In addition, WRAP-EM heat transfer logic precludes the return to nucleate boiling once the CHF has been reached and the clad temperature has exceeded the coolant saturation temperature by more than 300°F.[4] This is slightly different from the 10 CFR 50 Appendix K requirement which states that return to nucleate boiling is not allowed once CHF is reached.[1]

- Pin Swelling and Flow Blockage - Data were derived from Reference 23. Values were selected to cause rupture for the minimum stress value at a given temperature. Based on flow blockage data for fast and slow ramp heating rates, values were selected which gave the greater blockage for each burst temperature.

4.3.4 Results

Selected thermodynamic properties of the reactor system as calculated by WRAP-BWR-EM are discussed in the following paragraphs. All results are reasonable and the behavior in various parts of the reactor system is consistent. The analysis required 200 min of CPU time on an IBM 360/195 computer. The bulk of the CPU time, 140 min, was for the blowdown analysis (TWRAM), and the reflood analysis (NORCOOL) took 30 min.

The pressure behavior of the system is illustrated by the steam dome pressure in Figure 24. The immediate pressure decrease is due to the large mass and energy loss out the break. The MSIV began closing at 4.5 sec and is fully closed at 7.5 sec resulting in a pressure increase. At 11.5 sec, the recirculation suction line uncovers, producing an open pathway for steam to escape resulting in an immediate pressure drop. At 13 sec, the lower plenum flashes, causing a slight decrease in the depressurization rate due to decreased quality at the break. The depressurization rate remains constant until the system drops below about 250 psia. The pressure continues to drop, but at a continually decreasing rate, until the blowdown calculation ends at 90 sec due to end of critical flow at the break.

The break flows are shown in Figure 25. The pump side flow decreases rapidly due to the impedance of the recirculation pump and the jet pump drive junction (area = $0.35 ft^2$) which limits

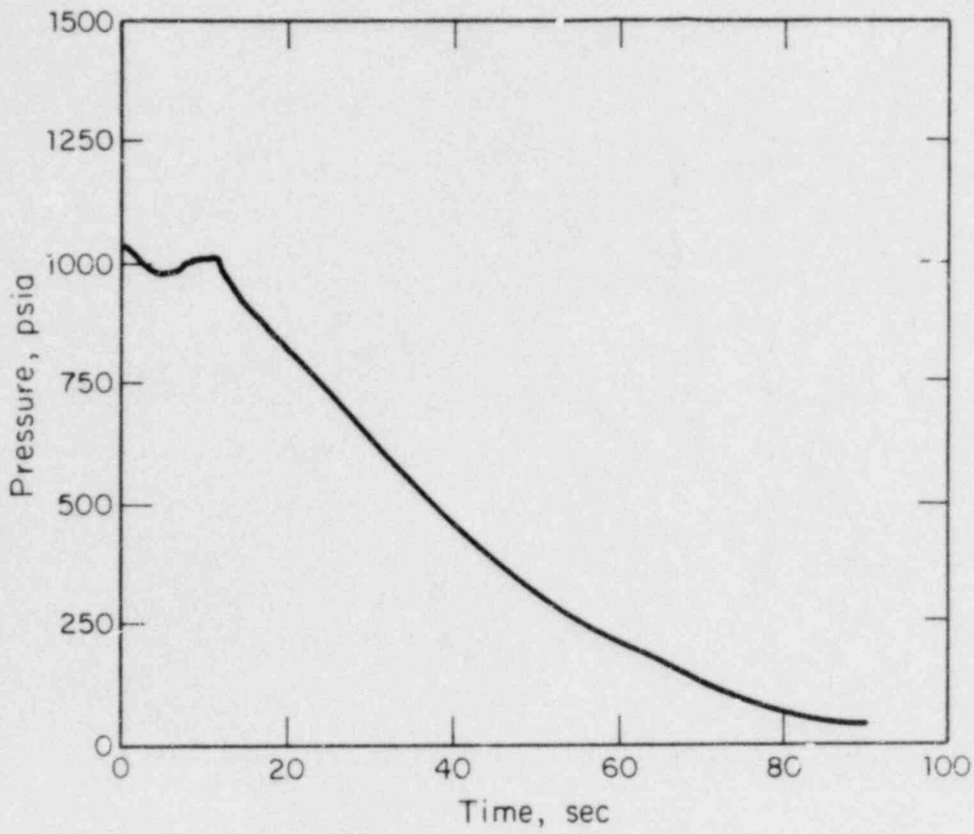


FIGURE 24. Steam Dome Pressure BWR/6 Blowdown Analysis

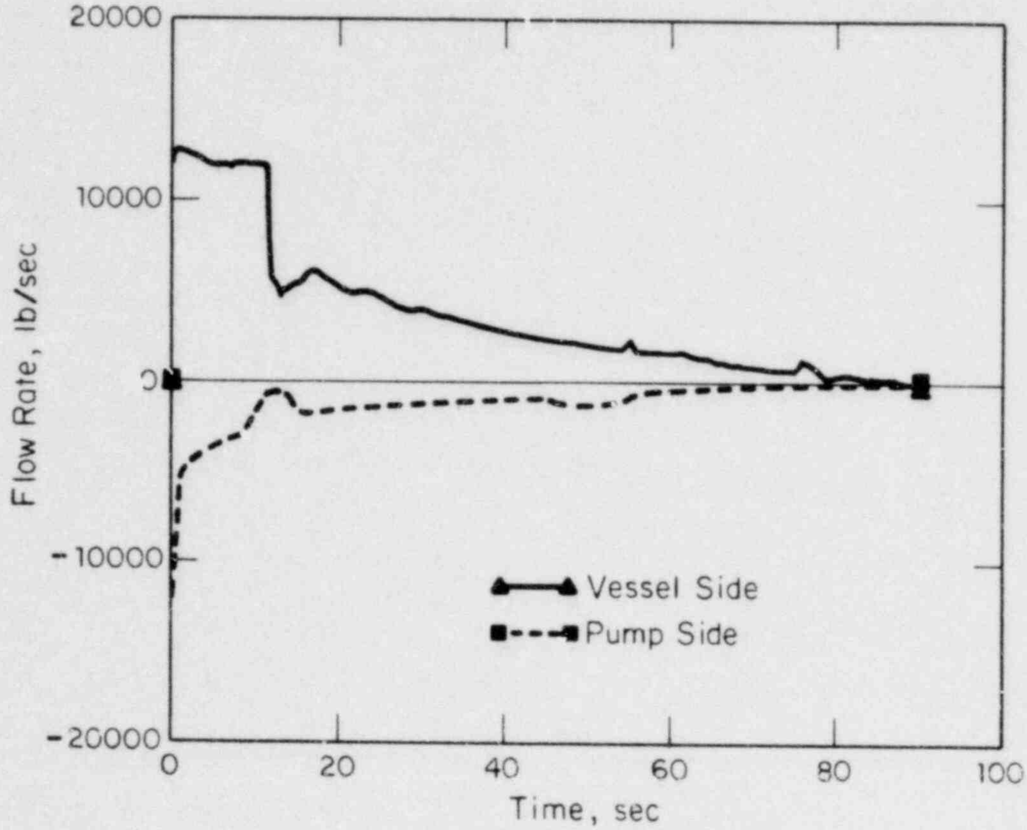


FIGURE 25. Break Flows BWR/6 Blowdown Analysis

the flow into the discharge line. At approximately 1.0 sec, the jet pump drive junction chokes and that flow becomes the effective break flow since the water in the discharge line at the time of the accident flowed out the break within the first second of the transient. At 9.0 sec, the jet pump saturates causing two-phase fluid to exit the break and causing a noticeable drop in the break flow on the pump side. Lower plenum flashing at 13 sec produces a small increase in the break flow which then slowly decreases during the remainder of the transient.

By contrast, the vessel side break flow remains high as water experiences no impedance emptying directly from the downcomer out the break. Not until 11.5 sec, at which time the lower downcomer has emptied to the suction line level, does the break flow suddenly decrease. Lower plenum flashing produces a small temporary increase in the break flow.

Figure 26 shows the core inlet flow. The sudden flow decrease due to the loss of broken-loop jet pump flow stops at approximately 1.0 sec as the flow out the break becomes critical. The flow recovers slightly and then slowly decreases until 9 sec when the jet pumps uncover causing them to flash. The flashing causes a flow spike at the core inlet which rapidly dissipates. The core flow then remains low yet positive until 13 sec when the lower plenum flashes. After this second flashing occurs, the flow stays positive until approximately 47 sec. The flow then oscillates around zero until 66 sec when the LPCI begins injection and small positive core flow is restored.

The behavior of the clad temperatures for the heat slab stack representing the 14 hottest pins in the core is illustrated in Figure 27. The top of the pin exhibits a very early DNB which lasts until core flow oscillations rewet it at approximately 6.0 sec. After jet pump flashing and the greatly reduced core inlet flow at 9.0 sec, a second DNB occurs at the top of the pins lasting until lower plenum flashing at 13 sec. The rest of the figure illustrates how the core slowly uncovers following lower plenum flashing as each axial level, except the bottom, experiences the sudden temperature jump associated with dryout. The highest powered plane in the core uncovers at 48 sec.

The reflood analysis was performed by the NORCOOL[13] module. The nodalization is illustrated in Figure 28. The HPCS is modeled as a horizontal injection in the upper node of the upper plenum at a temperature of 100°F. The LPCI, also at 100°F, is injected at the top of the bypass region. At the beginning of the flood calculation, all regions except the portion of the lower plenum containing water are assigned void fractions of 1.0.

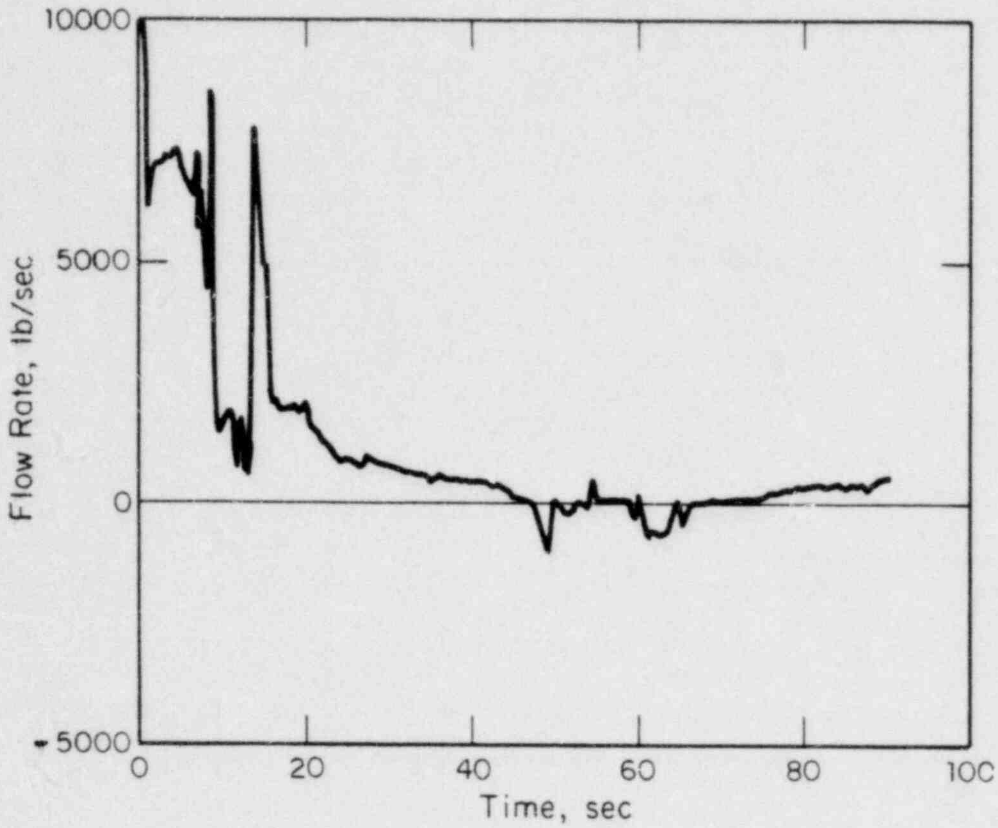


FIGURE 26. Core Inlet Flow BWR6 Blowdown Analysis

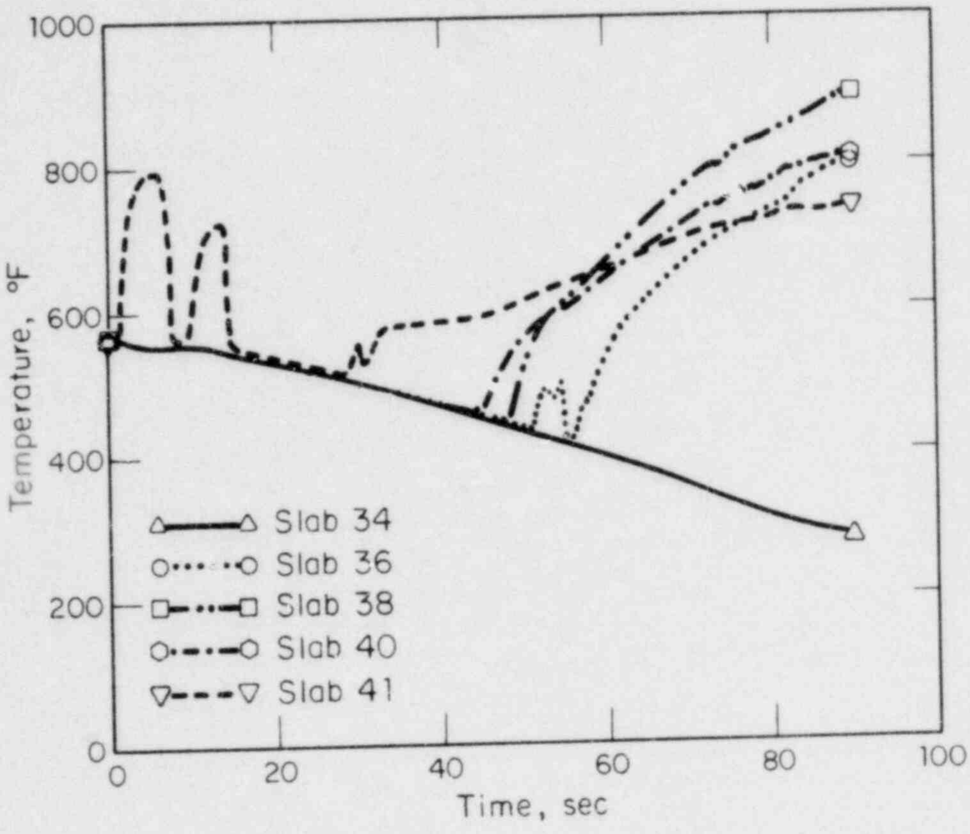
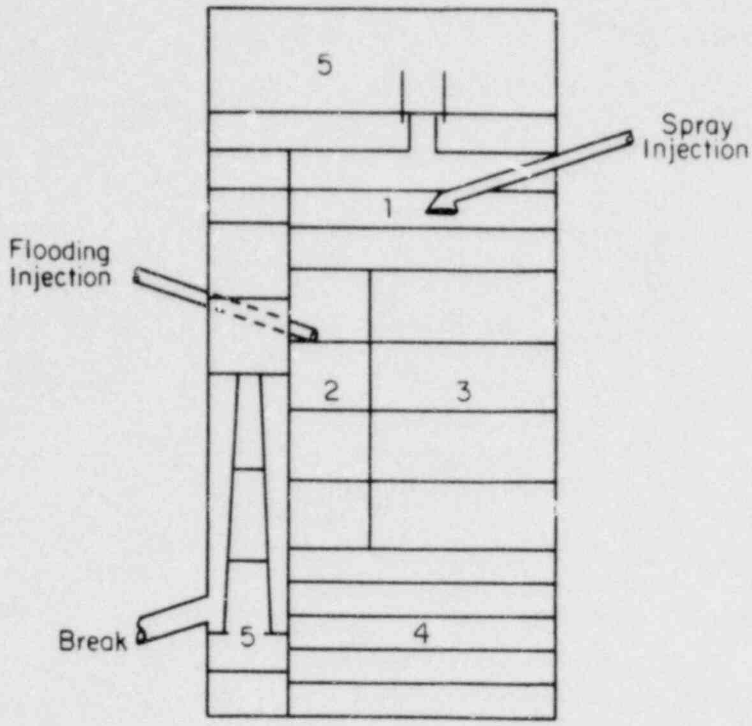


FIGURE 27. Hot Pins Clad Temperatures BWR/6 Blowdown Analysis



Region Number	
1	Upper Plenum Steam Separator
2	Bypass
3	Core
4	Lower Plenum
5	Steam Dome, Downcomer and Diffuser

FIGURE 28. NORCOOL Nodalization for the BWR/6

The time-dependent behavior of the quench front, the parameter of interest in the calculation, is illustrated in Figure 29. The heated portion of the core begins at an elevation of 17.4 ft. Rising rapidly for the first 45 sec, the front moves past the cooler portion of the core. There is a distinct reduction of the quench front velocity at 135 sec as the front reaches the highest powered of the four axial segments. However, the rise of the front continues, and the rate of rise increases as the cooling of this segment (Figure 30) enhances the quenching process. NORCOOL uses one stack of heat slabs to model the entire core, so that the maximum clad temperature (Figure 30) is for an average powered bundle; not the hottest bundle. However, a drop in temperature which occurs well before the actual quenching, because of improved heat transfer during the middle stages of reflooding, applies to all bundles. This drop in temperature before quenching will not be seen in the hot plane analysis because of EM criteria.

The hot plane analysis for the WRAP-BWR-EM system was performed with the MOXY code.[11] Input included the reactor power, surface heat transfer coefficients, and fluid temperatures at the hot plane as a function of time as calculated in TWRAM up until the time of the end of lower plenum flashing. After this time, MOXY uses its own internal power table and assigns heat transfer coefficients as specified in Appendix K.[1] Thus, the ultimate PCT is sensitive to the time which is designated as end of lower plenum flashing. For this case, the time of 44.0 sec was input to MOXY. As indicated in Figure 27, this time is slightly conservative since the hot plane does not uncover until 48 sec.

Other input includes the results of 35 GAPCON[14] calculations, one for each active rod in the half bundle including all rods along the diagonal. The hot plane quench time was calculated by NORCOOL to be at 216.0 sec.

For conservatism, the inactive rod and canister quench times used in MOXY to predict radiative heat transfer were set to 216.0 sec also. The PCT predicted by MOXY was 1765°F at 215 sec. No rupture of the clad is indicated.

As part of the verification, WRAP results were compared to available results from a GE analysis.[21] Although every effort was made to ensure that the two analyses used the same input data, some key input data for the GE analysis (e.g. initial power level, power peaking, and fuel burnup) were not available to SRL at the time the WRAP analysis was started. Therefore, differences between the results of the two calculations may be due to either input data or computational models. Comparisons of calculated

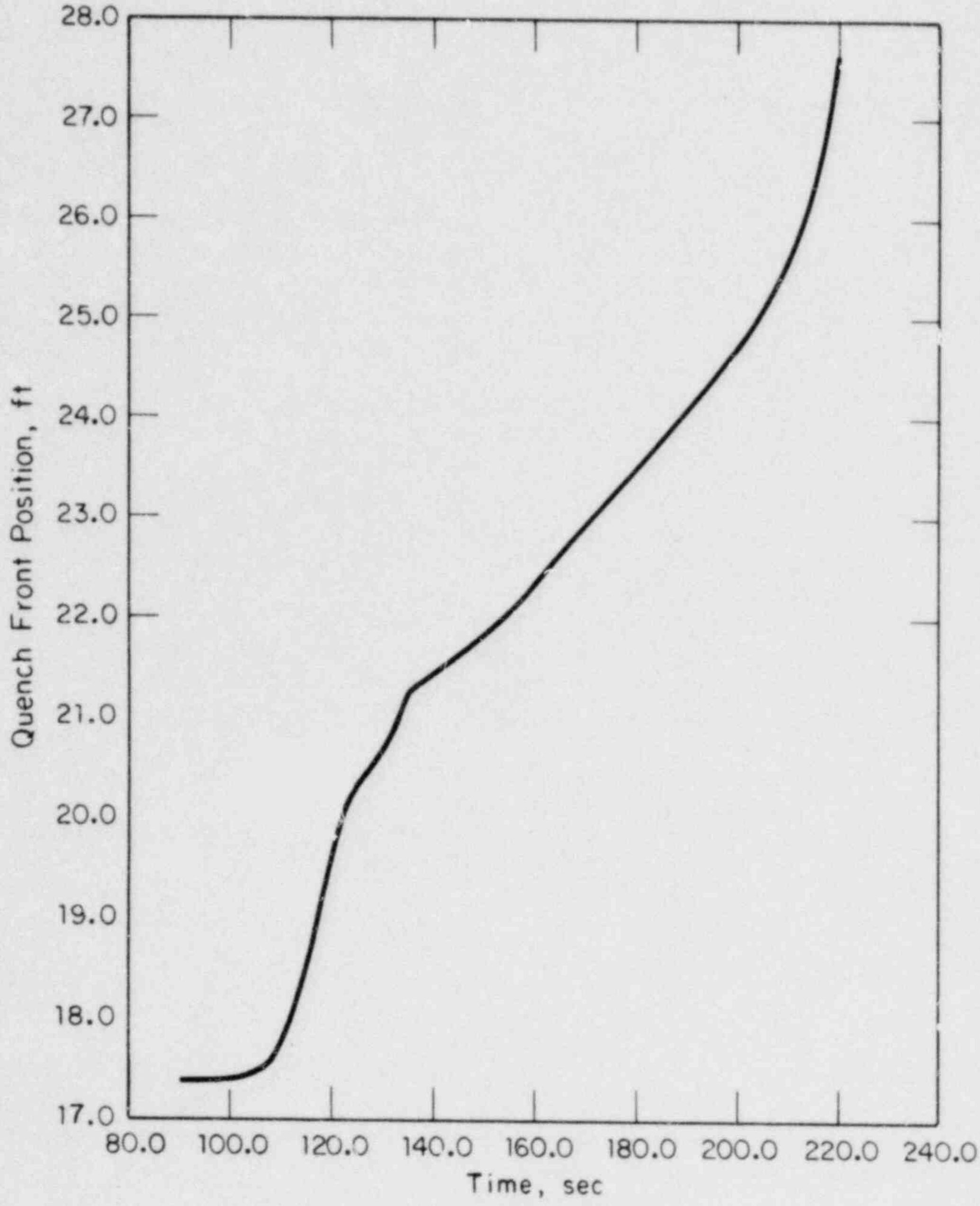


FIGURE 29. BWR/6 Quench Front Position vs Time (NORCOOL)

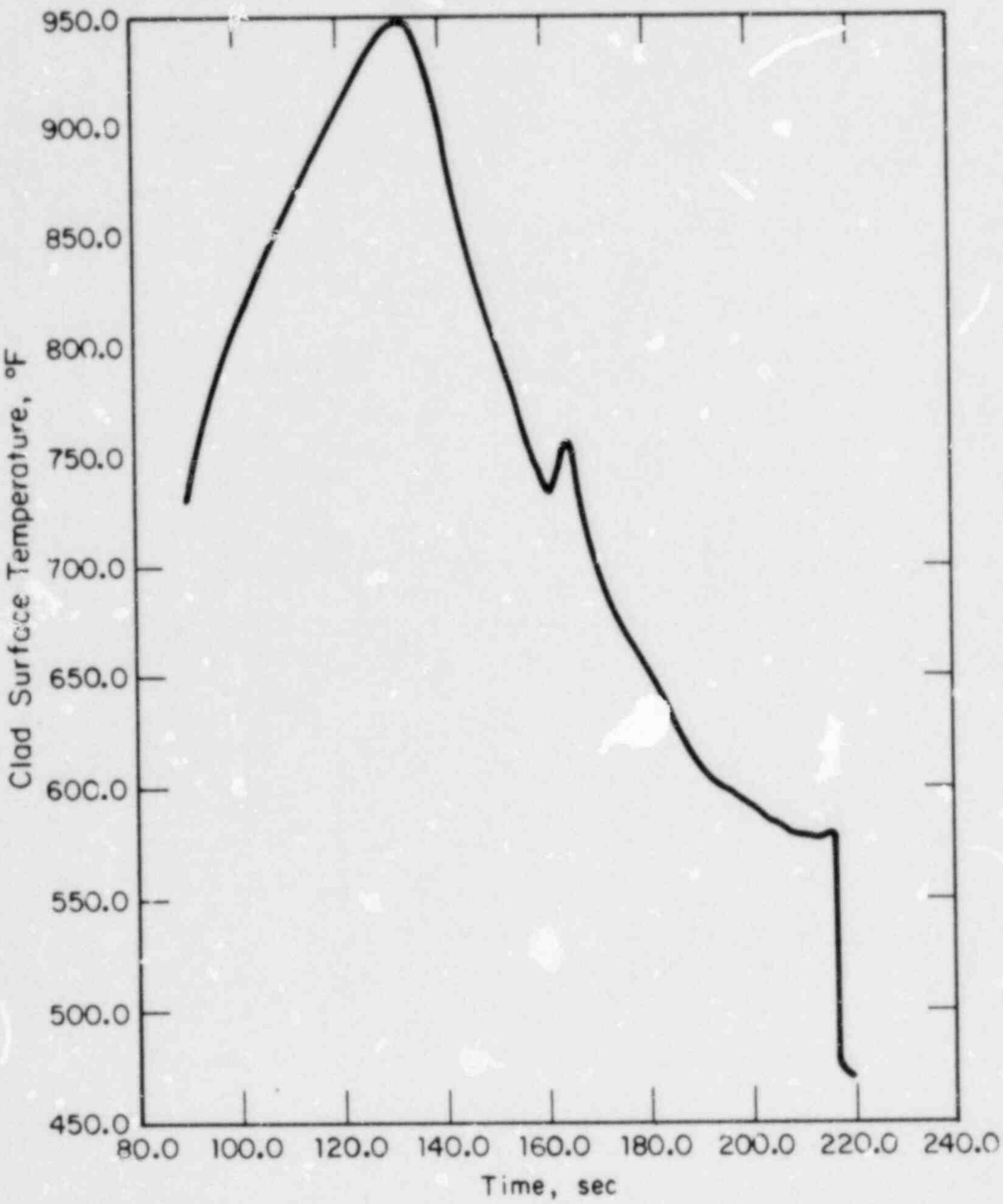


FIGURE 30. BWR/6 Clad Surface Temperature During Reflood (NORCOOL)

pressure, core inlet flow, and clad surface temperature presented in the following paragraphs are based on available data.

The core average pressures for the two calculations are shown in Figure 31. The GE repressurization after MSIV closure at 4.5 sec is faster than the WRAP results and may be due to a higher power.* Also, GE may conservatively assume the MSIV closes instantaneously while the WRAP input included a 3 sec closing time so the valve is not fully closed until 7.5 sec. The GE pressure calculation implies that the recirculation suction line uncovers at approximately 10 sec which is 2 to 2.5 seconds earlier than the WRAP analysis shows. It is known that in the GE nodalization, a 0.1 ft^2 reactor water cleanup line is modeled which evidently couples the intact and broken loop recirculation suction lines.[16] This extra line increases the water lost from the intact loop during the break which can hasten the downcomer mass depletion. This may also be one reason for the faster depressurization rate seen in the GE analysis after 15 sec (Figure 31).

Figure 32 shows the respective core inlet flow rates. At 8.0 sec, the WRAP analysis shows a sharp flow increase due to the intact loop jet pump flashing that results in a discharge flow spike. The GE calculation, in sharp contrast, shows a precipitous drop in flow rate which eventually becomes negative for a 2-sec period (9 to 11 sec). This may be due to different treatments of the jet pumps in the respective codes. The GE results indicate that as the jet pump uncovers, the discharge flow stops and the core flow reverses since the driving force is absent. The WRAP calculation indicates an initial surge from flashing in the jet pump and then the reduction in discharge flow with the core flow remaining positive. Due to the faster depressurization noted in Figure 13, the GE calculation indicates lower plenum flashing approximately 2 sec earlier than the WRAP analysis and a substantially higher flow spike.

The difference in the GE- and WRAP-calculated PCT is shown in Figure 33. The obvious deviation at 10 sec indicates that GE calculates DNB at the highest powered plane earlier than WRAP. In addition, lower plenum flashing does not seem to rewet the plane, indicating the presence of a heat transfer lockout in their fuel

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* After the WRAP analysis was completed, NRC forwarded new GE power data. GE had run their analysis using an initial power which was 105% of nominal power, while the WRAP calculation was run at 102% of nominal power.

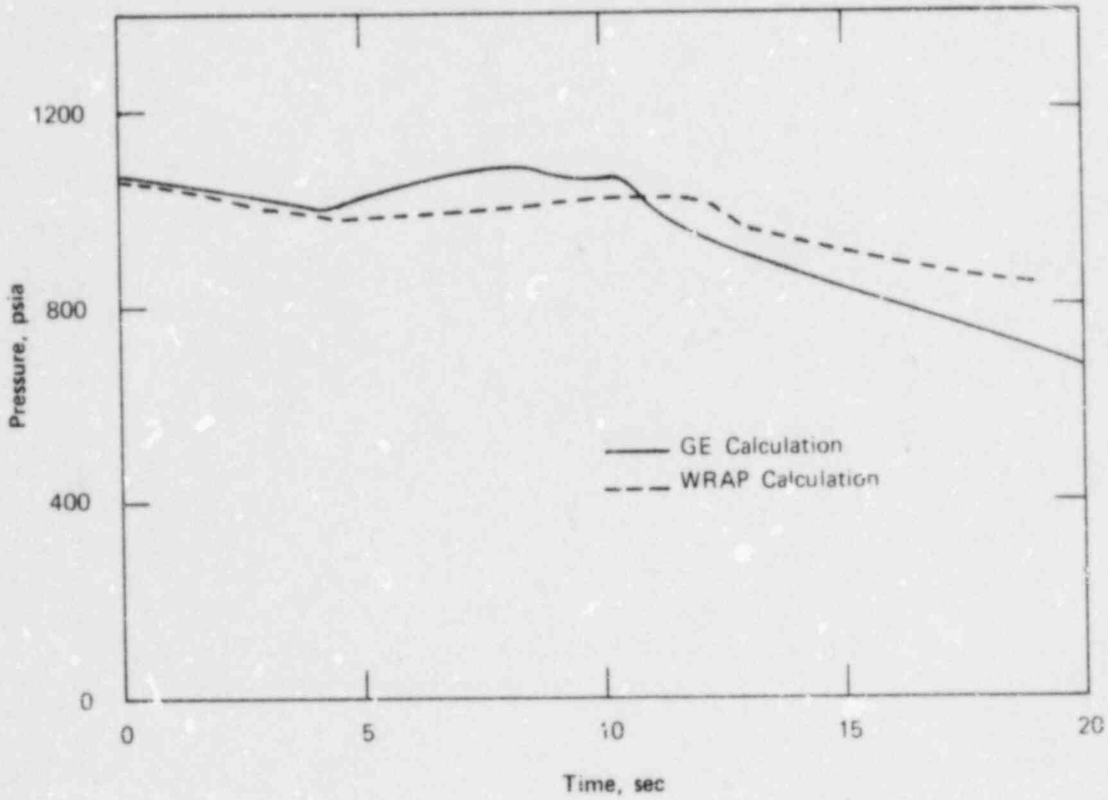

FIGURE 31. Core Average Pressure (BWR/6 Large Break)

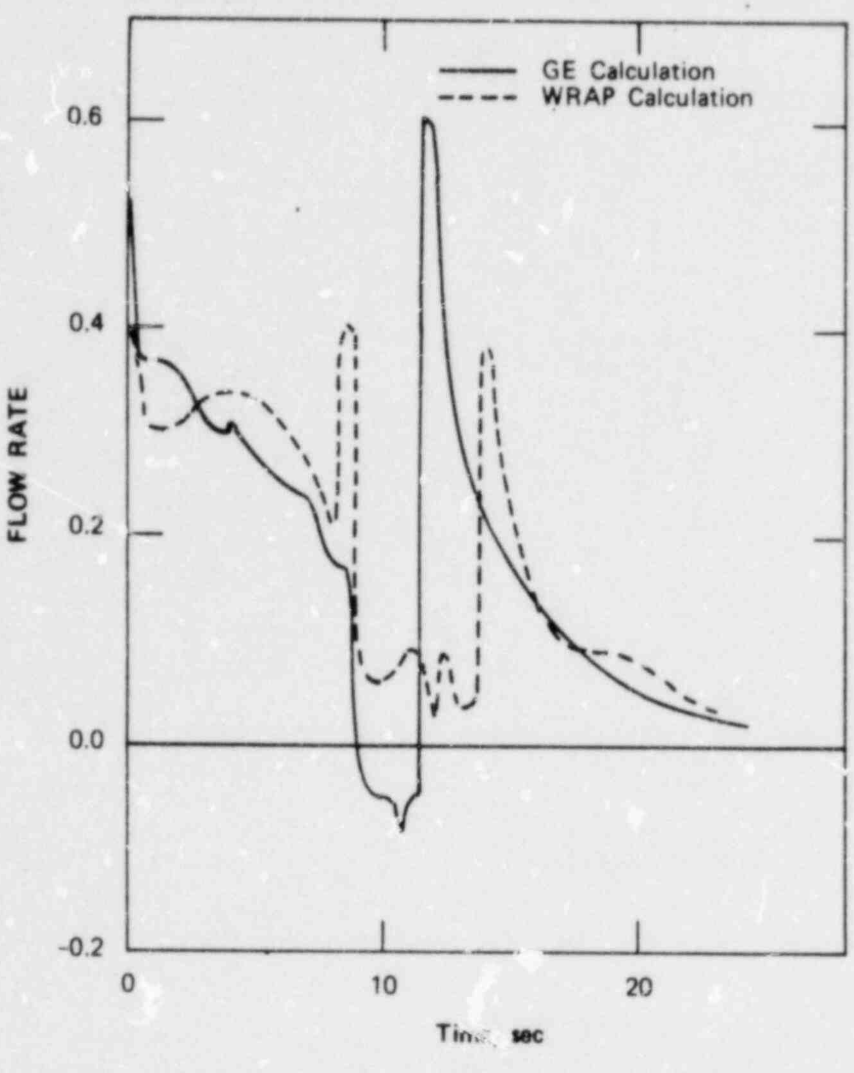

FIGURE 32. Normalized Core Inlet Flow (BWR/6 Large Break)

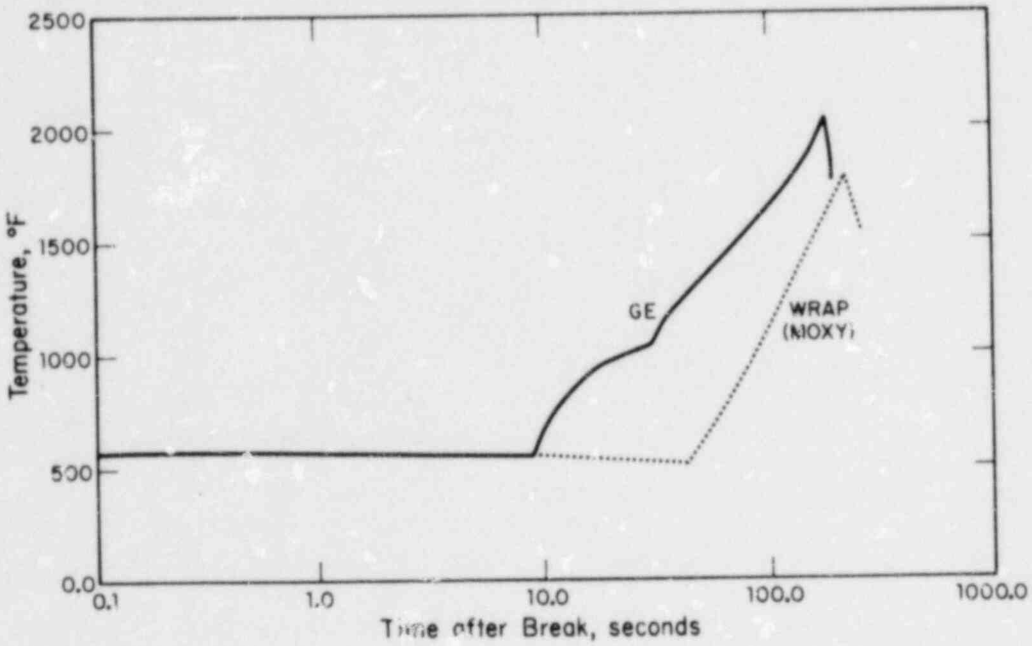

FIGURE 33. BWR/6 Peak Clad Surface Temperature

analysis program.* The WRAP (MOXY) analysis shows that the highest powered plane remains in nucleate boiling until well after lower plenum flashing. DNB is not reached until 48 sec at which time the hot plane uncovers and remains uncovered until quenching at 216 sec predicted by NORCOOL. The difference in temperatures at early times (about 10 sec) is thought to be due to different calculated core flows. WRAP computes a core flow spike at about 9 sec (Figure 15) while the GE analysis shows flow stagnation and reversal. This flow difference will cause a considerable variation in the heat transfer and it occurs just prior to the time when the GE analysis indicates DNB. The reverse flow seen by GE is particularly important since it is not clear how it effects the critical heat flux correlation they use. GE employs the General Electric Critical Quality (X_c) - Boiling Length Correlation (GEXL) which is apparently invalid at low mass flow rates ($<1 \times 10^5$ lb/hr-ft^2)[24,25] and reverse flow.** Thus, their calculated CHF during these flow regimes may be very low.

4.3.5 Additional Analysis

The WRAP calculation used the Hench-Levy CHF correlation since the input data required for the GEXL correlation was not available at that time. A subsequent WRAP calculation which employed the GEXL[6] correlation (using 3.0-in. heat slabs as recommended by NRC) indicated that the heat flux at the highest powered slab remained well under the CHF through the 15 reactor sec that the problem was executed. The flow remained positive through the core and the GEXL correlation was applicable. If the flow had reversed, a different correlation would be used in WRAP (the MOD7 correlation)*** which is directly proportional to the absolute value of the flow rate. Thus, if the flow is nearly stagnant, (as it must be when it reverses), the CHF becomes very low and DNB could occur.

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* The models in WRAP prevent a return to nucleate boiling only if the difference between the clad surface and the saturation temperature has exceeded 300°F.[4]

** Information received at a joint Savannah River Laboratory - Idaho National Engineering Laboratory - Nuclear Regulatory Commission meeting at Idaho Falls, ID (August 6-8, 1979).

*** Interoffice Correspondence from K. G. Andie to H. Sullivan, Additional RELAP4/WRAP CHF Correlations, (COND-3-79), Idaho National Engineering Laboratory (June 14, 1979).

4.3.6 Uncertainties

The major uncertainty in comparing the WRAP and GE analysis is that all the GE input and the GE code models were not available. To define error bounds or variations in the WRAP analysis would require parametric calculations which were beyond the scope of this study. These calculations are included in a BWR reference study. Calculations for the reference study have shown that renodalization of the downcomer, and specification of a non-zero bubble rise velocity in the lower plenum both give an earlier time for departure from nucleate boiling and higher clad temperatures for the hot plane in the core.

The only model varied during this study was the CHF correlation. The GE (Hench-Levy) and the GEXL correlation gave similar results in the TWRAM blowdown calculation.

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APPENDIX-ADDITIONAL WRAP INPUT DATA

A.1 TLTA Calculations for Tests 6007 and 6406

The normalized power tables used in the calculations to model the actual test power transients are given in Table A-1. The ECCS flows for Test 6406 are given in Table A-2. Table A-3 gives the time step specifications used for the calculations.

A.2 BWR/4

The ECCS flows used to model for the Core Sprays and the Low Pressure Coolant Injection System (LPCIS) for the BWR/4 are given in Table A-4. The time step specifications for the BWR/4 calculation are given in Table A-5. Figure A-1 shows the axial power profile used for the BWR/4 analysis.

A.3 BWR/6

The ECCS flows used to model the High Pressure Core Spray and the Low Pressure Coolant Injection for the BWR/6 are given in Tables A-6 and A-7. The time step specifications for the BWR/6 are given in Table A-8.

Table A-1

Normalized Power for TLTA Tests 6007 and 6406

	Normalized Power	
Time (sec)	Test 6007	Test 6406
0	1.0000	1.0000
1	0.9223	0.9223
2	0.7972	0.7579
3	0.6935	0.6448
4	0.6070	0.5575
5	0.5339	0.4861
6	0.4706	0.4167
7	0.4144	0.3671
8	0.3839	0.3532
9	0.3190	0.2738
10	0.2786	0.2421
15	0.1459	0.1459
20	0.0899	0.0899
25	0.0677	0.0677
30	0.0586	0.0635
40	0.0521	0.0556
50	0.0494	0.0530
75	0.0455	0.0530
100	0.0431	0.0530
150	0.0402	------
500	------	0.0530

Table A-2

ECCS Flow Tables for TLTA Test 6406

High Pressure Core Spray		Low Pressure Core Spray		Low Pressure Coolant Injection	
Time (sec)	Flow (lbm/sec)	Time (sec)	Flow (lbm/sec)	Time (sec)	Flow (lbm/sec)
0.0	0.0	0.0	0.0	0.0	0.0
27.0	0.0	78.0	0.0	86.0	0.0
28.0	0.80	80.0	0.32	90.0	0.13
30.0	0.85	90.0	0.65	100.0	0.35
40.0	0.96	100.0	0.87	110.0	0.51
50.0	1.03	110.0	1.05	120.0	0.63
60.0	1.05	120.0	1.22	130.0	0.73
70.0	1.11	130.0	1.32	140.0	0.77
150.0	1.13	140.0	1.38	150.0	0.79
230.0	1.12	150.0	1.45	160.0	0.81
296.0	1.12	160.0	1.47	170.0	0.83
		170.0	1.51	180.0	0.85
		200.0	1.56	190.0	0.86
		230.0	1.61	200.0	0.87
		260.0	1.65	230.0	0.91
		296.0	1.70	290.0	0.97

Table A-3

Time Step Specifications for TLTA Calculations of Tests 6007 and 6406

End of Interval(sec)	Minimum Timestep Size(sec)	Maximum Timestep Size(sec)
TEST 6007		
1.0	10^{-7}	10^{-3}
10.0	10^{-7}	10^{-2}
20.0	10^{-6}	10^{-2}
100.0	10^{-6}	$5X10^{-2}$
TEST 6406		
1.0	10^{-7}	10^{-3}
10.0	10^{-7}	10^{-2}
20.0	10^{-6}	10^{-2}
100.0	10^{-6}	10^{-2}
400.0	10^{-6}	10^{-2}

Table A-4

BWR/4 ECCS Flow Tables

Core Spray Pressure (psia)	Flow Rate (gal/min)	LPCIS Pressure (psia)	Flow Rate (gal/min)
0.0	2019.0	17.6	580.37
59.0	1892.4	23.7	574.24
137.0	1665.8	26.1	571.85
178.0	1539.2	41.2	556.39
237.0	1262.0	42.4	555.19
268.0	1009.6	98.0	494.15
280.0	757.22	102.0	489.35
287.0	504.81	310.0	0.0
297.0	252.41	3000.0	0.0
304.0	0.0		
10,000.0	0.0		

Table A-5

Time Step Specifications for BWR/4 Blowdown Calculation

End of Interval (sec)	Minimum Timestep Size (sec)	Maximum Timestep Size (sec)
0.005	10^{-6}	10^{-3}
0.010	10^{-5}	10^{-3}
0.100	10^{-5}	10^{-2}
1.0	10^{-5}	10^{-2}
20.0	10^{-5}	5×10^{-2}
30.0	10^{-5}	10^{-1}
120.0	10^{-5}	10^{-1}

Table A-6

High Pressure Core Spray Flow Table for BWR/6

Pressure (psi)	Flow (lbm/sec)
0.0	678.4
215.0	678.4
1162.0	193.3
1195.0	19.724
1195.1	0.0

Table A-7

Low Pressure Coolant Injection Flow Table for BWR6

Pressure (lbs/in^2)	Flow (lbm/sec)
0.0	1467.9
15.0	1467.9
35.0	1387.0
53.0	1284.0
71.0	1192.7
87.0	1100.9
103.0	1009.2
117.0	917.47
130.0	825.7
142.0	733.9
152.0	842.2
163.0	550.5
169.0	458.7
173.0	367.0
177.0	275.2
181.0	183.5
184.0	91.73
187.0	0.0

Table A-8

Time Step Specifications for BWR/6 Blowdown Calculation

End of Interval (sec)	Minimum Timestep Size (sec)	Maximum Timestep Size (sec)
0.005	10^{-6}	10^{-3}
0.01	10^{-5}	10^{-3}
1.00	10^{-5}	10^{-2}
21.0	10^{-5}	5×10^{-2}
76.0	10^{-5}	2×10^{-2}
200.0	10^{-5}	5×10^{-3}

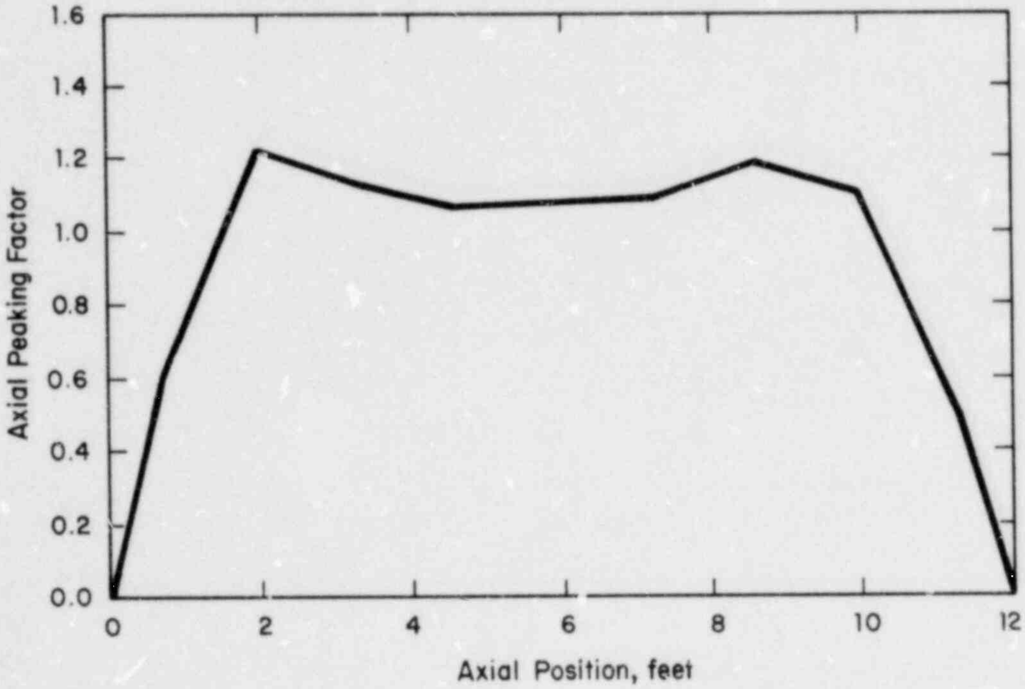

FIGURE A-1. Axial Position (ft) BWR/4 Axial Power Profile