BAW-10193NP-A

Topical Report
January 2000

RELAP5/MOD2-B&W For Safety Analysis of B&W-Designed Pressurized Water Reactors

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October 15, 1999

Mr. J. J. Kelly, Manager
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SUBJECT: SAFETY EVALUATION OF TOPICAL REPORT BAW-10193P, "RELAP5/MOD2-B&W FOR SAFETY ANALYSIS OF B&W-DESIGNED PWRS" (TAC NO. M93346)

Dear Mr. Kelly:

The U.S. Nuclear Regulatory Commission (NRC) has completed its review of the subject topical report that was submitted by Framatome Technologies, formerly B&W Nuclear Technologies, by letter dated August 14, 1995. The intent of the submittal was to use the RELAP5/MOD2-B&W computer code to perform future non-loss-of-coolant accident (non-LOCA) safety analyses on B&W-designed pressurized-water reactors (PWRs), replacing two previously approved non-LOCA codes, CADDS and TRAP2. The topical report presented benchmarks of RELAP5/MOD2-B&W calculations to data from test facilities and plant transients, as well as comparisons to CADDS and TRAP2 predictions, to demonstrate that RELAP5/MOD2-B&W properly predicts the phenomena exhibited by B&W-designed PWRs during non-LOCA events.

The staff finds that the subject topical report is acceptable for referencing in licensing applications to the extent specified and under the limitations delineated in the report and in the associated NRC safety evaluation. The safety evaluation, which is enclosed, defines the basis for acceptance of the topical report. This closes the staff's efforts on TAC No. M93346.

The staff will not repeat its review of the matters described in the subject report, when the report appears as a reference in license applications, except to ensure that the material presented applies to the specific plant involved. In accordance with the procedures established in NUREG-0390, the NRC requests that Framatome Technologies publish accepted versions of the submittal, proprietary and non-proprietary, within 3 months of receipt of this letter. The accepted versions shall incorporate (1) this letter and the enclosed safety evaluation between the title page and the abstract, and (2) an "A" (designating "accepted") following the report identification symbol.

If the NRC's criteria or regulations change so that its conclusions about the acceptability of the report are invalidated, Framatome Technologies or the applicant referencing the report, or both, will be expected to revise and resubmit its respective documentation, or submit justification for the continued effective applicability of the report without revision of the respective documentation.

Sincerely,

Stewart N. Bailey, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 693

Enclosure: Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FRAMATOME TECHNOLOGIES GROUP

TOPICAL REPORT BAW-10193P

"RELAP5/MOD2-B&W FOR SAFETY ANALYSIS OF

B&W-DESIGNED PRESSURIZED-WATER REACTORS"

1.0 INTRODUCTION

By letter (Reference 1) dated August 14, 1995, Framatome Technologies Group (FTG), formerly B&W Nuclear Technologies (BWNT), submitted Topical Report BAW-10193P, "RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurized Water Reactors," for staff review. FTG intended to use the RELAP5/MOD2-B&W code (Reference 2) to perform future safety analyses of non-LOCA (loss-of-coolant accident) transients and accidents for the Babcock & Wilcox (B&W)-designed pressurized-water reactors (PWRs), replacing the currently used CADDS (Reference 3) and TRAP2 (Reference 4) codes. Currently, CADDS is used to analyze system responses to primary system transients such as reactivity transients, loss of primary flow events, and anticipated transients without scram; and TRAP2 is used to calculate system responses to the secondary system initiated events, such as steam line break, turbine trip, feedwater line break, and steam generator tube rupture. Consistent with how theTRAP2 and CADDS codes are currently used for safety analyses, RELAP5/MOD2-B&W will be used for predicting the reactor coolant system (RCS) and core power responses to non-LOCA events, while the LYNXT code (Reference 5) will continue to be used for calculating minimum departure from nucleate boiling ratio (DNBR) for the core hot channel.

1.1 Background – RELAP5/MOD2-B&W

RELAP5/MOD2-B&W is an FTG version of the advanced system analysis code RELAP5/MOD2. RELAP5/MOD2 was developed by the Idaho National Engineering Laboratory as a best-estimate code for analyzing a wide variety of light-water reactor (LWR) system transients. The fundamental equations, constitutive models and correlations, and method of solution of RELAP5/MOD2 are described in NUREG/CR-4312 (Reference 6). The code is designed to model the behavior of all major components in the reactor system during transients and accidents ranging from large-break and small-break LOCAs to anticipated operational transients involving plant control and protection systems. This code is organized into modules by components and functions to simulate the primary coolant system, secondary system, feedwater train, system controls, and core neutronics. Special component models include pumps, valves, heat structures, electric heaters, turbines, separators, and accumulators. RELAP5/MOD2-B&W retains virtually all of the features of the original RELAP5/MOD2 code, while making certain modifications either to add predictive capabilities of the constitutive models or to improve code execution. More significant modifications are the addition of models and features to meet the 10 CFR Part 50, Appendix K requirements for emergency core cooling system (ECCS) evaluation model (EM) calculations. The details of RELAP5/MOD2-B&W are described in Topical Report BAW-10164P-A, Revision 3.

RELAP5/MOD2-B&W uses a one-dimensional (axial), transient, two-fluid hydrodynamic model to calculate the flow of a steam-water two-phase mixture. This two-fluid model uses six field equations (i.e., two phasic-continuity, momentum, and energy equations each), which provide the capability to calculate the characteristics of non-homogeneous, non-equilibrium flow. The hydrodynamics model also contains options for invoking simpler models, such as homogeneous flow, thermal equilibrium, and frictionless flow models, which can be used independently or in combination. The system model is solved numerically using a semi-implicit finite difference technique. The user can also select an option for solving the system model using a nearly implicit finite difference technique that allows for violation of the material Courant limit, and is suitable for steady-state, and slowly varying, quasi-steady-state transient calculations.

The code uses a point-kinetics model with six delayed neutron groups to calculate reactor power as a function of time. It contains provisions for fuel temperature, moderator temperature, and density reactivity feedback. Other reactivity feedbacks such as those caused by boron concentration changes and tripped-rod reactivity are provided with input tables for generalized reactivity with respect to time.

The constitutive models include models for defining flow regimes, and flow-regime-related models for calculating wall friction, interphase mass transfer, heat transfer, and drag force. Also included are a core structure heat transfer model and a fuel pin heat conduction model with a dynamic fuel cladding gap conductance model. The core heat transfer package can calculate heat transfer coefficients for various heat transfer regimes from single-phase convection, nucleate boiling, to post-critical heat flux (CHF) heat transfers.

Other special features include dynamic pressure loss models associated with abrupt area change for single-phase and two-phase flows, a centrifugal pump performance model with two-phase degradation effects, choked flow models with treatment for horizontal stratification, non-homogeneous two-phase flow, countercurrent flow models, crossflow junction, decay heat models, a fine mesh renodalization scheme for heat conduction, liquid entrainment, a motor valve model, a relief valve model, control system, and trip system.

The B&W-designed nuclear steam supply system (NSSS) is unique in that it employs once-through steam generators (OTSGs), in contrast to the U-tube recirculating steam generators (RSGs) in the PWRs designed by other vendors. The OTSG is a counterflow, single pass, tube and shell heat exchanger that produces superheated steam at a constant secondary pressure over the entire load range. The boiling heat transfer area and secondary inventory vary with load, requiring special modeling capabilities to properly predict plant response. For application of RELAP5/MOD2-B&W to the B&W-designed plants, Revision 2 of RELAP5/MOD2-B&W added the BWUMV CHF correlation, the Wilson model for determining interphase drag, and a countercurrent flow limit model for performing small-break LOCA analyses. Revision 3 of RELAP5/MOD2-B&W included enhancement to the EM fuel pin model, EM heat transfer model, and models to support use of the code for analyses of OTSG plants. These models included the Becker CHF correlation, the BWNT slug-drag model, the high auxiliary feedwater (AFW) model, and the Chen nucleate boiling heat transfer coefficient void ramp.

The NRC has approved RELAP5/MOD2-B&W for LWR LOCA and non-LOCA transient analyses (Reference 2). The staff has also approved specific application of RELAP5/MOD2-B&W for performing LOCA and non-LOCA analyses on PWRs with RSGs (References 7 & 8) and for performing LOCA analyses on B&W-designed PWRs (Reference 9). The purpose of BAW-10193P is to obtain NRC approval to extend the application of RELAP5/MOD2-B&W for safety analyses of non-LOCA transients of the B&W-designed PWRs.

2.0 EVALUATION

To support the use of RELAP5/MOD2-B&W for safety analyses of non-LOCA transients of B&W-designed PWRs, BAW-10193P presents the following benchmarks of the RELAP5/MOD2-B&W calculations against various data and code predictions:

- benchmarks of OTSG test facility data to demonstrate adequacy of RELAP5/MOD2-B&W modeling of OTSG in predicting boiling length in the steam generator (SG) secondary side, and primary-to-secondary heat transfer during upset conditions,
- benchmarks of B&W-designed PWR plant data to demonstrate the ability of RELAP5/MOD2-B&W to predict the phenomena exhibited during non-LOCA events, and
- comparisons to the CADDS and TRAP2 calculations of non-LOCA events to demonstrate the similarity in the predictions of the system and core power responses between RELAP5/MOD2-B&W and these approved codes.

The sections that follow describe the staff evaluations of these benchmarks.

2.1 Benchmarks Against OTSG Test Data

Section 4 of BAW-10193P discusses the benchmarks of RELAP5/MOD2-B&W against the 19-tube model OTSG tests performed at the Alliance Research Center Nuclear Steam Generator Test Facility. The 19-tube full-length model OTSG is a single-pass, counterflow, tube and shell heat exchanger. It can be used to simulate OTSGs with either an aspirator or integral economizer, referred to herein as aspirator-OTSG (AOTSG) and integral economizer-OTSG (IEOTSG), respectively. Primary inlet flow entered at the top of the SG, flowed downward through the tube bundle, and exited at the bottom. Secondary feedwater flow entered by way of an external downcomer, through the bottom of the tube bundle, boiled as it passed by the outside of the tube bundle, and exited at the top as superheated steam. When run in the AOTSG mode, steam bled from the tube region, which simulated the aspirator and raised the feedwater temperature to saturation conditions by mixing the water with steam. In the IEOTSG mode, the steam bleed was closed and the subcooled feedwater entered at the bottom of the tube nest.

The following four sets of tests were simulated with RELAP5/MOD2-B&W:

- two sets of steady-state tests performed in 1969 and 1971 to determine the thermal performance of the AOTSG and IEOTSG, respectively, and
- two sets of loss of feedwater flow (LOFW) tests from scaled full power conditions, performed in 1977, for the AOTSG and IEOTSG designs, respectively.

The steady-state OTSG tests were performed for a range from 20 to 100 percent of the scaled full power with primary pressure and inlet conditions, feedwater conditions, and secondary pressure held constant for each test. The boiling lengths (dryout locations) as a function of scaled power level were determined from primary tube and secondary-side thermocouples.

The AOTSG LOFW test was initiated from the scaled full-power conditions by simultaneously tripping the feedwater pump and closing the feedwater isolation valve. The SG was allowed to boil dry and then the feedwater was restarted. Secondary steam flow and temperature and primary outlet temperature were measured during the tests. The IEOTSG LOFW test had the same procedure, except for the closure of the downcomer isolation valve rather than the feedwater isolation valve. However, the test data suggested that the feedwater isolation valve was actually closed during the IEOTSG LOFW test instead, thus allowing feedwater to trickle into the tube region of the downcomer. Therefore, as described in FTG's response to staff Question 2A (Ref. 10), the boundary conditions used in the RELAP5/MOD2-B&W benchmark for the IEOTSG LOFW test contained an estimate of the average rate of liquid displacement from the downcomer.

2.1.1 <u>RELAP5/MOD2-B&W Model Description</u>

The RELAP5/MOD2-B&W model of the 19-tube OTSG test facility utilized 11 axial volumes in the primary tube region and in the secondary shell region. Primary-to-secondary heat transfer was modeled using 11 heat structures between the primary and secondary sides. The external downcomer was modeled with five axial control volumes, representing the piping from the steam/feedwater mixing region to the tube bundle inlet. A feedwater aspiration was provided by a single junction component that connected the tube bundle region to the external downcomer. A junction connection between the shell side of the heat exchanger and the control volume representing the steam/water mixer was included. Time-dependent volume and junction components were used to set the primary- and secondary-side coolant inlet flowrates and temperatures.

The following benchmarks were performed with certain features available in RELAP5/MOD2-B&W for the OTSG shell side:

- use of a specific CHF correlation on the shell side of the tube heat structure to provide a better prediction of the dryout point in the OTSG,
- use of the BWNT slug flow drag model with default multipliers (in RELAP5/MOD2-B&W) to reduce the interphase drag in the slug and annular-mist flow regimes, and

- use of a multiplicative weighting factor to force the boiling suppression factor of the Chen nucleate boiling correlation to zero as the steam void fraction approaching 1.0.

These features, incorporated in Revision 3 of BAW-10164P-A, have been approved by NRC (Reference 2).

2.1.2 Comparisons with Test Results

Tables 4-2 and 4-4, and Figures 4-4 and 4-8 of BAW-10193P compare the boiling lengths above the lower tube sheet at various power levels predicted by RELAP5/MOD2-B&W to those measured during the AOTSG and IEOTSG steady-state tests. They show good agreement between the RELAP5/MOD2-B&W predictions and the test data. Table 4-2 also shows the AOTSG steady-state boiling lengths calculated with the base RELAP5/MOD2 code, Cycle 36.05.

These boiling lengths calculated with the RELAP5/MOD2 base code, however, differed significantly from the test data below 80 percent scaled power. FTG attributed the improvement of the RELAP5-MOD2-B&W calculations over the RELAP5/MOD2 base code to the use of the specific CHF correlation. This CHF correlation was developed from heated rod bundle dryout data and will be used in future safety analyses.

Table 4-3 and Figures 4-6 and 4-7 of BAW-10193P present the benchmark results of the AOTSG LOFW test. The RELAP5/MOD2-B&W calculations of the initial conditions agree well with the measured values of the primary and secondary system fluid temperatures preceding the initiation of the LOFW test. For the LOFW test progressing, the RELAP5/MOD2-B&W calculations of steam flow agreed well with the data except for some sharp step changes in the calculated steam flow. These step changes occur as the secondary-side liquid-steam mixture level crosses the control volume boundaries, resulting in sudden changes in heat transfer as the control volumes in the tube region systematically dry out and, later, refill. FTG stated that the addition of control volumes in the nucleate boiling region would decrease the magnitude of the step changes, but the number of steps would increase. The resulting predictions of heat transfer and primary outlet temperature would be approximately the same as the current prediction.

Figure 4-7 shows differences between the predicted and observed primary-side outlet temperatures. FTG attributed these differences to the heat capacity of the resistance thermal detector (RTD) used in the test. The RTD heat capacity caused a lag in the measured temperature response such that the actual fluid temperatures were higher than the recorded values during heatup and lower than the recorded values during the refill. In response to staff question 2 (Ref. 10), FTG performed a Laplace transform of the calculated primary-side outlet temperature to account for the RTD time constant. The adjusted temperature results show good agreement with the actual RTD output, demonstrating that, if a thermal lag were applied to the code prediction to account for RTD capacity, a good match would be obtained with the measured data. This indicates that the RELAP5/MOD2-B&W prediction of transient heat transfer is in good agreement with the test, as demonstrated by the good agreement between the predicted and measured steam flow. The benchmarks demonstrate that RELAP5/MOD2-B&W with the control volume arrangement used in this benchmark can predict the shell-side boiling length at various power levels, as well as primary-to-secondary heat transfer of the AOTSG design.

Table 4-5 and Figures 4-10 and 4-11 of BAW-10193P show the benchmark results of the IEOTSG LOFW transient. The RELAP5/MOD2-B&W predictions of the steam temperature and the primary outlet temperature agree well with the test data for the first 20 seconds of the transient. After 20 seconds, a deviation occurs when the predicted primary outlet temperature rapidly approaches the inlet temperature as the IEOTSG dries out, but the observed temperature remains much lower than the inlet temperature, indicating continued heat transfer. FTG stated that the continued heat transfer is not supported by the measured steam flow, and that the deviation between the predicted and observed temperatures is primarily due to the RTD heat capacity.

Figure 4-10 shows the code prediction of dryout time to be 2 seconds less than the observed time. This is because the code overpredicted the steam flow from the IEOTSG during the dryout period. This results in early dryout and a low prediction of primary outlet temperature caused by an overprediction in heat transfer. The overprediction of steam flow arose as the mixture level crossed the control volume boundaries. FTG concluded that the noding detail used to predict the IEOTSG test data is too crude to produce an accurate result and that additional modeling is required. FTG, in its response to staff question 2 (Ref. 10), indicated that the benchmark will not be refined at this time, because it has no plans to perform IEOTSG plant safety analysis. FTG stated that before (or concurrent with) the licensing submittal of an IEOTSG plant safety analysis, it will submit an updated benchmark of these test data for NRC review. The staff concludes that RELAP5/MOD2-B&W may not be used for the safety analyses of PWRs with IEOTSGs until FTG submits and the staff accepts an updated benchmark of the IEOTSG LOFW test case.

2.2 Benchmarks to Plant Data

Section 5 of BAW-10193P describes benchmarks of RELAP5/MOD2-B&W against the data of the following four transient events or tests of B&W-designed PWR plants:

- Three-Mile-Island Unit 2 (TMI-2) LOFW event of March 26, 1979
- Rancho Seco loss-of-ICS (integrated control system) power event of December 26, 1985
- Four-pump coastdown data from Oconee Unit 1 and Crystal River Unit 3
- Three-Mile-Island Unit 1 (TMI-1) natural circulation test of October 7, 1985

The benchmarks were performed with a generic B&W lowered-loop 177 fuel assembly plant model depicted in Figure 5-1 of BAW-10193P. The special RELAP5/MOD2-B&W features described in Section 2.1.1 of this safety evaluation (SE) were employed for the OTSG model. The B&W high AFW model with AFW injection from high elevation location was used in conjunction with a two-region SG model. This allowed for the heat transfer in the tube region wetted by AFW to be calculated separately from the heat transfer in the tubes that are unwetted by AFW.

In each benchmark, the initial conditions were set to the plant conditions that existed preceding the event or test, and the transient was simulated by imposing the plant boundary conditions, taken from the data recorded by the plant recall computer, or estimated using available data. When required, the core decay heat input was calculated from the plant power history using 1979 ANS (American Nuclear Society) 5.1 methodology. The predicted values of primary pressure, secondary pressure, primary system fluid temperatures and pressurizer level were compared with the plant values.

2.2.1 Benchmark of TMI-2 LOFW Event

The TMI-2 LOFW event occurred as a result of the loss of both main feedwater (MFW) pumps while the plant was operating at 97 percent power. This event caused a coincident turbine trip, resulting in the secondary pressure increase and primary-to-secondary heat transfer reduction. The mismatch between the core heat generation and SG heat removal caused the RCS pressure to increase, the power-operated relief valve (PORV) to open, and the reactor to trip on high RCS pressure. During the post-trip RCS cooldown and contraction as the core power dropped to the decay heat level, however, the PORV failed to close when the RCS pressure fell below the low-pressure setpoint, and the RCS continued to depressurize. Approximately 40 seconds into the event, the SG water level dropped to the low-level setpoint and the AFW control valves opened automatically to supply AFW to maintain minimum SG levels. However, the AFW block valves between the control valves and the SGs were closed, preventing the AFW from being delivered. Consequently, the SGs dried out, and the RCS began to reheat. Eventually, at about 8 minutes into the event, the AFW was restored to the SGs. This benchmark was performed for the first 2 minutes of the event to focus on predicting the plant behavior during the LOFW period.

Tables 5-1 through 5-6 and Figures 5-2 through 5-10 of BAW-10193P showed the benchmark results of the TMI-2 LOFW event. The code properly predicted the primary- and secondary-system pressurization rates following the turbine trip, and also predicts the timing of PORV lift and reactor trip. The calculations of the post-trip RCS pressure and temperatures, and SG liquid levels and dryout time agreed with the plant data. The only significant deviation from the plant data occurred in the calculated pressurizer liquid level, which agreed with the plant data until the reactor trip. After the reactor trip, the plant data appeared to indicate a much greater outsurge than was predicted by the code. At 50 seconds into the event, the pressurizer liquid level was calculated to be 191 inches compared to the recorded plant data of 159 inches. However, FTG stated that the plant data were probably not reliable, and the level should be about 189 inches. This was based on experience with operating B&W plants that the pressurizer level should decrease by 5.5 inches for every 1°F decrease in average system temperature. Therefore, the predicted pressurizer level was very close to the actual level.

The benchmark of the TMI-2 LOFW event shows that RELAP5/MOD2-B&W is appropriate for analyzing overheating events on B&W-designed PWRs.

2.2.2 Benchmark of the Ranch Seco Loss of ICS Power Event

In 1985, while operating at 76 percent power, the Rancho Seco plant experienced a loss of dc power for the integrated control system (ICS). The loss of ICS power caused a reduction of the MFW flow, and an increase of total steam flow from the opening of turbine bypass valves (TBVs) and atmospheric dump valves (ADVs), resulting in an RCS overcooling.

Tables 5-8 through 5-10 and Figure 5-11 of BAW-10193P showed the transient boundary conditions, including the core power, MFW flow, AFW flow, main steam safety valve (MSSV) flow, and the primary makeup and high-pressure injection (HPI) flows. Table 5-9 shows that, after an earlier termination, the AFW flow to SG-A was restored at 976 seconds into the event, following damage to the isolation valve for AFW flow to SG-A. In addition, the recorded AFW flow to SG-A went off-scale for a portion of the transient; thus the AFW flow during this time was estimated from system conditions and the AFW pump head/capacity curve.

The benchmark results are shown in Figures 5-12 through 5-18 and Tables 5-11 and 5-12 of BAW-10193P. The loss of ICS power caused a reduction of the MFW flow, and an increase of total steam flow resulting from opening of the TBVs and ADVs. Initially, the RCS temperatures and pressure increased because the increased heat removal by the increased steam flow cannot overcome the heat removal reduction from the loss of MFW flow. At approximately 15 seconds into the event, the reactor tripped on high RCS pressure. The code predictions of this scenario were consistent with the plant computer data.

When the reactor tripped, the turbine also tripped, causing the secondary pressure to increase to the MSSV lift setpoint. The secondary pressure decreased subsequently as steam was relieved through the MSSVs. This caused the RCS to undergo a post-trip cooldown and contraction as the reactor power fell to decay heat, emptying the pressurizer. As the SGs continued to depressurize from the open ADVs, full AFW flow was started, and the RCS continued to cool and depressurize to the threshold for engineered safety features actuation system (ESFAS) actuation at abcut 200 seconds. The actuation of the ESFAS initiated the HPI flow, and slowed the RCS contraction rate caused by the continued feeding and depressurization of the SGs. RCS depressurization continued until the fluid flashed in the reactor vessel upper head occurred at approximately 400 seconds. At approximately 500 seconds, the RCS pressure stabilized as the upper head liquid flashing and HPI addition compensated for the contraction of the primary system. The code prediction of the RCS temperatures and pressure, pressurizer level, and secondary pressure and SG levels agreed well with the recorded values during this period.

In the next period, as the HPI volumetric flow exceeded the RCS contraction rate, the RCS started to repressurize, ending the flashing in the upper head, and the pressurizer started to refill. By 700 seconds, the secondary relief valves were closed, and the AFW flow to SG-A was terminated. This ended the RCS overcooling, and caused the pressurizer level and the RCS pressure to increase at a greater rate. At 976 seconds, however, the SG-A AFW isolation valve was damaged, resulting in a restoration of the AFW flow to SG-A. Consequently, the RCS cooldown resumed, and the RCS contraction rate increased, so that the RCS pressure and the pressurizer level stabilized. At 1150 seconds, pressurizer spray flow was actuated to decrease RCS pressure. At about 1550 seconds, ICS power was restored, and all ICS demand signals were reduced to zero percent, ending the AFW flow and overcooling of the RCS.

The code underpredicted the RCS repressurization, which FTG attributed to the uncertainty in measured HPI flows that were input in the benchmark calculation. In response to staff question 4 (Ref. 10), FTG reanalyzed RCS pressure response in the loss of ICS power event using an increased HPI flow. The results demonstrate that an increase in HPI flow has little effect on the pressure prediction during the depressurization because of the dominant effect of the RCS contraction caused by overcooling. However, once the cooling of the RCS was reduced, the effect of HPI flow on the pressure and pressurizer level increased significantly. The result provided also agreed more closely with the data.

The code prediction of secondary pressure during the entire transient is excellent. The code predicted secondary liquid levels very well, given the uncertainties in the AFW flow estimate. The predicted RCS temperatures, RCS pressure, and pressurizer level matched up well with the recorded values. This benchmark demonstrates RELAP5/MOD2-B&W's capability for analyzing secondary-system-initiated events.

2.2.3 Benchmark of Flow Coastdown Data

Four-pump coastdown tests were performed from hot, full-pressure, zero-power conditions during the startup tests at Oconee Unit 1 and Crystal River Unit 3, both of which are of the B&W lowered-loop 177-FA design. RELAP5/MOD2-B&W was benchmarked against the flow and pump speed data recorded from these tests. The results of these comparisons, shown in Figure 5-19 of BAW-10193P, shows that the predicted pump response essentially overlays the plant data. This demonstrates that the pump inertia, pump frictional torque values, and reactor coolant loop flow resistance input to the RELAP5/MOD2-B&W plant model yield an accurate calculation of the system flow rate.

2.2.4 Benchmark of TMI-1 Natural Circulation Test

A low-power natural-circulation (NC) test was conducted on October 7, 1985, at the TMI-1 plant to demonstrate the NC heat removal capability of B&W-designed PWRs. The NC test was initiated by tripping the reactor coolant pumps while the unit was operating at approximately 3 percent power with full RCS flow and SG liquid levels controlled to the NC setpoints. Throughout the NC test, SG pressures were controlled to within 11 psi of the initial value, and the SG levels were maintained at about 50 percent on the operating range by a control system using AFW. The RCS pressure was regulated by the intermittent use of the pressurizer spray and adjusting letdown flow during pump coastdown, and by the use of the pressurizer heaters and letdown flow after NC was established. These test conditions were used as boundary conditions for the RELAP5/MOD2-B&W benchmarks.

The B&W 177-FA lowered-loop plant model was used for the benchmark analysis. There are a few differences in the boundary condition inputs to the code relative to the test data. The reactor power input to the code, shown in Table 5-13 of BAW-10193P, was equal to the measured power multiplied by a correction factor of 1.12. The correction factor of 1.12 (see FTG's response to staff question 5, Reference 10, for the derivation) was necessary because the out-of-core neutron detectors used for power measurement were calibrated at a temperature higher than the reactor vessel downcomer fluid temperature during the test. The inputs of the setpoints of pressurizer heaters 3 and 4, shown in Table 5-14 of BAW-10193P, were slightly lower than the plant data. The SG level boundary condition in the benchmark, shown in Figure 5-20 of BAW-10193P, was maintained near the 50 percent level by a simplified control system as opposed to the larger variation observed in the test data. However, the effects of these differences on the predictions were small (see FTG's response to staff question 5, Reference 10).

The benchmark results are shown in Table 5-16 and Figures 5-21 through 5-23 of BAW-10193P. The hot leg temperature, RCS pressure, and pressurizer liquid level rose during the RCS pump coastdown, and decreased subsequently with the decline in core power until they stabilized. The code predictions of these parameters agreed well with the test data, although they were overpredicted during the pump coastdown period. The equilibrium RCS fluid temperature difference was calculated to be 34°F compared to 35°F in the test. Therefore, the calculated NC flow was within 3 percent of the test result. BAW-10193P attributed the slightly higher NC flow predicted by the code to the plant model used in the analysis. This plant model had only three axial control volumes for the core, so that the core thermal center in the model was 2 feet below the mid-core elevation. Consequently, the SG-to-core thermal center difference in the model was greater than in the plant, yielding an RCS NC flow slightly greater than the test. However, the code properly predicted the SG thermal center during NC, indicating accurate calculations of the heat transfer in the tube region wetted by AFW, and the heat transfer to the secondary pool. This benchmark demonstrates that RELAP5/MOD2-B&W is suitable for analyzing the response of B&W-designed PWRs for NC events.

2.3 Benchmark of RELAP5/MOD2-B&W Against CADDS and TRAP2

Section 6 of BAW-10193P describes comparisons of the RELAP5/MOD2-B&W predictions of the control rod withdrawal and the main steam line break (MSLB) transients against the predictions of the NRC-approved codes, CADDS and TRAP2, respectively. CADDS has been used for the analyses of the primary system response of B&W-designed PWRs to such transients as reactivity insertion and loss-of-primary-flow events. TRAP2 has been used for the calculations of the core power and system responses to the secondary-system-initiated events, such as MSLB, turbine trip, loss of feedwater, and steam generator tube rupture accidents. The intent of the RELAP5/MOD2-B&W comparisons with the CADDS and TRAP2 predictions is to show that, with the same conservative initial and boundary conditions used in the safety analyses, RELAP5/MOD2-B&W will predict system and core power responses of various non-LOCA events similar to those predicted by CADDS and TRAP2.

2.3.1 RELAP5/MOD2-B&W – CADDS Comparison of Startup Events

The following three startup events of control rod withdrawal from a low power condition were analyzed with both the RELAP5/MOD2-B&W and CADDS codes for comparisons:

- the withdrawal of a single control bank from hot zero power condition,
- the withdrawal of all control rods from hot zero power condition, and
- an intermediate rod withdrawal with a spectrum of reactivity insertion rates.

These control rod withdrawal events resulted in a reactivity insertion into the core and an RCS overpressurization. Different reactivity insertion rates were used to simulate these cases.

The CADDS analyses of the rod withdrawal events used lumped single-loop modeling of the RCS consisting of the hot leg, SG, cold leg, reactor core and bypass, and pressurizer. The RELAP5/MOD2-B&W analyses used a simplified generic B&W 177-FA plant model shown in Figure 6-1 of BAW-10193P, which is a more detailed representation of the plant than the CADDS model. The model consisted of two hot legs and SGs, four cold legs with reactor coolant pumps, the reactor core, and the pressurizer. The RCS pumps were explicitly modeled as compared to the CADDS model that used a specific flow rate and loop time delay input. However, the following features of the RELAP5/MOD2-B&W model were purposely altered to match the CADDS model formulation:

- deletion of the upper reactor vessel head region,
- exclusion of all heat structures but the fuel pin, SG tubing, and pressurizer shell metal,
- use of a single control volume and heat structure modeled for the core region, and
- use of a constant heat demand to model the SG heat transfer.

Table 6-1 of BAW-10193P compares the RCS initial and boundary conditions between the CADDS and RELAP5/MOD2-B&W calculations. The reactor core kinetics parameters and the reactor trip setpoints are shown in Table 6-2.

The comparisons between the RELAP5/MOD2-B&W and CADDS predictions of the "single control bank withdrawal" and "all-rods withdrawal" cases are shown in Tables 6-3 and 6-4, and Figures 6-2 through 6-15 of BAW-10193P. In both cases, the reactor power increased as a result of reactivity insertion from the control rod withdrawal. The heat addition to the core caused the increases of the RCS pressure and temperature. In the single-bank-withdrawal case, Doppler reactivity feedback terminated the power excursion before a high flux trip was reached, and the reactor subsequently tripped on high RCS pressure. In the all-rods-withdrawal case, the reactor tripped on the high neutron flux. After the reactor trip, control rod insertion sharply reduced the reactor power and the pressurization rate. The overpressure ended after the lift of the pressurizer safety valves.

In both cases, the RELAP5/MOD2-B&W's predictions of the neutron power, thermal power, and fuel temperature responses agreed well with the CADDS predictions, except for the time delays of the sharp declines of the power and fuel temperature for the single-bank-withdrawal case. These delays are due to the slower system pressurization predicted by RELAP5/MOD2-B&W that resulted in the reactor trip on high RCS pressure to be later than that predicted by CADDS. The CADDS prediction of pressurizer pressure response closely approximated an adiabatic compression of the pressurizer steam region, whereas the RELAP5/MOD2-B&W prediction considered the real effects of condensation at the steam-liquid interface and on the surface of the pressurizer shell metal. The pressurizer model difference resulted in later reactor trip with attendant greater peak thermal power, but lower peak RCS pressure than predicted by CADDS.

After the reactor trip, the rates of RCS fluid expansion, pressurizer insurge, and pressurization reduced as the core power decreased. Although the CADDS prediction showed that the RCS pressurization rate remained unchanged after reactor trip until the lift of pressurizer safety valves, the RELAP5/MOD2-B&W predictions of these sequences of events are more consistent with the reduction of thermal power after reactor trip.

Intermediate rod withdrawal with a spectrum of reactivity insertion rates was also analyzed with both RELAP5/MOD2-B&W and CADDS codes. Figures 6-16 and 6-17 of BAW-10193P show the comparisons of peak neutron and thermal powers as a function of reactivity insertion rate predicted by the two codes. These figures showed good agreement between the two codes.

The overall comparisons demonstrate RELAP5/MOD2-B&W to be suitable for analyzing the system response during reactivity transients on B&W-designed PWRs.

2.3.2 RELAP5/MOD2-B&W – TRAP2 Comparison of Main Steam Line Break

The RELAP5/MOD2-B&W and the TRAP2 calculations were compared for two MSLB accidents: a 6.28-square-foot double-ended rupture and a 2.0-square-foot split break of a steam line in the steam generator B (SG-B). The RELAP5/MOD2-B&W MSLB analyses used the generic large-detail B&W lowered-loop 177-FA plant model, shown in Figure 6-18 of BAW-10193P. This model was altered to be consistent with the TRAP2 model by:

- deletion of the upper reactor vessel head region,
- exclusion of all heat structures except for the fuel pin and SG tubing,
- addition of secondary steam and feedwater piping, and a feedwater pump simulation,
- use of the same break geometry and critical flow model, AFW flow table for the unaffected SG, and HPI flow versus pressure table.

The reactor core neutronics parameters and reactor protection trip setpoints are shown in Table 6-6, and the ESFAS setpoints and delay times are shown in Table 6-7 of BAW-10193P. Both the double-ended rupture and split-break cases were initiated from the same plant conditions. Table 6-5 of BAW-10193P shows the initial conditions established by the RELAP5/MOD2-B&W and TRAP2 codes. The comparisons between the RELAP5/MOD2-B&W and TRAP2 transient analysis results are shown in Table 6-8 and Figures 6-19 through 6-28 of BAW-10193P for the double-ended rupture case, and in Table 6-9 and Figures 6-29 through 6-38 for the split-break case.

The plant system responses to the double-ended rupture and split break cases were very similar. Each MSLB caused decreases in the SG pressure and saturation temperature, an increase in primary-to-secondary heat transfer, and attendant decreases in the RCS pressure and temperature. Because of negative moderator temperature coefficient (MTC), the RCS cooling caused the core fission power to increase. In the double-ended rupture case, the RCS depressurization proceeded at a faster rate, resulting in the reactor trip on low RCS pressure. In the split-break case, the power surge resulted in a reactor trip on high neutron flux at a later time than the double-ended break. After the reactor trip, the RCS pressure continued to decrease because of overcooling, resulting in the actuation of the ESFAS on low RCS pressure. The ESFAS actuation initiated closures of the main steam isolation valve (MSIV) and main feedwater isolation valve (MFIV), and actuations of the AFW flow to the unaffected SG and the HPI flow with respective time delays. Following the MSIV closure, the unaffected SG repressurized and became a heat source, while the affected SG continued to depressurize at a faster rate as it became the sole source of the break flow. After the MFIV closure, the unaffected SG continued to fill with the AFW flow. Meanwhile, the affected SG started to dry out, which, however, was delayed as liquid in the feedwater pipe began to flash, pushing liquid into the SG.

The RELAP5/MOD2-B&W predictions of the break flow, RCS pressure and temperature, secondary pressure, core reactivity, and power agreed well with those of TRAP2. There are differences between the calculations of the two codes in the SG secondary mass inventory, cold leg temperature, total reactivity, and neutron power after the MFIV closure. BAW-10193P attributed these prediction differences to the differences between the two codes in the calculation of steam-liquid phase slip in the secondary system. The TRAP2 bubble rise velocity inputs on the secondary side were typically a constant 0.5 foot per second, which means that the control volume fluid conditions were effectively assumed homogeneous. On the other hand, RELAP5/MOD2-B&W used an NRC-approved mechanistic model (B&W slug flow drag model) to calculate the steam-liquid phase slip in the SG and feedwater piping control volumes, thus providing more realistic calculations of the SG boiling length and dryout. Because RELAP5/MOD2-B&W calculated higher phase slip in the SG, the primary-secondary heat transfer was lower as the break flow continued through the break, resulting in higher cold leg temperature prediction and earlier minimum cold leg temperature than the TRAP2 predictions. Higher phase slip in the feedwater piping control volume predicted by RELAP5/MOD2-B&W also resulted in less liquid transported into the SG as the piping liquid flashed. Therefore, it calculated earlier SG dryout and a shorter boiling length than did TRAP2, resulting in less heat transfer. Consequently, RELAP5/MOD2-B&W predicted higher RCS temperature, and less severe core reactivity and power just preceding SG dryout than the TRAP2 predictions.

The RELAP5/MOD2-B&W and TRAP2 comparisons of the MSLB events demonstrate that, given conservative initial and boundary conditions, RELAP5/MOD2-B&W produces conservative results, similar to those predicted by TRAP2.

2.4 RELAP5/MOD2-B&W Non-LOCA Safety Analysis Methodology

In response to staff question 1 (Ref. 10), FTG added an appendix to BAW-10193P, "Non-LOCA Analysis Methodology for B&W-Designed Plants." The appendix gives guidance on using RELAP5/MOD2-B&W for safety analyses of various transients and accidents. The guidance covers (1) the NSSS model noding details, (2) the options for the constitutive models and correlations, and (3) determining input assumptions, initial conditions, and boundary conditions. In general, FTG intends to conform to the accident analysis methods and licensing bases described in the Updated Final Safety Analysis Reports (UFSARs) of the B&W-designed plants, except for the use of RELAP5/MOD2-B&W in place of CADDS and TRAP2, and other exceptions noted in the appendix.

The staff reviewed the appendix and finds that the guidance is consistent with those methodologies chosen for the benchmark analyses. Figures A.1 and A.2 in the appendix show the noding details of (1) the large detail plant model for the analyses of those transients dominated by the performance of the SG or secondary plant systems, or both, and (2) the reduced detail plant model for those transients dominated by the core response, respectively. Table A.1 in the appendix specifies which plant models should be used for various transients. These noding details are consistent with those used in the benchmarks. The user input options for the constitutive models and correlations for the safety analyses described in Section A.2 of the appendix are consistent with those used in the benchmark analyses.

Section A.3 of the appendix presents guidance on safety analysis input assumptions regarding the initial conditions, boundary conditions, reactivity coefficients, effects of control system, loss of offsite power, and single failure assumptions. These assumptions are consistent with those described in the UFSARs, except for (1) the initial condition of pressurizer level and (2) the core decay heat calculation. Although the guidance still follows the original UFSAR safety analyses in setting the initial pressurizer level to the nominal value for most transients, it also advises setting the initial pressurizer level to a value greater than or equal to nominal value plus measurement uncertainty for those events that cause an increase in pressurizer liquid level or RCS pressure. This assumption is more conservative than the nominal value used in UFSARs, and is therefore acceptable.

Regarding the core decay heat, the guidance advises the use of the ANS 1971 decay heat standard plus actinide decay for analyzing non-LOCA transients. This is different from the original UFSAR analyses of using 1.2 times the ANS 1971 decay heat standard, or later, the use of the ANS 1979 decay heat standard. The ANS 1979 standard more accurately predicts core decay heat following a reactor trip, and presents a method to conservatively apply uncertainties; but the ANS 1971 decay heat model is conservative and does not require verification for every reload core design. The 1971 decay heat standard plus heavy isotope actinide contribution has been used in Sequoyah Units 1 and 2 (Ref. 11). For the design calculations of B&W designed PWRs including LOCA analyses, the decay heat contribution of the heavy isotope actinides has been calculated with a heavy isotope decay heat model, referred to as the B&W heavy isotope model (Ref. 12). Figures A.3 and A.4 of the appendix show that the ANS 1971 decay heat standard plus the B&W heavy isotope actinide contribution bound the ANS 1979 decay heat plus 2 sigma uncertainty for a wide variation in fuel assembly enrichment and burnup. Therefore, this model will be used for all non-LOCA transients of the B&W-designed PWRs, except for MSLB. For MSLB at end of cycle for which it is desired to minimize heat input to the RCS, 0.9 times the ANS 1971 decay heat standard will be used. The staff finds this decay heat calculation for non-LOCA safety analyses acceptable.

3.0 REFERENCES

1. Letter from J. H. Taylor (BWNT) to U. S. Nuclear Regulatory Commission, "Submittal Of Topical Report BAW-10193P, RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurized Water Reactors, August 1995," JHT/95-85, August 14, 1995.

2. BAW-10164P-A, Rev. 3, "RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," July 1996.

3. BAW-10098, Rev. 1, "CADDS - Computer Application to Direct Simulation of Transients in PWRs With or Without Scram," January 1978.

4. BAW-10128, "TRAP2 - FORTRAN Program for Digital Simulation of the Transient Behavior of the Once-Through Steam Generator And Associated Reactor Coolant System," August 1976.

5. BAW-10156-A, Rev. 1, "LYNXT - Core Transient Thermal-Hydraulic Program," March 1991.

6. V. H. Ransom, et al, "RELAP5/MOD2 Code Manual," NUREG/CR-4312, EGG-2396, August 1985.

7. BAW-10168P-A, Rev. 3, "RSG LOCA - B&W Loss-of Coolant Accident Evaluation Model for Recirculating Steam Generator Plants," December 1996.

8. BAW-10169P, "RSG Plant Safety Analysis," October 1987.

9. BAW-10192P-A, "BWNT LOCA - BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants," June 1998.

10. Letter from J. J. Kelly (Framatome Technologies) to U. S. Nuclear Regulatory Commission, "Response to NRC's Request for Additional Information Regarding Topical Report BAW-10193P," May 4, 1999, FTI-99-1523.

11. Letter from J. H. Taylor (Framatome Cogema Fuels) to U.S. Nuclear Regulatory Commission, "Submittal of Topical Report BAW-10220P, 'Mark-BW Fuel Assembly Application for Sequoyah Nuclear Units 1 and 2,' March 1996," March 5, 1996, JHT/96-20.

12. Letter from J. J. Cudlin (Framatome Technologies) to U.S. Nuclear Regulatory Commission, "Additional Information on the Actinide Decay Heat Model in BAW-10193," August 30, 1999, FTI-99-2750.

4.0 CONCLUSION

The staff has reviewed the benchmarks of the RELAP5/MOD2-B&W code against various data from the OTSG tests and plant transients, and the calculations of the approved safety analysis codes. The good agreements of these benchmarks demonstrated acceptability of RELAP5/MOD2-B&W for performing safety analyses of non-LOCA events for the B&W-designed PWRs.

As discussed in Section 2.1.2 of this SE, the noding detail used for the benchmark of the IEOTSG test data was not sufficient to produce an accurate prediction of the primary-to-secondary heat transfer for the IEOTSG. FTG stated that before (or concurrent with) the licensing submittal of an IEOTSG plant safety analysis, an updated benchmark of the IEOTSG test data will be submitted for NRC review. Therefore, RELAP5/MOD2-B&W may not be used for analyses of PWRs with the IEOTSGs until FTG submits and the staff accepts an updated benchmark of the IEOTSG LOFW test.

Principal Contributor: Y. Hsii

Date: October 15, 1999

FRAMATOME
TECHNOLOGIES

May 4, 1999
FTI-99-1523

Project No. 693

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Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention: Mr. J. L. Birmingham

Subject: Response to NRC's Request for Additional Information Regarding Topical Report BAW-10193P.

Reference: J. L. Birmingham to R. Schomaker, "Request for Additional Information Regarding Topical Report BAW-10193P," March 1, 1999.

Gentlemen:

In the reference letter, the NRC Staff issued a Request for Additional Information (RAI) regarding BAW-10193P. Enclosed is a response to this RAI, which includes answers to the questions as well as proposed change pages to the topical report. The change pages also include a reanalysis of the RELAP5/MOD2-B&W benchmarks to the CADDS computer code. As we discussed in the February 22, 1999 telephone call, FTI has increased the number of high-pressure data points in the water property file used by RELAP5/MOD2-B&W. This has a small effect on the pressure solution in the pressurizer at pressures above 2400 psia.

FTI requests that in accordance with 10CFR2.790 the enclosed response to the NRC RAI be considered "Proprietary." Information supporting this request is included in the attached affidavit. One additional copy of the response is attached in which the proprietary material has been identified.

If you have any questions, please give the undersigned a call at 804-832-2964, or Mr. Marty Parece at 804-832-2474.

Very truly yours,

J. J. Kelly, Manager
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Response To NRC Request For Additional
Information On BAW-10193P

NRC Question 1:

To demonstrate RELAP5/MOD2-B&W's suitability for safety analyses of non-LOCA events for the B&W-designed PWRs, BAW-10193P provides benchmarks of the RELAP5/MOD2-B&W code by comparing its calculated results with test facility data, PWR plant data, and the CADDS and TRAP2 calculations of a few transients. These benchmark calculations used specific plant and system noding details (e.g., Figure 5-1 for the analyses of plant test data, Figures 6-1 and 6-18 for the analyses of control rod withdrawal and steam line break to compare with the CADDS and TRAP2 analyses, respectively) and specific options of the constitutive models and correlations in the code (e.g., use of the [c, e] CHF correlation for the steam generator shell side heat transfer, the multipliers developed for regions of small hydraulic diameters for the interphase drag in the slug and annular-mist flow regimes, and a [c, e] applied to the Chen boiling suppression factor).

To support the validity of the conclusions drawn from these benchmarks comparisons for application of RELAP5/MOD2-B&W to the safety analyses of the design basis events, you should document the RELAP5/MOD2-B&W safety analysis methodologies for various design basis non-LOCA events, such as a steam line break, reactivity insertion events, and loss of primary flow events, etc. The methodologies should include (a) noding details of the reactor and plant system, (b) specific options of constitutive models and correlations in RELAP5/MOD2-B&W, and (c) guidance on input assumptions, if different from the to-be-replaced CADDS and TRAP2 analyses, with respect to initial and boundary conditions, reactivity feedback coefficients, control systems, and other important parameters that affect plant response to the non-LOCA events. Also included should be the bases or justifications for the noding details, chosen options, and input assumptions, an explanation for any inconsistency with those used in benchmarking, and identification of plant-specific items which will be submitted and reviewed on a plant-specific basis.

Response:

An appendix will be added to the topical report. The appendix describes the noding details to be used to model the NSSS for various accidents and lists the options for constitutive models and correlations. In addition, guidance on input assumptions, initial conditions and boundary conditions is included. The appendix is attached herein for licensing review.

NRC Question 2:

With regard to the RELAP5/MOD2-B&W analyses of the once-through steam generator (OTSG) and integral economizer OTSG (IEOTSG) loss-of-feedwater flow (LOFW) tests described in Sections 4.4 and 4.6, respectively:

A. *You stated that, although the IEOTSG LOFW test procedure called for the closure of the steam generator model downcomer isolation valve to terminate feedwater for initiation of LOFW, the test data suggested that the feedwater isolation valve was closed instead, thus allowing feedwater to trickle into the tube region of the downcomer. The input boundary conditions used in the RELAP5/MOD2-B&W analysis include an estimate of* *the average rate of liquid displacement from the downcomer. Explain how the average liquid displacement rate was calculated and used to simulate the closure of the feedwater isolation valve in the test.*

Response:

Figure 4-10 in BAW-10193 shows that there was a time delay between restart of feedwater and restart of steaming. If the downcomer pipe was full, as it should have been, then steaming would have started immediately upon feedwater restart. We postulated that the time delay observed in the test was the time required to refill the downcomer pipe to make up for the mass lost during the test. The mass lost from the downcomer pipe was estimated by multiplying the feedwater flow at 35 seconds times the time delay of about 1.5 seconds. The resulting mass **[c, d]** was transferred from the downcomer pipe to the tube bundle over approximately 22 seconds, as shown in Figure 4-10 of the topical report.

B. *Figure 4-11 shows a deviation between the predicted and the observed steam generator primary outlet temperatures after 20 seconds where the predicted temperature rapidly approached the inlet temperature as the IEOTSG dried out, and the observed temperature remained much lower than the inlet temperature, indicating continued heat transfer. You stated that the continued heat transfer is not supported by the measured steam flow, and that the deviation between the predicted and the observed temperatures is primarily due to the RTD heat capacity, which caused a lag in the measured temperature response such that the actual fluid temperatures were higher than the recorded values during heatup and lower than recorded values during refill. (1) Provide an estimate of the lag in temperature measurement due to RTD heat capacity and exam whether the deviation is entirely caused by this effect. (2) Explain whether the RTD heat capacity effect also applies to the temperature measurement in the OTSG LOFW test.*

Response:

1) The lag constant on the RTDs is estimated to be approximately **[d]** seconds. To determine what the observed values would be if the RELAP5/MOD2-B&W predictions were "measured" by the RTDs, the code prediction of primary outlet temperature in Figure 4-11 of the topical report was run through a Laplace transform that simulated a lag of five seconds. Figure 1 (attached) shows the resulting adjusted temperature. When compared with the test data, the adjusted temperature prediction is consistent with the steam flow prediction. In other words, RELAP5/MOD2-B&W overpredicts the steam flow during the dryout portion of the test. This should result in early dryout—as observed in Figure 4-10 in BAW-10193—and should result in a low prediction of primary outlet temperature caused by an overprediction in heat transfer. In fact, the adjusted RELAP5/MOD2-B&W temperature prediction shows this trend as compared with the measured values. FTI concluded that the noding detail used to predict the IEOTSG test data is too crude to provide an accurate result and that additional modeling is required.

   The benchmark will not be refined at this time, because FTI has no plans to perform IEOTSG plant safety analysis. Prior to (or concurrent with) the licensing submittal of an IEOTSG plant safety analysis, FTI will submit for NRC review an updated benchmark of these test data.

2) With regard to the OTSG LOFW test, because the test used the same equipment as the IEOTSG test, the RTD thermal lag applies to the OTSG LOFW test. As stated above, the lag constant is approximately [d] seconds. Figure 2 shows a comparison of the primary outlet temperature recorded at the test with the RELAP5/MOD2-B&W prediction from Figure 4-7 in BAW-10193. Also shown is the code prediction of primary outlet temperature after it was run through a Laplace transform that simulated a [d] lag. The adjusted temperature—which mimics the value that would be observed if the RELAP5/MOD2-B&W predicted outlet temperature was "measured" by the RTD—agrees well with the actual RTD output. This indicates that the RELAP5/MOD2-B&W prediction of transient heat transfer is in good agreement with the test. This is consistent with the predicted steam flow shown in Figure 4-6 of BAW-10193. The measured steam flow provides a direct indication of the primary-to-secondary heat transfer following a LOFW. The RELAP5/MOD2-B&W prediction compares well with the data.

C. *For the RELAP5/MOD2-B&W analysis of the OTSG LOFW test, you attributed the differences between the predicted steam generator primary outlet temperature and the observed ARC data shown in Figures 4-7 after 20 seconds in the transient primarily to the sudden changes in the heat transfer as the control volumes in the nucleate boiling region systematically dryout and, later, refill. You also stated that increasing the number of control volumes in the nucleate boiling region would result in the prediction of primary outlet temperature approximately the same as the current prediction. For the analysis of the IEOTSG LOFW test, you attributed the cause of the code prediction of dryout time being 2 seconds less than the observed dryout time of about* [d] *seconds to the overprediction of the steam flow from the IEOTSG during the dryout period, which you attributed to the mixture level crossing the control volume boundaries. Discuss either (a) why, despite its large differences in comparison to data, the RELAP5/MOD2-B&W code and noding details are acceptable for analyses of non-LOCA events of the B&W-designed PWRs, or (b) any improvements to the RELAP5/MOD2-B&W code or inputs are needed to improve prediction of the steam generator response to LOFW transients.*

Response:

As stated in the response to Question 2.B, the noding arrangement used in the IEOTSG benchmark appears to overpredict the primary-to-secondary heat transfer, overpredict the steam flow and underpredict the dryout time. When the code prediction is adjusted to account for the RTD time constant in the test, the prediction of primary outlet temperature is clearly too low during the dryout phase. The benchmark will not be refined at this time, because FTI has no plans to perform IEOTSG plant safety analysis. Prior to (or concurrent with) the licensing submittal of an IEOTSG plant safety analysis, FTI will submit for NRC review an updated benchmark of these test data.

With regard to the OTSG LOFW test, the steam flow and primary outlet temperature predictions show step changes as control volume boundaries are crossed by the secondary mixture. Notwithstanding, the steam flow prediction in Figure 4-6 of BAW-10193 compares well with the measured data, albeit sometimes greater than and occasionally less than the measured values. Similarly, after the RELAP5-predicted primary outlet temperature was adjusted to account for the effects of the facility RTD (Figure 2), it is demonstrated that the computer code with this control volume arrangement provides a good prediction of the primary-to-secondary heat transfer during this test. The ability of this model to adequately predict transient heat transfer is further demonstrated in the benchmarks to plant transients wherein the reactor trip times and steam generator liquid levels were predicted properly (Sections 5.2 and 5.3 of BAW-10193). Consequently, no changes are required to the OTSG model or the computer code.

The topical report text for the OTSG LOFW test will be changed to state that the sharp discontinuities in the steam flow occur as the mixture level crosses control volume boundaries. The text will also be modified to attribute the differences between predicted and measured primary outlet temperature to the thermal lag of the RTD. The proposed change pages are attached herein for licensing review.

NRC Question 3:

Regarding the RELAP5/MOD2-B&W analysis of the TMI-2 LOFW event, Figure 5-10 indicated agreement between the code prediction of the pressurizer level and the plant data up to the reactor trip at about 20 seconds. After the reactor trip, the code overpredicted the plant data. You concluded, based on the experience with the B&W plants showing the pressurizer level should decrease by 5.5 inches for every degree decrease in average system temperature, that the plant data are probably not reliable during this period. Are the pressurizer level data prior to the reactor trip reliable, and why? What conclusion can one draw when the code predictions are in good agreement with the plant data if the data are not reliable?

Response:

The pressurizer level indication at TMI-2 is temperature compensated. An RTD in the pressurizer provides a reference temperature to obtain a density correction factor from a function generator. Experience indicates the instrumentation is relatively accurate for pressurizer level changes of less than [c] and pressure changes of less than [c] The plant response prior to 24 seconds is within these limits and the measured pressurizer level during this period follows the rule of thumb for B&W-designed plants. Consequently, the plant data appear to be reliable prior to 24 seconds (which is approximately 15 seconds after reactor trip).

After 24 seconds, the system pressure decreases from 1950 psia to 1670 psia over the next 100 seconds. Under these conditions experience shows the instrumentation will indicate a value lower than the actual water level in the pressurizer. This appears to hold true for the TMI-2 pressurizer level data (i.e., the actual water level in the pressurizer is greater than the indicated value). However, the magnitude of the measurement error in the TMI-2 data is greater than what would be expected. Therefore, the possibility of a calibration error that would render the entire pressurizer level trace unreliable cannot be eliminated.

Topical report section 5.2.2 will be revised to state that comparison of the predicted pressurizer level to the temperature-compensated pressurizer level data might not be valid because of the apparent large uncertainty in the plant data. The trend of the RELAP5/MOD2-B&W prediction of pressurizer level is consistent with the reactor coolant temperatures and yielded a good match to the reactor coolant system pressure recorded during the event. The proposed change pages are attached herein for licensing review.

NRC Question 4:

In the RELAP5/MOD2-B&W analysis of the Rancho-Seco loss of ICS power event (Section 5.3), the code calculations of the primary pressure agreed well with the plant data during the first 500 seconds of RCS depressurization up to pressure stabilization due to the flashing of the water in the reactor vessel upper head and HPI injection. In the subsequent period when the HPI flow exceeding the primary coolant contraction rate, the code underpredicted the RCS repressurization, which you said to be most likely caused by the uncertainty in measured HPI flows that were input to the model, and by absence of reactor coolant pump seal injection flow in the model. You stated that this conclusion is supported by the predicted recovery of pressurizer level compared with the plant data.

A. *Please elaborate on the measured HPI flow uncertainty you were referring to, including the magnitude and sources of uncertainties. Explain why the HPI injection rate input uncertainty causes underprediction of RCS repressurization, but provided relatively good agreement of in the predicted and observed RCS pressure in the first 500 seconds prior to RCS repressurization. Explain why this good agreement is not just a coincidence in light of the HPI input uncertainty.*

   Response:

   Formation of the reactor vessel head void was the primary reason the depressurization of the RCS was arrested. Although HPI provides a significant addition of mass to the RCS, the contraction of the RCS fluid was much greater than the volume of HPI put into the system. Consequently, uncertainties in HPI flow have little effect on the RCS pressure prediction prior to termination of the overcooling. This is demonstrated in Figure 5, which shows the calculated pressure response with an increase in HPI flow of approximately 80 gpm between 200 and 800 seconds. Increasing HPI flow has little effect on the pressure prediction during the depressurization because the dominant effect is the contraction of the primary system due to the overcooling. However, once the cooling of the RCS is reduced, the effects of HPI flow on the pressure and pressurizer level increase in significance.

   Relative to the HPI flow measurement uncertainty, FTI did not perform a detailed uncertainty analysis. There is no accurate way to estimate the uncertainty. FTI inspected the HPI flow data (Figure 3) and reviewed the HPI system piping isometrics for Rancho-Seco. It appeared that the measured flow for the "A" loop was substantially greater than the other loops because the flow was delivered through an HPI nozzle as well as the make-up nozzle (lowered-loop B&W-designed PWRs use HPI pumps for normal makeup). There did not appear to be a good reason for the difference in flows between the "B", "C", and "D" loops. Therefore, the flows in the "C" and "D" loops were set equal to the values measured in the "B" loop (Figure 4) from the time of SFAS until the operator terminated flow in those loops (reported in the plant sequence of events as approximately 16.5 minutes). Reanalysis of the event yielded the pressure and pressurizer level predictions in Figures 5 and 6, respectively. Note that the pressure and pressurizer level predictions improve after 500 seconds, but prior to 500 seconds there is minimal change.

Regarding the reactor coolant pump seal injection flow, the design value is approximately seven gpm (net) per pump. FTI was unable to determine if the measured HPI flow included this 28 gpm of seal injection flow.

B. *Since the engineered safety features actuation system (ESFAS) was not actuated until about 200 seconds into the transient when the RCS pressure decreased to the ESFAS actuation threshold, why were mass flow rates of about 35 lbm/s for high pressure injection train A (HPI-A) and about 5 lbm/s for HPI-D shown in Figure 5-11 prior to 200 seconds? Are these HPI flow uncertainties included in your conclusion? Were these HPI mass flow rates prior to 200 seconds modeled in RELAP5/MOD2-B&W input as you stated that the HPI injection flow were modeled directly from the plant recall computer data? If so, is it not that the depressurization would be underpredicted?*

Response:

ESFAS occurred at 200 seconds, initiating flow through all four HPI nozzles. The flow prior to that time was due to operator action to begin flow to the "A" and "D" nozzles. These actions are called out in Table 5-12 of the topical report. The recorded flows from the plant computer prior to 200 seconds were input to the RELAP5/MOD2-B&W prediction.

C. *How does the delay in the predicted recovery of pressurizer level compared with the plant data support your conclusion that underprediction of the repressurization is caused by HPI uncertainty and absence of modeling pump seal injection flow, instead of any other causes?*

Response:

The conclusion was based originally on review of the *rate* of pressure increase from 700 to 900 seconds. During this time period the RCS temperature was approximately constant. Therefore, the only contribution to RCS liquid volume was HPI flow. Since the rate of pressure increase—apparently caused by refilling the system and compressing the pressurizer bubble—was too low in the prediction, FTI concluded that the HPI flow in the simulation was too low.

Based on the comparison of results shown in Figures 5 and 6 with the predictions in Figures 5-14 and 5-18 of BAW-10193, FTI continues to attribute the underprediction in pressure and pressurizer level to the uncertainties in HPI flow rate. An increase in HPI flow rate of 80 gpm resulted in an improvement in the predicted RCS pressure and pressurizer level. Although FTI considers the HPI flow rate to be the dominant uncertainty, there are other sources of uncertainty that could cause deviations from the data:

1. Pressurizer level measurement uncertainty could be as much as 13 inches of standard water.

2. RCS volume used in the model could differ from the plant by up to one percent because of geometrical uncertainties and because there is no way to

model in RELAP5/MOD2-B&W the contraction of the RCS metal that occurs during cooldown.

NRC Question 5:

With regard to the boundary conditions (Section 5.5.1) used in the benchmark against TMI-1 natural circulation test:

A. *Explain why the correction factor of 1.12 used for the reactor power input to the code resolves the neutron detector calibration problem where the reactor vessel downcomer fluid temperature during the test was colder than the temperature at which the out-of-core neutron detectors were calibrated.*

   Response:

   The power values recorded during the test were taken from the out-of-core neutron detectors. These detectors measure the fast neutron flux entering the detectors and the associated instrumentation converts the value to a power level reading. If the downcomer fluid temperature is less than the value at which the detectors were calibrated, more neutrons are moderated in the downcomer fluid and fewer fast neutrons reach the detector. This leads to a measured power that is less than the actual power level.

   After the test was performed, GPUN estimated that the actual power was 1.12 times greater than the measured values. This was later verified using a FTI-proprietary correlation developed to determine the power measurement error. The correlation is of the form:

   c, e

So, the correction factor to be applied to the measured power to obtain actual core power is 1/0.891 or 1.12.

B. *Explain why (1) the RELAP5/MOD2-B&W input of the setpoints shown in Table 5-14 for the pressurizer heater banks 3 and 4 are 15 psi lower than the plant setpoints, and (2) the input steam generator level boundary condition shown in Figure 5-20 is different from the plant data.*

Response:

The RELAP5/MOD2-B&W input of the setpoints for pressurizer heater banks 3 and 4 are 15 psi lower than the plant setpoints because of an input error in the simulation. The error was discovered during preparation of BAW-10193. Although this error could be partly responsible for the underprediction in RCS pressure after 1000 seconds (heater bank 4 was off in the simulation when it should have been on), FTI decided not to re-run the transient because we thought the effect on pressure would be small.

In regard to the steam generator level boundary condition, no attempt was made to mimic the level observed in the test. The plant control system maintained the steam generator level at 50 percent ± 2 percent. Four percent on the operate range level indication corresponds to approximately 12 inches, which is less than the height of a steam generator control volume (approximately 5 feet) in this region of the steam generator model. Because this small variation in water level has no effect on the calculation, a simplified control system was used in the simulation to maintain the steam generator level near 50 percent (the target value for the test).

Figure 1. Comparison of Measured and Predicted Primary Outlet Temperatures For the 19-Tube Model IEOTSG LOFW Test

d

Figure 2. Comparison of Measured and Predicted Primary Outlet Temperatures For the 19-Tube Model OTSG LOFW Test

d

Figure 3. HPI Flow Rates Recorded During the Rancho-Seco Loss of ICS Power Event And Used in BAW-10193

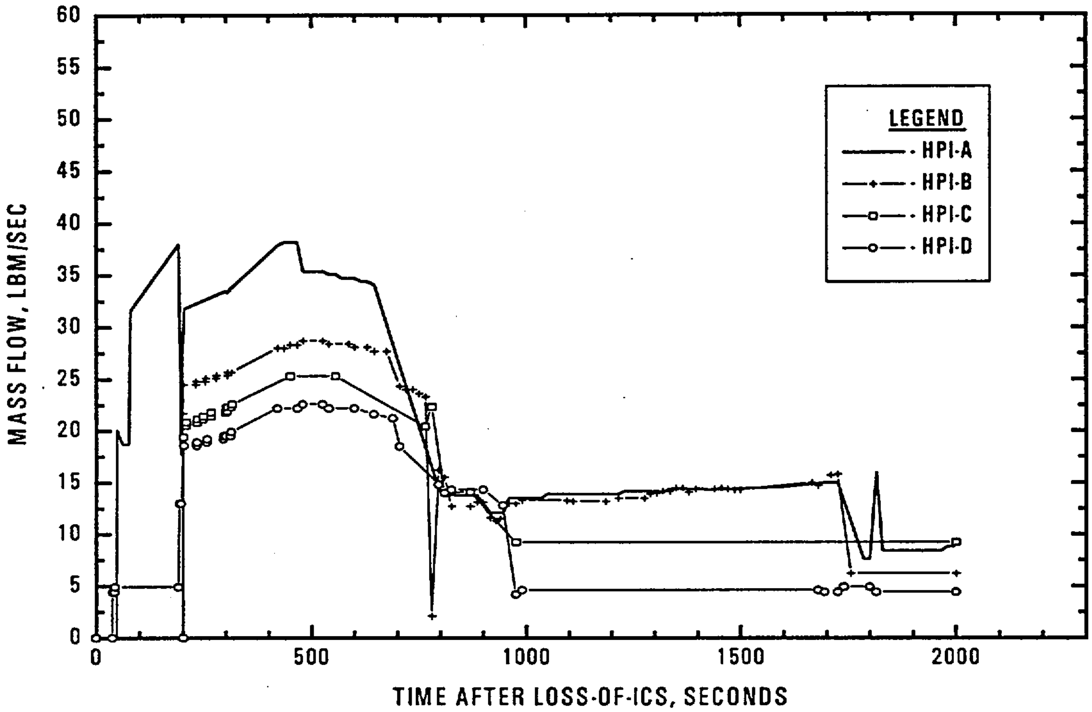


Figure 4. Revised HPI Flow Rates For Reanalysis of the Rancho-Seco Loss of ICS Power Event

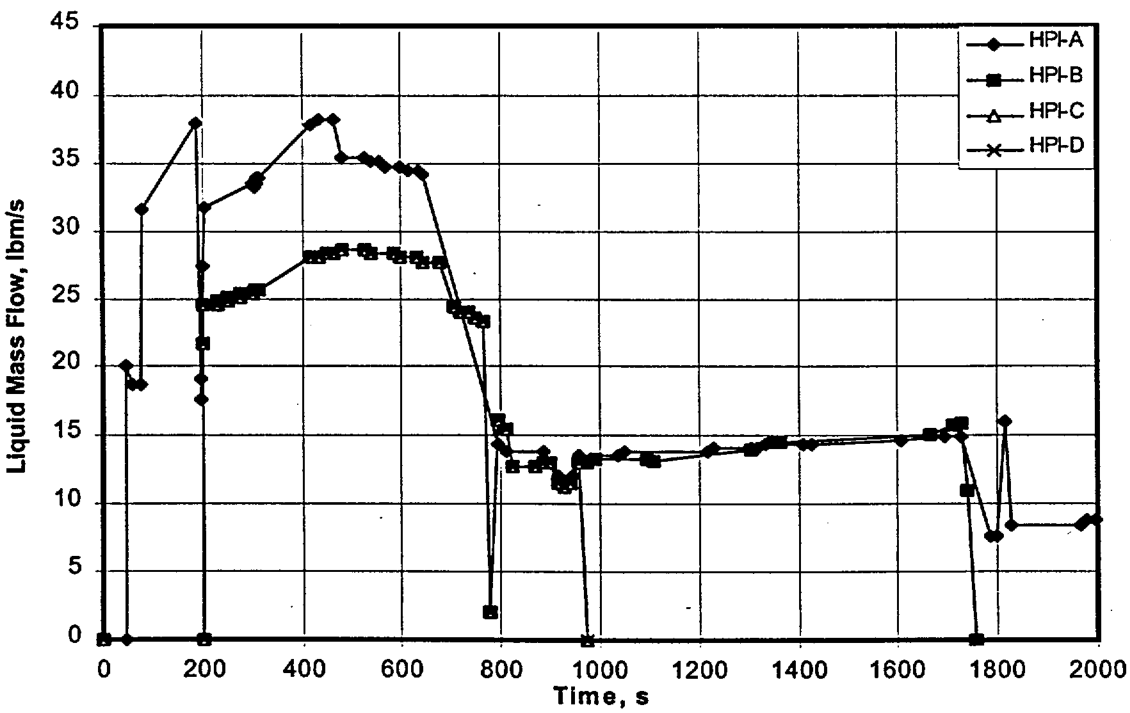


Figure 5. Predicted Reactor Coolant System Pressure for the Rancho-Seco Loss of ICS Power Event Using Revised HPI Flow Rates in Figure 4.

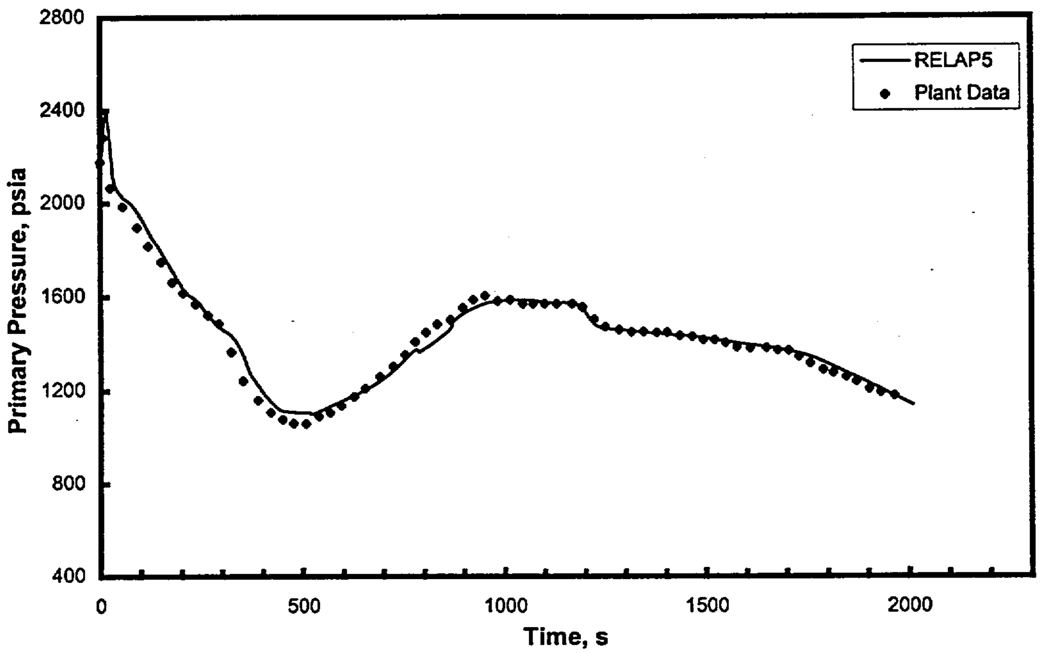


Figure 6. Predicted Pressurizer Level Response for the Rancho-Seco Loss of ICS Power Event Using Revised HPI Flow Rates in Figure 4.

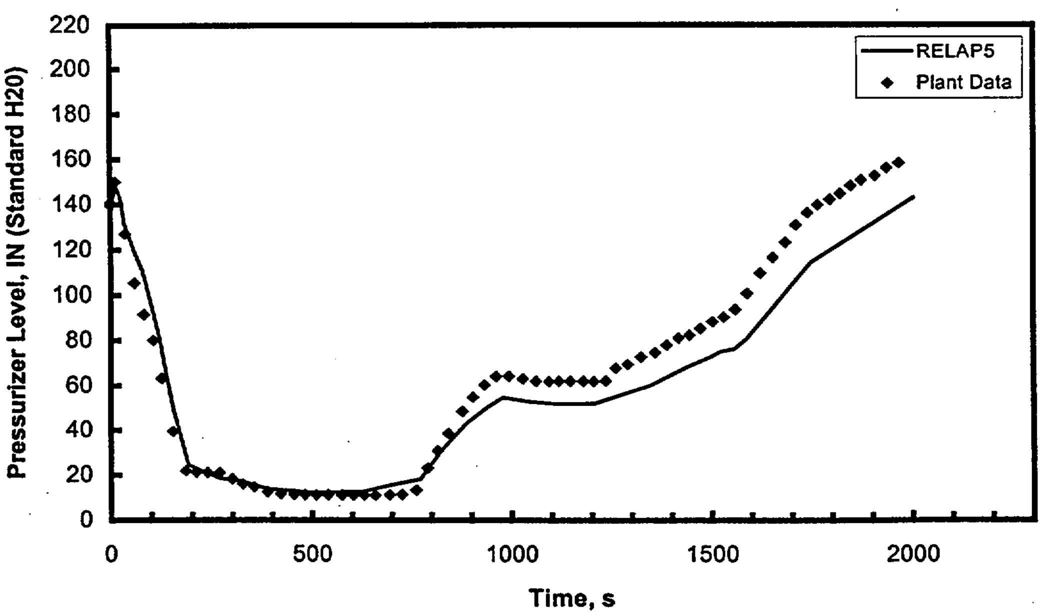
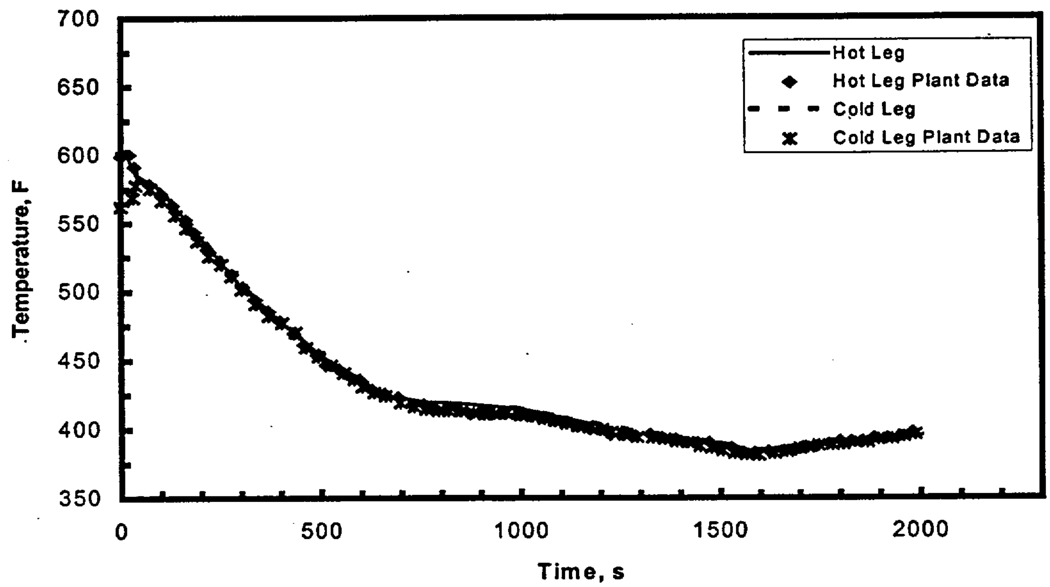


Figure 6. Predicted Reactor Coolant Temperatures for the Rancho-Seco Loss of ICS Power Event Using Revised HPI Flow Rates in Figure 4.



ACKNOWLEDGMENT

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RECORD OF REVISION

Rev. No.	Change Sect/Para.	Description of Change
0		Initial Release

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1. EXECUTIVE SUMMARY

In the mid-1970's, the Nuclear Power Division of B&W (currently Framatome Technologies Group) developed the CADDS and TRAP2 computer codes to perform non-loss-of-coolant accident (Non-LOCA) analyses of B&W-designed PWRs. It is the intent of FTG to replace the CADDS and TRAP2 computer codes with RELAP5/MOD2-B&W for performing non-LOCA safety analyses on B&W-designed PWRs. RELAP5/MOD2-B&W is a state-of-the-technology, non-equilibrium, non-homogeneous, six-equation thermal-hydraulic simulation program. The ability of RELAP5/MOD2-B&W to model complex thermal-hydraulic phenomena outstrips that of either CADDS or TRAP2.

FTG compared RELAP5/MOD2-B&W predictions with the following tests, plant transients, and computer code simulations:

1. Model 19-tube once-through steam generator (OTSG) steady-state boiling length tests.
2. Model 19-tube OTSG loss-of-feedwater test.
3. Model 19-tube integral economizer OTSG steady-state boiling length tests.
4. Model 19-tube integral economizer OTSG loss-of-feedwater test.
5. Three Mile Island Unit 2 loss-of-feedwater event of March 26, 1979.
6. Rancho-Seco loss-of-ICS power event of December 26, 1985.
7. Four-pump coastdown data from Oconee Unit 1 and Crystal River Unit 3.
8. Three Mile Island Unit 1 natural circulation test of October 7, 1985.
9. CADDS predictions of the startup event for a 177 fuel assembly (FA) plant.
10. TRAP2 predictions of steam line break for a 177 FA plant.

The benchmarks to the 19-tube steam generator data demonstrate that the RELAP5/MOD2-B&W computer code properly predicts the secondary side nucleate boiling length as a function of load. Accurate prediction of the heat transfer surface area in the once-through steam generator is necessary for the accurate prediction of the B&W-designed plant response to upset conditions. Furthermore, the RELAP5/MOD2-B&W comparisons with the 19-tube steam generator loss-of-feedwater tests show that the code correctly predicts the primary-to-secondary heat transfer during steam generator dryout and refill.

The four plant transients that were benchmarked using RELAP5/MOD2-B&W exhibit the phenomena encountered by a B&W-designed PWR during loss-of-feedwater, turbine trip, severe overcooling, reactor coolant pump trips, and primary system natural circulation cooling. In each case, the plant response predicted by RELAP5/MOD2-B&W was in good agreement with the plant data.

The RELAP5/MOD2-B&W predictions of core response for the startup accident are similar to the CADDS predictions for a range of reactivity insertion rates. In fact, RELAP5/MOD2-B&W conservatively predicts peak core thermal power as compared with CADDS. RELAP5/MOD2-B&W predicts a lower peak reactor coolant system pressure as compared to CADDS because the RELAP5/MOD2-B&W prediction of system pressure properly reflects the effects of changes in primary system fluid expansion rates and includes pressurizer steam condensation effects.

The RELAP5/MOD2-B&W predictions of the plant response to main steamline breaks were in good agreement with those of TRAP2. The calculations of break flow, primary pressure, secondary pressure, primary system temperature, reactor trip time, ESFAS time, core reactivity, and core power matched well. The differences between the TRAP2 and RELAP5/MOD2-B&W predictions arise because RELAP5/MOD2-B&W mechanistically calculates the phase-slip in the steam generator and feedwater piping control volumes. The TRAP2 fluid conditions were effectively homogeneous, resulting in an overprediction of the steam generator boiling length and a delay in steam generator dryout.

The benchmarks to test facilities and to actual plant transients demonstrate that RELAP5/MOD2-B&W properly predicts the phenomena exhibited by B&W-designed PWRs.

Furthermore, RELAP5/MOD2-B&W, given the same conservative boundary conditions and input assumptions, calculates bounding results similar to the current NRC-approved computer codes, TRAP2 and CADDS. Therefore, RELAP5/MOD2-B&W is appropriate for performing Non-LOCA safety analyses of B&W-designed PWRs.

2. INTRODUCTION

The Babcock & Wilcox-designed Nuclear Steam Supply System (NSSS) is a two-loop pressurized water reactor (PWR) with one hot leg and two cold legs in each loop. The Babcock & Wilcox (B&W) design is unique because it employs once-through steam generators (OTSGs). The OTSG is a counter-flow, single-pass, tube and shell heat exchanger that produces superheated steam at a constant secondary pressure over the entire load range. The boiling heat transfer area and secondary inventory vary with load, requiring special modeling capabilities to properly predict plant response.

In the mid-1970's, the Nuclear Power Division of B&W (currently Framatome Technologies) developed the CADDS and TRAP2 computer codes to perform non-loss-of-coolant accident (Non-LOCA) analyses of B&W-designed PWRs. CADDS[1] (Computer Application to Direct Digital Simulation of Transients in Water Reactors With and Without Scram) is used to analyze control rod withdrawal, control rod ejection, loss of primary flow, reactor coolant pump startup, boron dilution, and anticipated transients without scram. The TRAP2[2] (Transient Reactor Analysis Program) computer code, developed from CRAFT[3], is used to calculate the core power and system responses to steam line break, turbine trip, loss of feedwater, feedwater line break, and steam generator tube rupture accidents.

It is the intent of FTG to replace the CADDS and TRAP2 computer codes with RELAP5/MOD2-B&W for performing non-LOCA safety analyses on B&W-designed PWRs. RELAP5/MOD2-B&W[4] is a state-of-the-technology, non-equilibrium, non-homogeneous, six-equation thermal-hydraulic simulation program. The ability of RELAP5/MOD2-B&W to model complex thermal-hydraulic phenomena outstrips that of either CADDS or TRAP2. Furthermore, it is preferable to perform safety analyses with a single computer code that is familiar to the nuclear industry. This increases regulatory acceptance and minimizes training and maintenance costs.

The objective of this topical report is to justify the use of RELAP5/MOD2-B&W for Non-LOCA safety analyses of B&W-designed PWRs. It is demonstrated through benchmarks to test facilities and to actual plant transients that RELAP5/MOD2-B&W properly predicts the phenomena exhibited by B&W-designed PWRs. It is also demonstrated that RELAP5/MOD2-B&W, given the same conservative boundary conditions and input assumptions, calculates bounding results similar to the current NRC-approved computer codes, TRAP2 and CADDS. Consequently, this report and the code topical report form the basis of justification for use of the RELAP5/MOD2-B&W for Non-LOCA safety analyses of B&W-designed PWRs.

It should be noted that RELAP5/MOD2-B&W will be used in the same way that TRAP2 and CADDS are currently used to analyze non-LOCA events. Specifically, RELAP5/MOD2-B&W will be used to predict the primary system response, secondary system response, and average core power response to non-LOCA events. Minimum departure from nucleate boiling ratio (MDNBR) in the core hot channel will be calculated using the LYNXT[5] computer code. Local core power distributions, that can affect MDNBR in the hot channel, will be determined using the NEMO[6] computer code.

Section 3 of this report briefly discusses the facility tests, plant transients and tests, and the TRAP2 and CADDS analyses that were benchmarked with RELAP5/MOD2-B&W. Benchmarks to test facility data are shown in Section 4. The comparisons to plant transient data are provided in Section 5. Comparisons to TRAP2 and CADDS predictions are in Section 6. References are listed in Section 7.

3. SELECTION OF BENCHMARKS

FTG has benchmarked a number of facility tests, plant transients, plant tests, and computer code predictions using RELAP5/MOD2-B&W. The benchmarks were chosen to show the capability of RELAP5/MOD2-B&W to predict a range of non-LOCA events including over-heating, over-cooling, loss-of-reactor coolant flow, primary system natural circulation, and reactivity insertion events. Ten different benchmarks are presented in this report. RELAP5/MOD2-B&W predictions are compared with the following tests, plant transients, and computer code simulations:

1. Model 19-tube OTSG steady-state boiling length tests.
2. Model 19-tube OTSG loss-of-feedwater test.
3. Model 19-tube integral economizer OTSG steady-state boiling length tests.
4. Model 19-tube integral economizer OTSG loss-of-feedwater test.
5. Three Mile Island Unit 2 (TMI-2) loss-of-feedwater event of March 26, 1979.
6. Rancho-Seco loss-of-ICS power event of December 26, 1985.
7. Four-pump coastdown data from Oconee Unit 1 and Crystal River Unit 3.
8. Three Mile Island Unit 1 natural circulation test of October 7, 1985.
9. CADDS predictions of the startup event for a 177 fuel assembly (FA) plant.
10. TRAP2 predictions of steam line break for a 177 FA plant.

RELAP5/MOD2-B&W was benchmarked to the model 19-tube steam generator boiling length tests to show that the code properly predicts the boiling length as a function of load. The proper prediction of boiling length is most important in accurately predicting OTSG transients.

The benchmarks to the model 19-tube steam generator loss-of-feedwater tests demonstrate that the code adequately calculates the primary-to-secondary heat transfer during boil-down and dryout following a loss-of-feedwater. Accurate calculation of the primary-to-secondary heat transfer is important to the simulation of system response to this overheating transient, which is typically required to be analyzed for all pressurized water reactors as an event of moderate frequency.

The TMI-2 event of March 26, 1979 was also an overheating event. The benchmark to this event shows that RELAP5/MOD2-B&W properly predicts the primary and secondary system pressurization rates following the turbine trip. This benchmark also demonstrates the ability of RELAP5/MOD2-B&W to predict properly full-scale steam generator dryout following a loss-of-feedwater.

The Rancho-Seco loss of ICS power event of December 26, 1985 began as a mild overheating transient due to a reduction in feedwater. Following reactor trip, a severe overcooling transient was experienced because of steam discharge through atmospheric relief valves that had failed open and because of overfeeding the steam generators with auxiliary feedwater. The benchmark of this event shows the ability of RELAP5/MOD2-B&W to properly predict the system responses during all phases of this complex plant transient.

The benchmark of reactor coolant pump speed following a four-pump coastdown demonstrates that the pump inertias, pump frictional torque values and reactor coolant loop flow resistances input to the RELAP5/MOD2-B&W plant model yield an accurate calculation of the system flow rate. Similarly, the benchmark of the TMI-1 natural circulation test of October 7, 1985 demonstrates that RELAP5/MOD2-B&W, with this plant model, properly predicts the transition from forced primary system flow conditions to natural circulation conditions.

The intent of the RELAP5/MOD2-B&W comparison with CADDS is to show that, with the same initial and boundary conditions, RELAP5/MOD2-B&W will predict system and core power responses similar to those predicted by CADDS. The CADDS computer code is primarily used to calculate the system and core power responses to reactivity insertion events. The code comparison was performed for control rod assembly bank withdrawal from a low power condition (start-up event) because this reactivity insertion event is the limiting primary system overpressure event for the B&W 177 FA plant design.

The RELAP5/MOD2-B&W comparison with TRAP2 shows that, with the same initial and boundary conditions, RELAP5/MOD2-B&W will predict system and core power responses similar those predicted by TRAP2. BWNT uses the TRAP2 computer code to calculate the system and core power responses to upsets in primary-to-secondary heat transfer (e.g., steam line break, loss-of-feedwater, turbine trip, feedwater line break). TRAP2 predictions of the main steam line break response were selected for comparison with RELAP5/MOD2-B&W because this severe overcooling event sets the core design limit for end-of-cycle moderator temperature coefficient.

4.0 BENCHMARKS TO TEST FACILITY DATA

Babcock & Wilcox performed numerous tests on 19-tube and 37-tube models of the once-through steam generator at the Alliance Research Center (ARC) Nuclear Steam Generator Test Facility (NSGTF). The objectives of these tests were to demonstrate the characteristics of B&W-designed steam generators and to provide data for computer code development. Four sets of model 19-tube tests were simulated with RELAP5/MOD2-B&W. The first set of benchmarks were to steady-state tests to show the ability of the code to predict the shell side nucleate boiling length at various power levels for the once through steam generator (OTSG). The second benchmark was a comparison with a loss-of-feedwater (LOFW) flow test to demonstrate the ability of RELAP5/MOD2-B&W to predict boil-down and refill of an OTSG. The last two sets of benchmarks are to steady-state boiling length tests and to a LOFW test on the integral economizer (non-aspirated) OTSG design.

4.1 Facility Description

The ARC NSGTF, Figures 4-1 and 4-2, provided the capability to test at full system pressure and temperature conditions. The primary system was a closed circuit test loop with a natural gas-fired furnace, a pressurizer, flow control valves, flow measuring devices, and a water conditioning system. The secondary system was also closed and consisted of steam flow control valves and measuring devices, feedwater heater and control valves, a flash tank, back pressure control valves, and a water conditioning system. The model steam generator was a single-pass, counterflow, tube and shell heat exchanger with 19 full-length tubes, 5/8-inch in diameter on a 7/8-inch triangular pitch. The tube bundle was enclosed in a hexagonal shell 3.935 inches across flats held in place by 16 tube support plates, each spaced at 3-foot intervals. The tube support plates were drilled in a manner to simulate the broached pattern of a full-size tube support plate.

Primary flow entered at the top and exited at the bottom. The secondary flow entered via an external downcomer. The flow then entered the bottom of the tube bundle and exited at the top as superheated steam. When run in the OTSG mode, a steam bleed from the tube region, which simulated the aspirator, raised the feedwater temperature to saturation before entering the tube nest. In the IEOTSG mode, the steam bleed is closed and the subcooled feedwater enters at the bottom of the tube nest.

4.2 RELAP5/MOD2-B&W Model Description

The 19-tube steam generator was modeled with eleven axial control volumes in the primary tube region and the secondary shell region as shown on Figure 4-3. Similarly. eleven heat structures were used to simulate the primary-to-secondary heat transfer. The external downcomer was modeled with five axial control volumes. The feedwater aspirator was provided by a single junction component that connected the tube bundle region to the downcomer. Feedwater inlet temperature and flow, secondary pressure, primary inlet temperature and flow, and primary pressure were input as boundary conditions using time-dependent volume and time-dependent junction components.

Special features, available in RELAP5/MOD2-B&W, were employed in the 19-tube OTSG model. First, the [c] critical heat flux correlation was used on the shell side of the tube heat structure to provide a better prediction of the dryout point in the OTSG. Second, the interphase drag in the slug and annular-mist flow regimes was reduced by use of the default multipliers developed for regions of small hydraulic diameters. This model produces results similar to the Wilson bubble rise model for pressures above 200 psia and provides a better prediction of liquid mass in the tube region. Finally, a linear ramp was applied to the Chen boiling suppression factor such that it was reduced from the calculated value to zero over a void fraction of [c, e] This prevented the Chen heat transfer coefficient from becoming unrealistically large as the void fraction approached 1.0 on the shell side of the OTSG.

4.3 Comparison With OTSG Steady-State Boiling Length Tests

In 1969 the ARC performed a series of steady-state tests on the model 19-tube OTSG to determine the nucleate boiling length as a function of scaled power level.[7] Each test was performed with primary pressure, primary inlet conditions, feedwater conditions, and secondary pressure held constant. The nucleate boiling length (dryout location) was determined from primary side and secondary side tube thermocouples.

The boundary conditions for five, [d] MWt plant-scaled tests are shown in Table 4-1. Using these boundary conditions, a steady-state calculation of each test was performed using RELAP5/MOD2-B&W. Also, for comparison purposes, each test was simulated with the RELAP5/MOD2 Cycle 36.05. The calculated boiling lengths are compared with the observed values in Table 4-2 and Figure 4-4.

The results show that the boiling lengths predicted by RELAP5/MOD2-B&W are in good agreement with the data over the range of simulated power levels. Furthermore, the RELAP5/MOD2-B&W predictions represent a significant improvement over the base RELAP5/MOD2 results at power levels less than [c,d] percent of scaled full power. This is primarily due to the use of the [c] critical heat flux (CHF) correlation in the RELAP5/MOD2-B&W simulation. That correlation, developed from heated rod bundle dryout data, calculates a higher CHF value at reduced feedwater flow rates; whereas, the Biasi-Zuber CHF correlation combination used in RELAP5/MOD2 predicts early dryout in an OTSG as the feedwater flow (power level) decreases.

4.4 Comparison With OTSG LOFW Test

The ARC performed several loss-of-feedwater tests on the model 19-tube model OTSG[8] One LOFW test, Run 29, performed on December 16, 1977, was benchmarked with RELAP5/MOD2-B&W. This test was a loss-of-feedwater from scaled full power conditions consistent with a [d] MWt plant.

The model OTSG was initialized to full scaled power consistent with a [d] MWt plant. The test was initiated by the simultaneous trip of the feedwater pump and closure of the feedwater isolation valve. The OTSG was allowed to boil dry. After the OTSG boiled dry, feedwater was turned on by starting the feedwater pump and by opening the feedwater isolation valve.

An attempt was made to hold the primary inlet temperature, primary flow and secondary pressure constant during the test. Primary outlet temperature, secondary steam flow, and secondary steam temperature were measured and recorded during the test.

The RELAP5/MOD2-B&W model was initialized to the test initial conditions as shown in Table 4-3. The predicted primary and secondary fluid temperatures are compared in Figure 4-5 with the measured values just prior to test initiation. The feedwater flow, primary inlet temperature, primary flow, and secondary pressure values that were measured during the test were input as boundary conditions. The calculated steam flow and primary outlet temperature are compared to the measured data in Figures 4-6 and 4-7, respectively. The predicted steam flow is in good agreement with the data, indicating that the calculated heat transfer is similar to that observed during the test. The sharp discontinuities in calculated steam flow occur as the mixture level crosses control volume boundaries. The addition of control volumes in the nucleate boiling region would decrease the magnitude of the step changes, but the number of steps would increase. The resulting predictions of heat transfer and primary outlet temperature would be approximately the same as the current prediction.

The differences between predicted and observed primary outlet temperatures are primarily due to the resistance thermal detector (RTD) heat capacity. The heat capacity of the RTD caused a lag in the measured temperature response such that the actual fluid temperatures were higher than the recorded values during heat-up and lower than the recorded values during the refill. If a thermal lag were applied to the code prediction to account for RTD heat capacity, a good match would be obtained with the measured data.

4.5. Comparison With IEOTSG Steady-State Boiling Length Tests

In June of 1971 the ARC performed a series of steady-state tests to determine the thermal performance of the integral economizer once-through steam generator (IEOTSG) under conditions proposed for the Bellefonte Nuclear Station.[9] The tests showed the subcooled nucleate boiling length, total boiling length and steam superheat as a function of scaled power level. The tests were performed in the same manner as the OTSG tests described in 4.3 except that the isolation valve in the aspirator pipe was closed. Each test was performed with primary pressure, primary inlet conditions, feedwater conditions, and secondary pressure held constant. The nucleate boiling lengths were determined from primary side and secondary side tube thermocouples.

The boundary conditions for five, [d] MWt plant-scaled tests are shown in Table 4-4. Using these boundary conditions, a steady-state calculation of each test was performed using RELAP5/MOD2-B&W. The calculated boiling lengths are compared to the observed values in Table 4-5 and Figure 4-8.

The RELAP5/MOD2-B&W predictions are in good agreement with the test data. The differences between the observed and calculated values are due to the course noding of the steam generator model. Since the control volume boundaries do not coincide with the observed boiling lengths, it is not possible for the predicted values to coincide with the observed values.

4.6 IEOTSG LOFW Test

The ARC IEOTSG loss-of-feedwater test used for this benchmark, Run 13, was preformed on December 15, 1977.[8] The model OTSG was initialized to full scaled power consistent with a [d] MWt plant. The aspirator steam bleed valve was closed to simulate the integral economizer OTSG. The test was conducted in a similar manner to the OTSG tests by the simultaneously tripping the feedwater pump and closing the feedwater isolation valve. The generator was allowed to boil dry. After the recorded steam flow reduced to zero, the feedwater was turned on by starting the feedwater pump and by opening the feedwater isolation valve.

An attempt was made to hold the primary inlet temperature, primary flow and secondary pressure constant during the test. Primary outlet temperature, secondary steam flow, and secondary steam temperature were measured and recorded during the test.

The test procedure indicated that the downcomer isolation valve, located in the downcomer outlet, was closed to terminate feedwater. However, the data strongly suggest that the feedwater isolation valve was closed instead. This allowed feedwater to trickle into the tube region from the downcomer. The justification for this conclusion is based upon two anomalies in the data. First, there was a significant delay in the restart of steam flow after the feedwater was re-initiated, indicating that the downcomer was not solid with liquid when the feedwater was restarted. Secondly, conservative heat balances demonstrated that the inventory in the generator was not large enough to maintain the primary-to-secondary heat transfer that was observed in the test at the end of blowdown. The boundary conditions used in the RELAP5/MOD2-B&W benchmark of this test include an estimate of the average rate of liquid displacement from the downcomer.

The RELAP5/MOD2-B&W input model is the same as that used in the OTSG benchmark except that downcomer nodes were eliminated, and heat structures were added for the OTSG secondary shell metal. The model was initialized to the test initial conditions shown on Table 4-6. The predicted primary and secondary fluid temperatures are compared in Figure 4-9 with the measured values just prior to test initiation. The feedwater flow, primary inlet temperature, primary flow, and secondary pressure values that were measured during the test were input as boundary conditions. The calculated steam flow and primary outlet temperature are compared to the measured data in Figures 4-10 and 4-11, respectively.

RELAP5/MOD2-B&W overpredicts the steam flow from the IEOTSG during the dryout period. Consequently, the predicted dryout time is [c, d] than the observed dryout time of approximately[c, d]seconds. The breaks in the predicted steam flow, identical to those observed in the OTSG LOFW prediction, are caused as the mixture level crosses control volume boundaries. Following restart of feedwater, the predicted steam flow is in excellent agreement with the measured values. The overprediction of steam flow during the dryout period also means that the primary-to-secondary heat transfer is overpredicted. Additional modeling--including study of the changes in the subcooled boiling region--are required to obtain an accurate prediction of this test.

The predicted primary outlet temperature appears to be in good agreement with the measured value until approximately[ ]c,d seconds. However, the measured data reflects the effects of RTD thermal lag. If the code prediction were adjusted to account for RTD thermal lag, it would be observed that the code prediction is lower than the measured values during the majority of the dryout phase of the test. This is consistent with the steam flow (and heat transfer), which is overpredicted as compared with the data.

4.7 Conclusions

The benchmarks to ARC 19-tube steam generator data demonstrate that the RELAP5/MOD2-B&W computer code properly predicts the secondary side nucleate boiling length as a function of load. Accurate prediction of the heat transfer surface area in the steam generator is necessary for the accurate prediction of the B&W-designed plant response to upset conditions. Furthermore, the RELAP5/MOD2-B&W comparisons with ARC 19-tube OTSG loss of feedwater tests show that the code correctly predicts the primary-to-secondary heat transfer during steam generator dryout and refill. Consequently, RELAP5/MOD2-B&W is an acceptable tool for calculating the response of operating B&W-designed plants to secondary side upsets.

Table 4-1. Boundary Conditions for ARC 19-Tube OTSG Steady-State Tests

d

Table 4-2. Comparison of Predicted and Observed Boiling Lengths For Steady-State Model 19-Tube OTSG Tests

d

Table 4-3. Initial Conditions for the Model 19-Tube OTSG LOFW Test

[d]

Table 4-4. ARC 19-Tube IEOTSG Conditions for Steady-State Boiling Tests

[d]

Table 4-5 Comparison of Predicted and Observed Boiling Lengths For Steady-State Model 19-Tube IEOTSG Tests

[d]

Table 4-6. Initial Conditions for the 19-Tube IEOTSG LOFW Test

[d]

FIGURE 4-1. SCHEMATIC DIAGRAM FOR THE NUCLEAR STEAM GENERATOR TEST FACILITY.

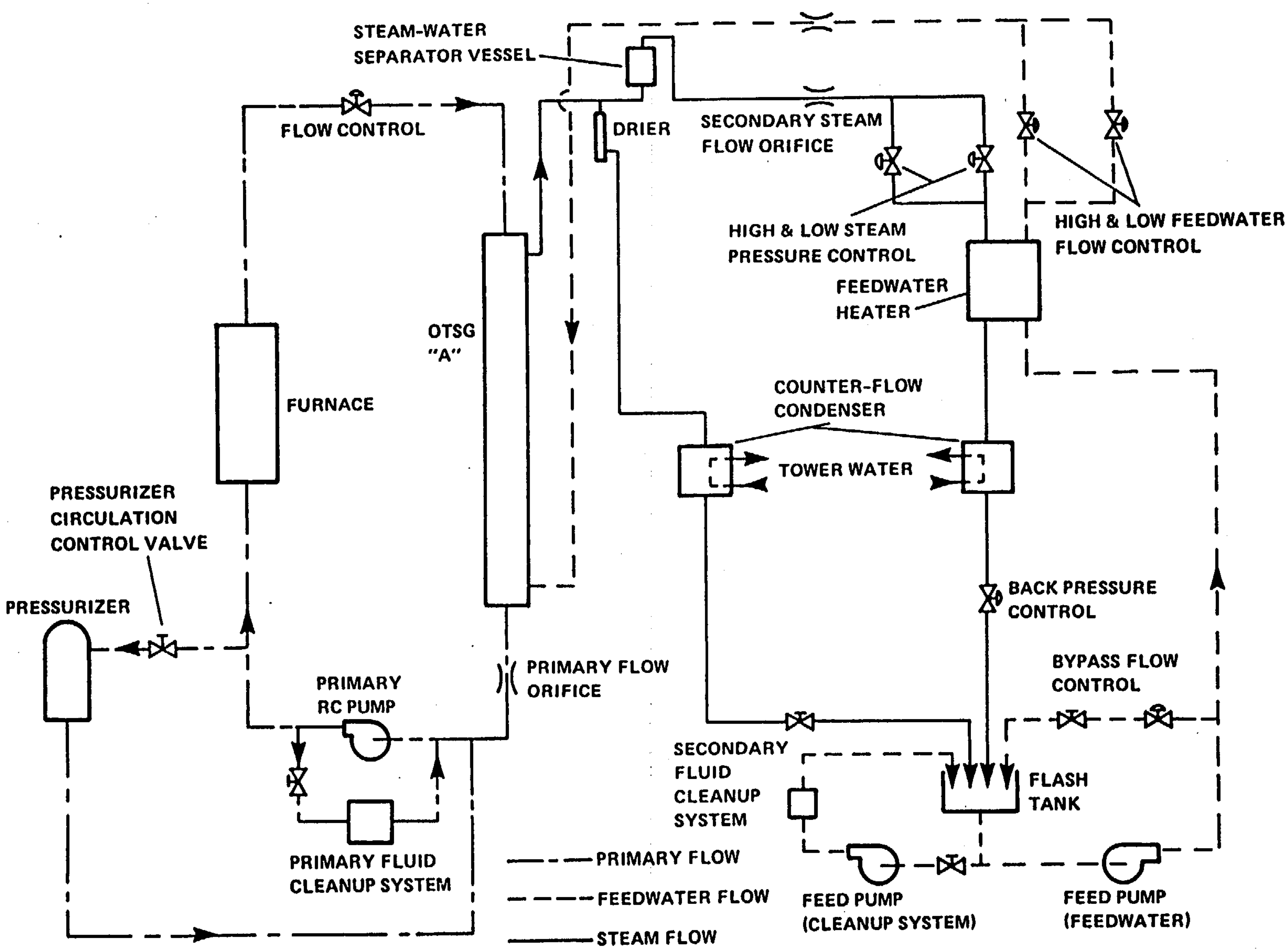


FIGURE 4-2. ARC 19-TUBE ONCE-THROUGH STEAM GENERATOR AND DOWNCOMER.

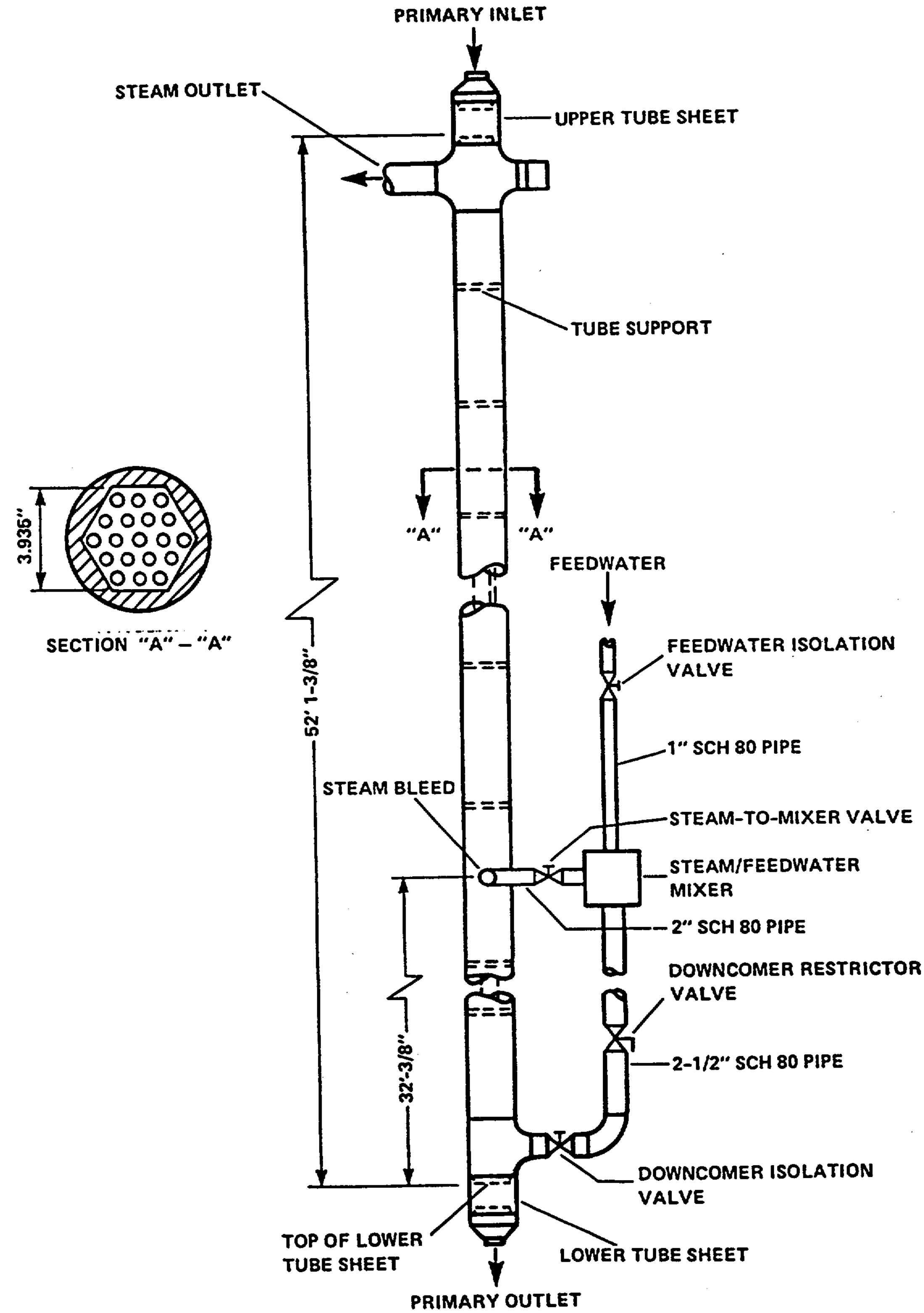


FIGURE 4-3. RELAP5/MOD2-B&W NODING FOR THE ARC 19-TUBE MODEL OTSG.

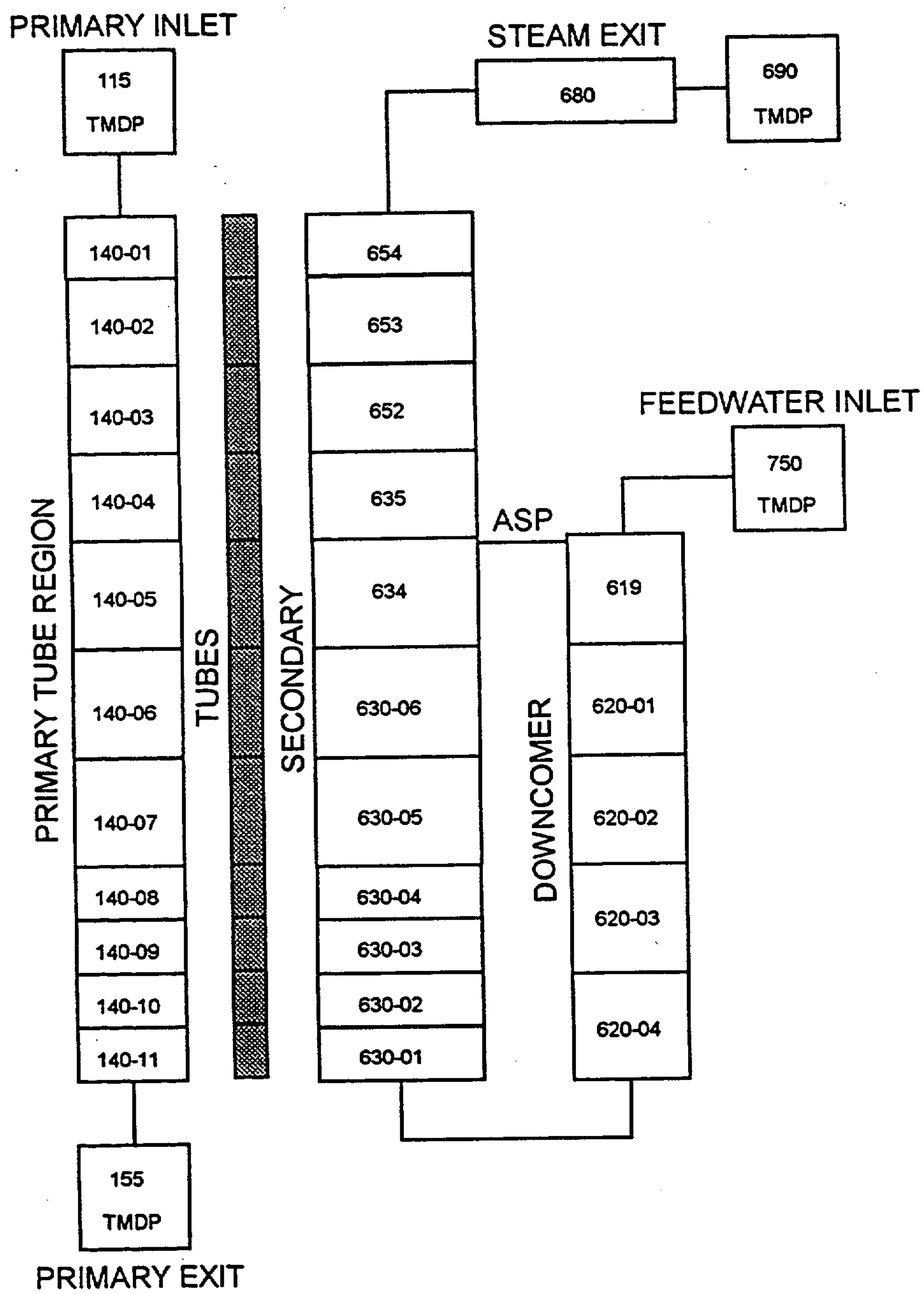


FIGURE 4-4. COMPARISON OF MEASURED VERSUS PREDICTED BOILING LENGTHS.

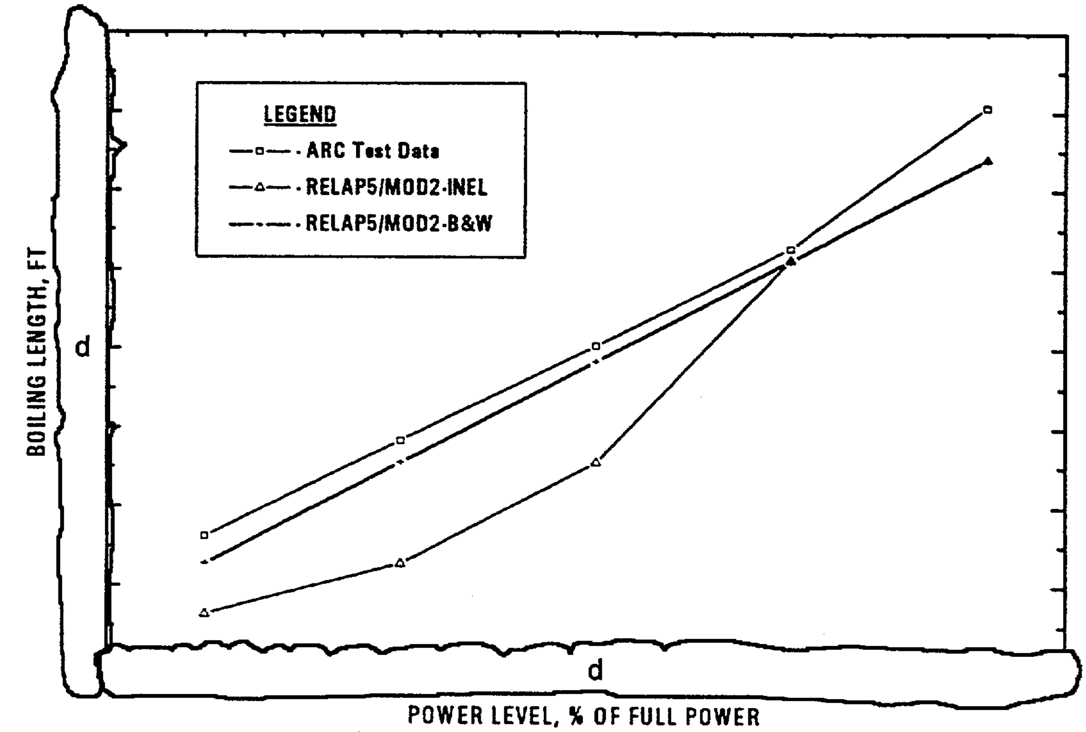


FIGURE 4-5. MEASURED AND PREDICTED AXIAL FLUID TEMPERATURES FOR THE 19-TUBE LOFW STEADY-STATE.

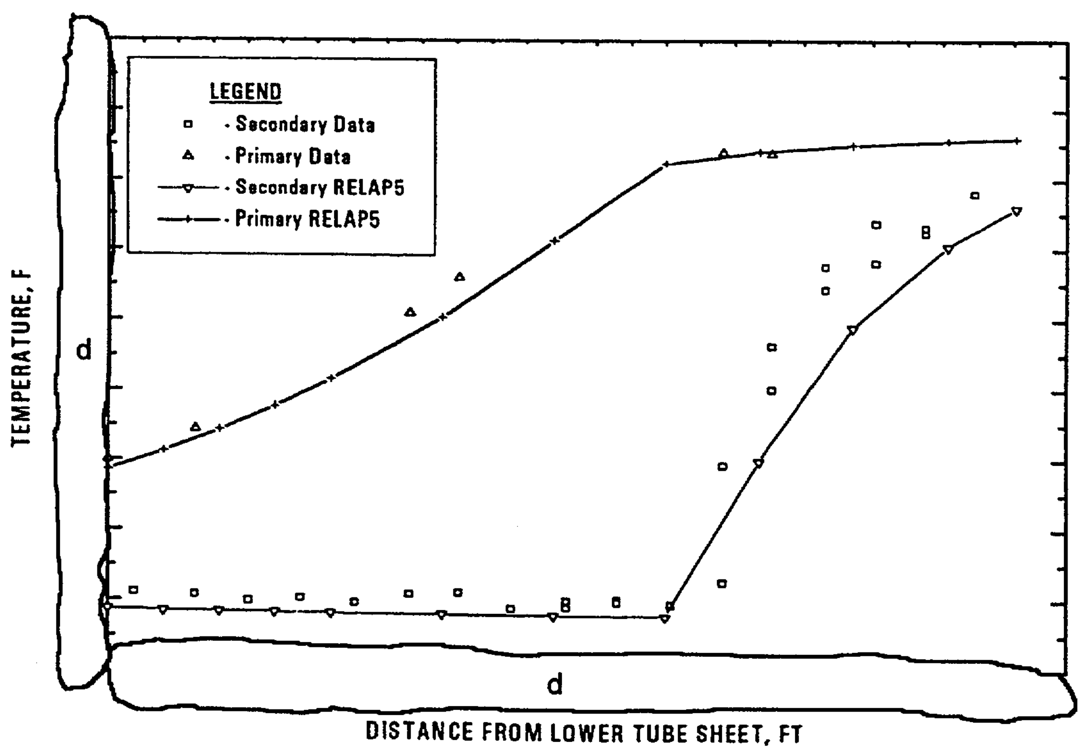


FIGURE 4-6. MEASURED VERSUS PREDICTED STEAM FLOW DURING THE 19-TUBE OTSG LOFW TEST.

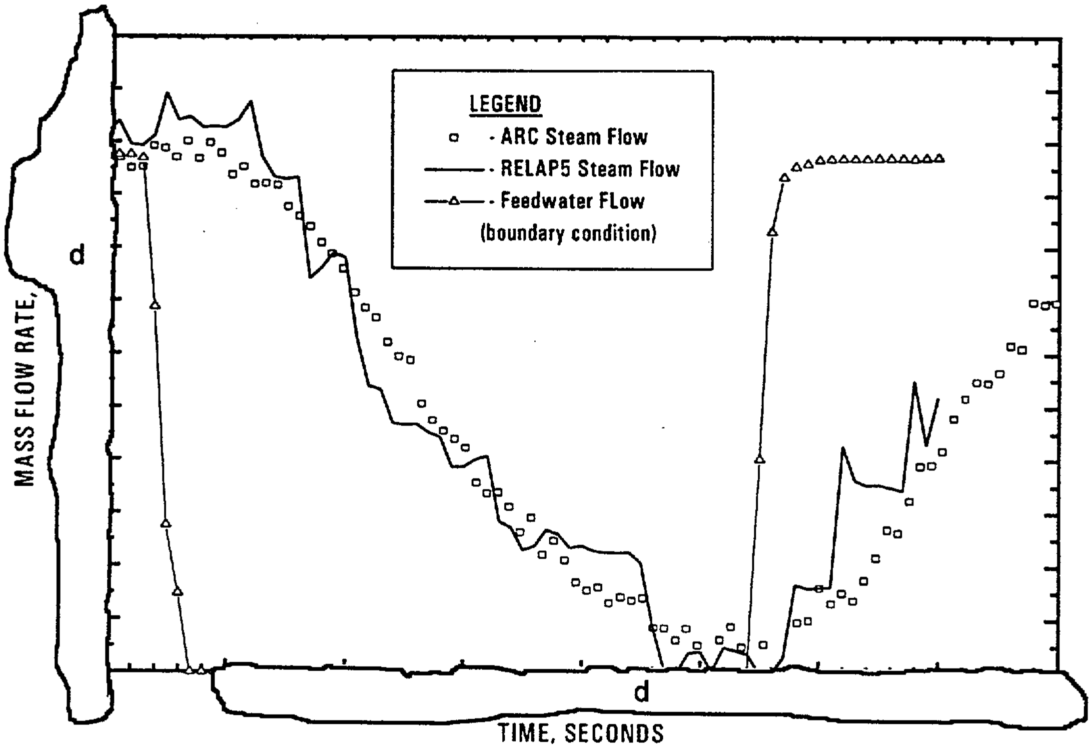


FIGURE 4-7. MEASURED AND PREDICTED PRIMARY OUTLET TEMPERATURES FOR THE 19-TUBE LOFW TRANSIENT.

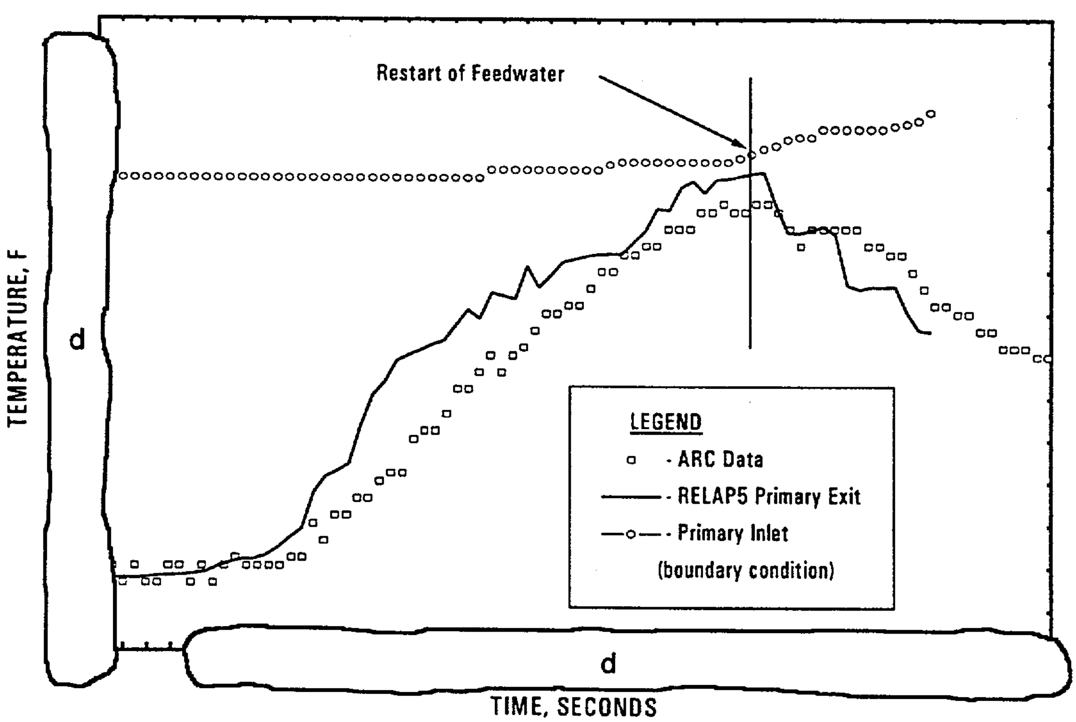


FIGURE 4-8. COMPARISON OF MEASURED VERSUS PREDICTED BOILING LENGTH FOR THE IEOTSG STEADY-STATE TESTS.

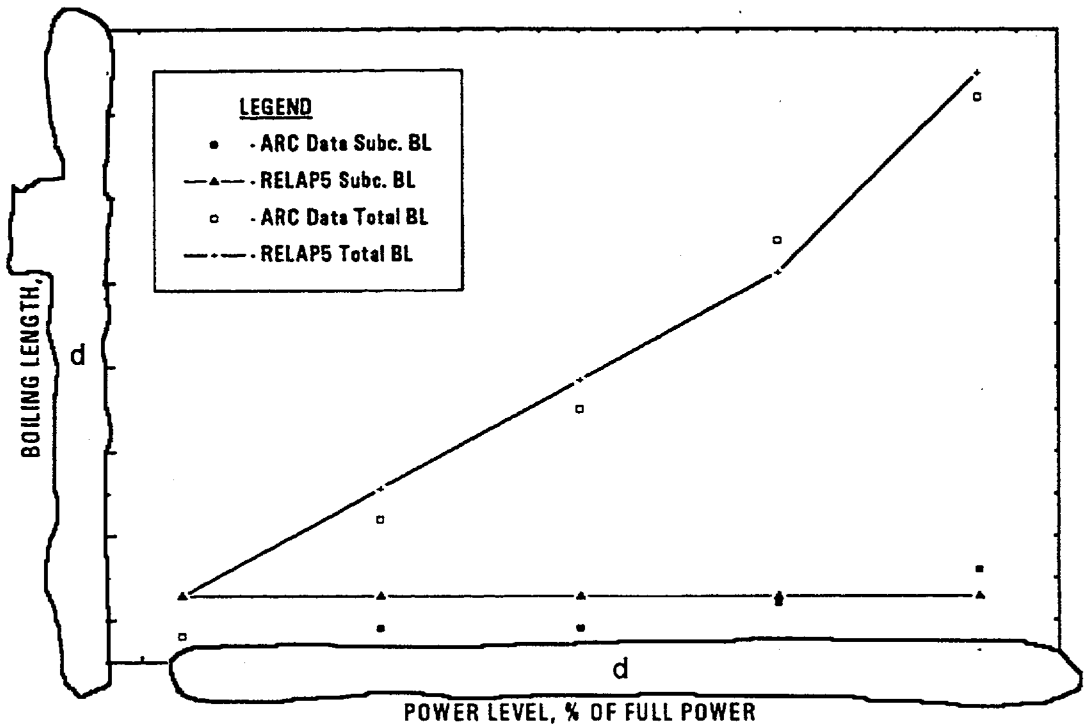


FIGURE 4-9. MEASURED AND PREDICTED AXIAL FLUID TEMPERATURES FOR THE 19-TUBE IEOTSG LOFW STEADY-STATE.

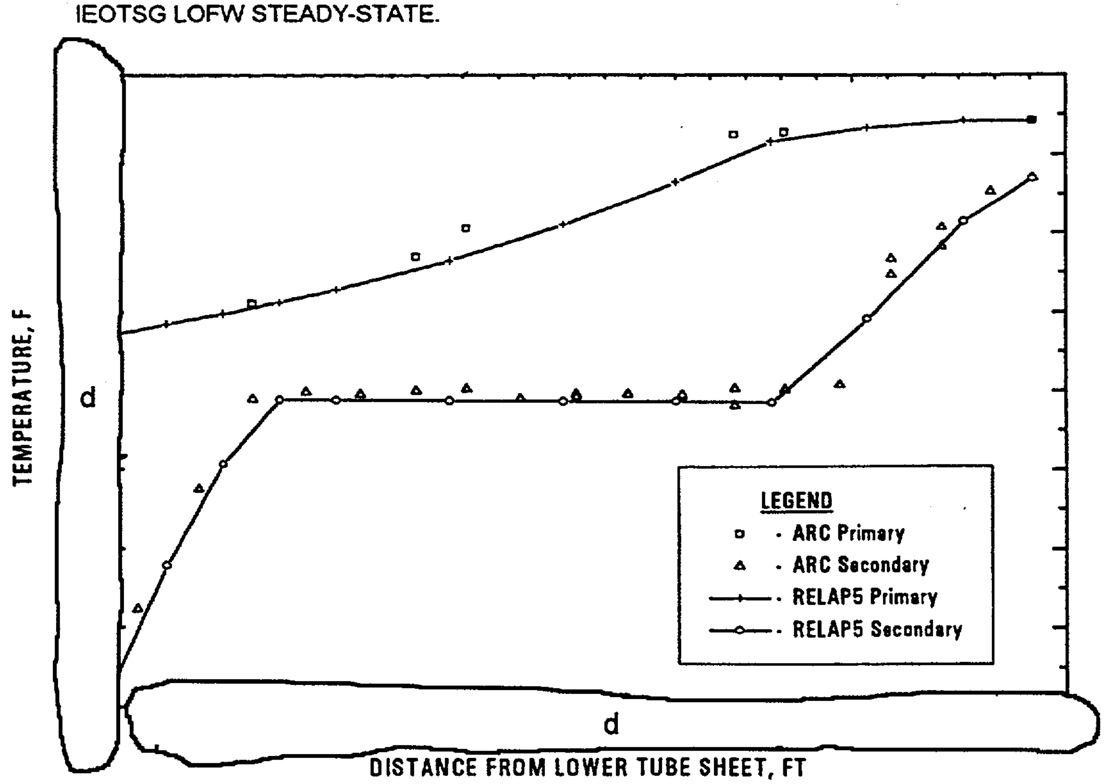


FIGURE 4-10. MEASURED AND PREDICTED STEAM FLOW FOR THE 19-TUBE IEOTSG LOFW TRANSIENT.

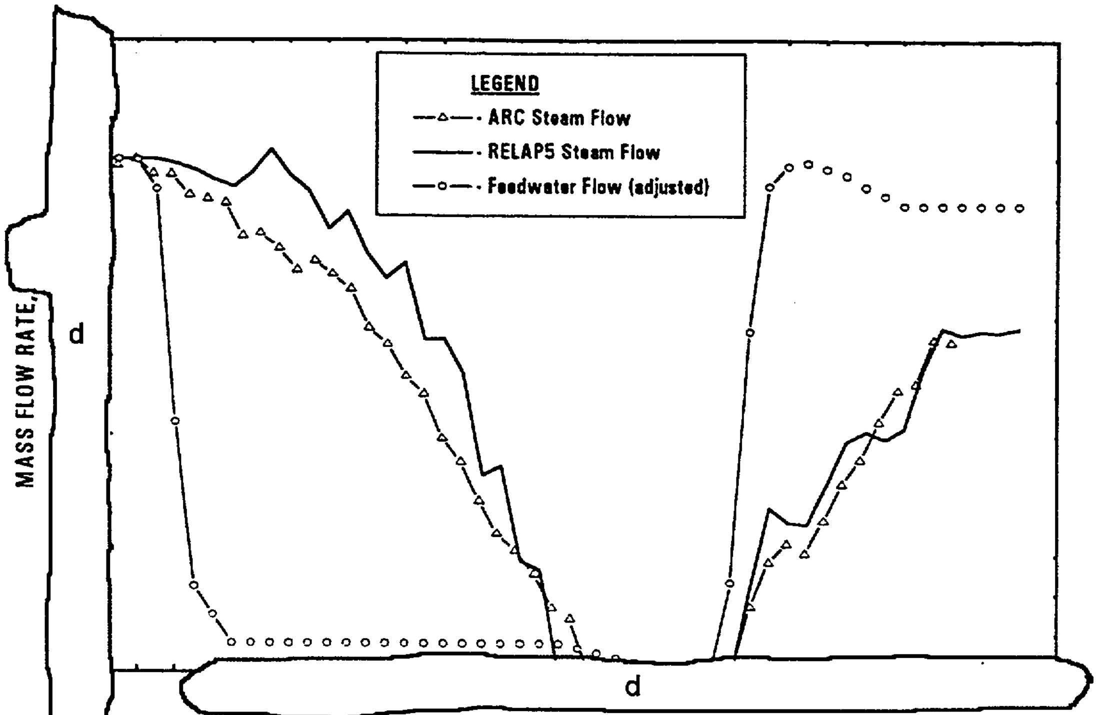
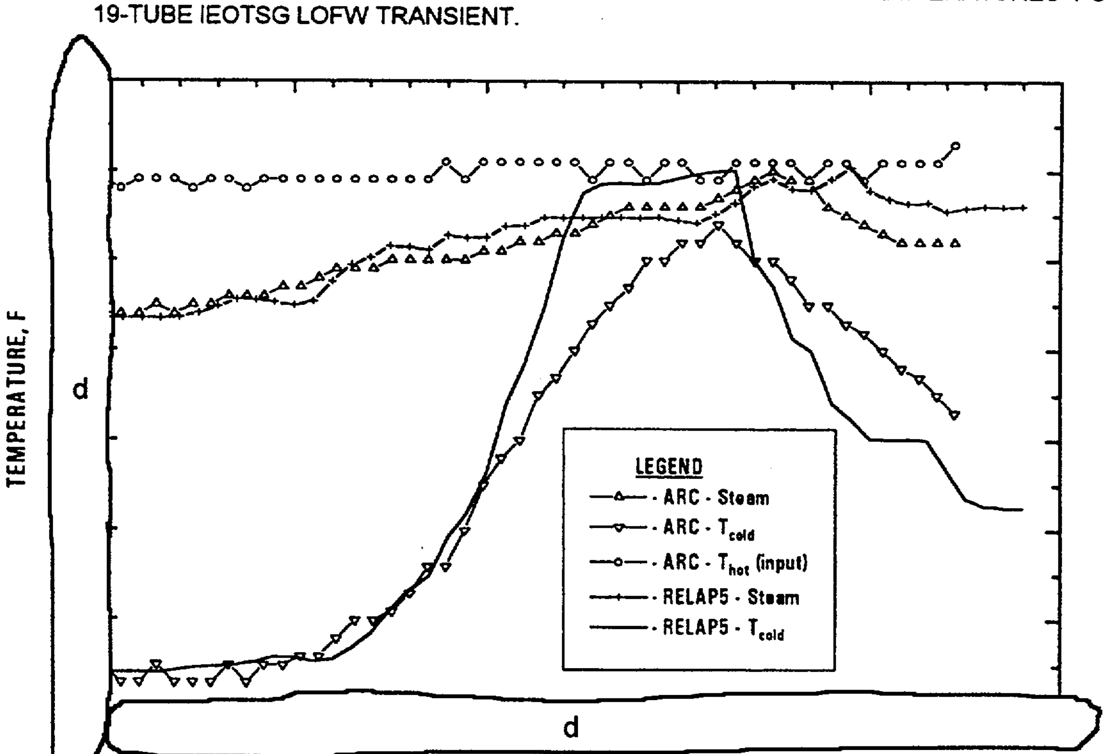


FIGURE 4-11. MEASURED AND PREDICTED PRIMARY AND STEAM OUTLET TEMPERATURES FOR THE 19-TUBE IEOTSG LOFW TRANSIENT.



5. BENCHMARKS TO PLANT DATA

Computer code benchmarks to plant transient data provide an effective way to determine the capacity of a computer code to predict plant behavior. Most important, the direct application of a code to simulate a plant event reduces concerns over the applicability of small-scale test facility results to a full-size plant. Four benchmarks were performed to provide evidence that the RELAP5/MOD2-B&W computer code can predict the response of B&W-designed pressurized water reactors:

1. Three Mile Island Unit 2 (TMI-2) loss-of-feedwater event of March 26, 1979.
2. Rancho-Seco loss-of-ICS power event of December 26, 1985.
3. Four-pump coastdown data from Oconee Unit 1 and Crystal River Unit 3.
4. Three Mile Island Unit 1 natural circulation test of October 7, 1985.

This section summarizes the results of the RELAP5/MOD2-B&W prediction of each plant transient or test. A discussion of each prediction is provided that includes a description of the plant event, a list of boundary conditions for the plant model, and an evaluation of the code prediction.

5.1 Methodology

A generic B&W lowered loop 177 fuel assembly plant model was used to perform all calculations. The base model (Figure 5-1) comprised:

a. Two hot legs.

b. Four cold legs with reactor coolant pumps.

c. A reactor vessel including core, downcomer annulus, and reactor vessel vent valves.

d. Two steam generators composed of one radial secondary and two radial primary regions (ten percent of the tubes in contact with auxiliary feedwater and ninety percent of the tubes unwetted by auxiliary feedwater).

e. A pressurizer including heaters, safety valves, power operated relief valve (PORV), and spray.

f. Boundary systems (e.g. high pressure injection, auxiliary feedwater, main feedwater).

g. Primary and secondary system metal heat structures.

Special features available in RELAP5/MOD2-B&W were employed in the once-through steam generator (OTSG) model. First, the[c]critical heat flux correlation was used on the shell side of the tube heat structures to provide a better prediction of the dryout point in the OTSG. Second, the interphase drag in the slug and annular-mist flow regimes was reduced by use of the default multipliers developed for regions of small hydraulic diameters. This model produces results similar to the Wilson bubble rise model for pressures above 200 psia and provides a better prediction of liquid mass in the tube region. Third, a linear ramp was applied to the Chen boiling suppression factor such that it was reduced from the calculated value to zero over a void fraction of [c, e] This prevented the Chen heat transfer coefficient from becoming unrealistically large as the void fraction approached 1.0 on the shell side of the OTSG. Finally, the B&W high auxiliary feedwater (AFW) model was used in conjunction with the two region steam generator model. This combination of models allows the heat transfer in the tube region wetted by AFW (ten percent of the total surface area) to be calculated separately from the heat transfer in the tubes that are unwetted by AFW.

In each benchmark, the model was initialized to the plant conditions that existed prior to the event or test. The event was then simulated by imposing the transient plant boundary conditions on the model. These boundary conditions were taken from the plant recall computer or were estimated using available data. When required, the core decay heat input for each benchmark was calculated from the plant power history using ANS 5.1 (1979) methodology. In each case, the predicted values of primary pressure, secondary pressure, primary system fluid temperatures and pressurizer level were compared with the plant values.

5.2. Benchmark of the TMI-2 LOFW Event

At approximately 4:00 a.m. on March 28, 1979 Three Mile Island Unit 2 experienced a loss-of-main feedwater while operating at 97% full power. The loss of both main feedwater pumps caused a coincident turbine trip. The subsequent increase in secondary pressure (increase in saturation temperature) combined with the loss-of-feedwater (LOFW) reduced the primary-to-secondary heat transfer in the steam generators. The resulting mismatch in core heat production and steam generator heat removal caused the primary system pressure to increase. The power operated relief valve (PORV) opened automatically to relieve pressure. However, the system continued to pressurize until the reactor tripped on high reactor coolant system pressure. The core heat generation dropped to the decay heat level and the primary system experienced a normal post-trip cooldown and contraction. Unlike during a normal reactor trip, the primary system continued to depressurize past the normal post-trip value because the PORV failed to close when the low pressure setpoint was reached.

Approximately 40 seconds into the event, the steam generator water level dropped to the low level setpoint and control valves opened automatically to provide auxiliary feedwater (AFW) to maintain minimum levels in the steam generators. The AFW block valves between the control valves and the steam generators were closed, however, which prevented auxiliary feedwater from being delivered. Consequently, the steam generators dried out, and the primary system began to reheat. AFW was restored to the steam generators about eight minutes after the initial LOFW. This benchmark was performed for the first two minutes of the event to focus on predicting the plant behavior during the LOFW period.

5.2.1 RELAP5/MOD2-B&W Boundary Conditions

The boundary conditions or components modeled in this benchmark include core reactivity coefficients, core decay heat, main feedwater, secondary safety valves, secondary atmospheric dump valves and the pressurizer PORV.

The core fission power response was calculated using the point kinetics model in RELAP5/MOD2-B&W. The kinetics parameters and control rod worth used in the simulation are shown in Table 5-1 and provide a power response typical of a rodded core near the beginning of the fuel cycle. The decay heat that was in put to the model (Table 5-2) was calculated using the actual TM 1-2 power history prior to the event.

The coastdown of main feedwater to each steam generator was also based on reactimeter data. Adjustments to the data at low flow rates were made to remove instrumentation noise. Specifically, an [d] of the main feedwater pump speed and flow was assumed, and a curve fit was made through the first 6 to 9 seconds of the flow data. The adjusted main feedwater coastdowns are listed in Table 5-3 and shown in Figures 5-2 and 5-3.

Following turbine trip, the secondary pressure in each steam generator increased until the turbine bypass valves and main steam safety valves (MSSVs) lifted. The MSSV characteristics shown in Table 5-4 provided a good prediction of the secondary pressure until the MSSVS reseated. After the MSSVs reseated, the secondary pressures oscillated between 970 and 1022 psig. This effect was simulated by modeling the modulating atmospheric dump valves on the steam generators.

Finally, to provide an adequate prediction of primary system pressure, it was necessary to model the pressurizer PORV. The PORV was modeled as a trip valve component with an area of 1.353 in^2, a discharge coefficient of 0.9 and an open setpoint of 2255 psig (nominal PORV lift setpoint at the time of the event).

5.2.2 RELAP5/MOD2-B&W Prediction

The plant model was initialized to the conditions existing at TMI-2 just prior to the event (Table 5-5). The transient was initiated by simulating the main feedwater reduction and coincident turbine trip. The subsequent sharp increase in secondary pressure (Figure 5-4) reduced the primary-to-secondary heat transfer. The reduction in heat transfer caused the cold leg temperature (Figures 5-5 and 5-6) and primary pressure (Figure 5-7) to increase. When the pressure reached the lift setpoint 3.3 sec after the LOFW, the PORV opened automatically. The pressure continued to increase until the reactor tripped at 8.8 seconds.

The PORV opening time and the reactor trip time are in good agreement with the plant data (Table 5-6).

Following reactor trip the primary system experienced a normal, post-trip cooldown and contraction. The predicted primary system pressure and temperatures are in good agreement with the plant data during this stage of the event. Furthermore, RELAP5/MOD2-B&W predicted the primary system liquid temperatures to stabilize near the values observed at the plant.

The temperatures remained stable until the steam generators dried out (Figures 5-8 and 5-9). Following steam generator dry-out, the primary system began to heat up. The RELAP5/MOD2-B&W predictions of the secondary liquid levels and steam generator dryout times were very good. However, the code predicted a greater primary system heatup rate than was observed at the plant. It is suspected that one of the auxiliary feedwater isolation valves at the plant was not completely closed. This would have allowed a small amount of AFW to enter one of the steam generators and remove some of the core decay heat.

The only significant deviation from the plant data occurs in the calculated pressurizer liquid level. The code prediction of pressurizer liquid level (Figure 5-10) agreed with the plant data until reactor trip. Following reactor trip, the plant data seem to indicate a much greater outsurge than predicted by RELAP5/MOD2-B&W. Experience with operating B&W plants shows that the pressurizer level should decrease by 5.5 inches for every one degree F decrease in average system temperature. Thus, the pressurizer level at 50 seconds should be 189 inches, not 159 inches as recorded. So, the plant data are probably not reliable during this period. The value predicted by RELAP5/MOD2-B&W at that time was 191 inches, which is in line with the rule-of-thumb. Given the apparent large uncertainty in the data, it is possible that none of the temperature-compensated pressurizer level data are reliable. However, the RELAP5/MOD2-B&W prediction of pressurizer level is considered to be appropriate because the predicted level is consistent with the changes in reactor coolant fluid densities and the predicted reactor coolant system pressure is in agreement with the plant data.

The benchmark of the TMI-2 LOFW event of March 28, 1979 is a good one. The primary and secondary system pressurization rates following the turbine trip were properly predicted by RELAP5/MOD2-B&W, providing a good prediction of PORV lift and reactor trip. The post-reactor trip primary system pressure, primary system temperatures, and steam generator liquid levels calculated by RELAP5/MOD2-B&W all agree with the plant data.

This benchmark shows that RELAP5/MOD2-B&W is appropriate for analyzing overheating events on B&W-designed pressurized water reactors.

5.3 Benchmark of the Rancho-Seco Loss of ICS Power Event

The Rancho-Seco plant had been operating at a steady-state power level of 76 percent when at approximately 4:14 a.m. on December 26, 1985, a loss of all integrated control system (ICS) DC power occurred. Main feedwater flow to the steam generators was immediately reduced as a result of the loss of ICS power. Total steam flow increased as the turbine bypass valves (TBVs) and atmospheric dump valves (ADVs) opened, and the positions of the turbine governor valves remained fixed. The additional heat removal provided by the increase in total steam flow could not compensate for the reduction in heat removal due to the loss of main feedwater flow. Consequently, primary system temperatures and pressure increased. Approximately 15 seconds after the loss of ICS power, the reactor tripped on high reactor coolant system (RCS) pressure. The turbine tripped shortly after reactor trip causing main steam safety valves (MSSVs) to lift.

Soon after reactor trip, the primary system underwent a normal, post-trip cooldown and contraction as core power generation fell to decay heat levels. The primary system continued to cool and depressurize, however, because full auxiliary feedwater (AFW) flow was started to the steam generators as they continued to depressurize from the open ADVs. RCS pressure decreased to the threshold for safety features actuation system (SFAS) actuation at 200 seconds. Upon the automatic actuation of SFAS, high pressure injection (HPI) flow was initiated. HPI was able to slow, but not counteract, the RCS contraction rate caused by the continued feeding and depressurization of the steam generators. Therefore, RCS pressure continued to decrease until the fluid in the reactor vessel upper head saturated and began to flash at approximately 400 seconds. At approximately 500 seconds into the event, RCS pressure stabilized at 1064 psia as the flashing of liquid in the reactor vessel upper head and HPI addition compensated for the contraction of the primary system.

Shortly afterwards, the HPI flow was more than sufficient to keep up with the RCS contraction rate. Therefore, RCS pressure started to increase. As RCS pressure increased, the flashing in the upper head ended, and the pressurizer started to fill. By 700 seconds, the secondary relief valves were closed, and AFW flow to steam generator-A was terminated. This ended the overcooling, causing the pressurizer level and primary system pressure to increase at a greater rate. At 976 seconds into the event, the isolation valve for AFW flow to steam generator-A was damaged. This resulted in a restoration of AFW flow to steam generator-A. At this time, the cooldown of the RCS resumed, and the RCS contraction rate increased such that the RCS pressure and the pressurizer level stabilized. At 1150 seconds, pressurizer spray flow was actuated to decrease RCS pressure.

At approximately 1550 seconds, ICS power was restored. Upon restoration of ICS power, all ICS demand signals were reduced to zero percent, terminating AFW flow. Consequently, overcooling of the RCS was terminated, and the RCS started to heat up.

5.3.1. RELAP5IMOD2-B&W Boundarv Conditions

The boundary conditions or components modeled in this benchmark included core power, main feedwater, auxiliary feedwater (AFW), primary system makeup flow, high pressure injection, main steam safety valves, turbine bypass system, and pressurizer spray. The core decay heat was estimated using the power history from the plant recall computer and ANS5.1 1979 methodology. The decay heat was combined with a typical post-trip fission power response. The resulting power table is given in Table 5-7.

Main feedwater flow (Table 5-8), auxiliary feedwater flow (Table 5-9), makeup flow and high pressure injection flow (Figure 5-11) were modeled directly from the data recorded by the plant recall computer. The recorded AFW flow to OTSG A went off-scale for a portion of the transient. The AFW flow during this time was estimated from system conditions and the AFW pump head/capacity curves.

The secondary pressure boundary condition was calculated assuming a fixed critical flow path of [c] from each OTSG to simulate the failed-open atmospheric dump valves and failed-open turbine bypass system. Because the exact position of the failed-open valves is not known, this value was determined by matching the pre-trip response of the secondary system pressure. The area was then held constant until operator action was modeled to close the valves manually. Following turbine trip, steam relief was provided by the main steam safety valves (MSSVs). The MSSV characteristics are shown in Table 5-10.

5.3.2 RELAP5/MOD2-B&W Prediction

The model was initialized to the plant conditions that existed prior to the event (Table 5-11). The event was initiated by starting the main feedwater (MFW) reduction and by opening the critical flow orifices that simulated the failed-open secondary relief valves. RELAP5/MOD2-B&W predicted the primary system temperatures (Figures 5-12 and 5-13) and pressure (Figure 5-14) to increase in direct response to the decrease in MFW flow. The increase in pressure resulted in a reactor trip on high primary system pressure at 15.2 seconds. This is in good agreement with the plant computer, which showed reactor trip at 15.0 seconds.

Upon reactor trip the turbine tripped, causing the secondary pressure (Figure 5-15) to increase to the main steam safety valve lift setpoint. Thereafter, the secondary pressure decreased as steam was relieved through the failed open secondary relief valves. Consequently, the decrease in secondary pressure, coupled with the overfilling of the steam generators (Figures 5-16 and 5-17), caused the primary system temperatures and pressure to decrease. Subsequently, the pressurizer emptied. The RELAP5/MOD2-B&W prediction of secondary pressure is excellent. Furthermore, the code predictions of secondary liquid levels are very good given the uncertainties in the AFW flow. Consequently, the predicted primary system temperatures, primary system pressure, and pressurizer level (Figure 5-18) match up well with the recorded values.

At about 500 seconds, the RCS pressure stabilized due to flashing of water in the reactor vessel upper head and due to addition of HPI fluid. The volumetric flow of the HPI exceeded the contraction rate of the primary coolant, so the primary system began to repressurize. The pressurization rate increased when the cooldown was terminated by closure of the secondary relief valves and by termination of AFW flow to OTSG A. The subsequent repressurization is underpredicted by RELAP5, most likely caused by uncertainty in measured HPI flows that were input to the model and to absence of reactor coolant pump seal injection flow in the model. This conclusion is supported by the delay in the predicted recovery of pressurizer level compared with the plant data.

At 976 seconds, AFW flow was re-established to OTSG A, and the cooldown of the primary system continued until all AFW flow was terminated at 1576 seconds. The predicted cooldown rate between 976 seconds and 1576 seconds matched the plant data. Following termination of AFW, RELAP5/MOD2-B&W predicted the primary system to heat up at the same rate recorded at the plant.

At 1150 seconds the operator started pressurizer spray to depressurize the primary system. Pressurizer spray flow was not recorded during the event. A good match to the observed depressurization rate was obtained by initiating spray flow of 50 gpm in the code benchmark. The flow rate was increased to 100 gpm at 1550 seconds and held constant thereafter. The benchmark analysis was terminated at 2250 seconds.

The RELAP5/MOD2-B&W prediction of the Rancho-Seco loss-of-ICS power event is very good. The primary-to-secondary heat transfer, following the reduction in main feedwater, was properly calculated as shown by the close prediction of reactor trip time. The calculated post-reactor trip secondary depressurization rate matches the plant data. Proper prediction of the depressurization rate depends upon accurate calculation of primary-to-secondary heat transfer in the tubes wetted by auxiliary feedwater, in addition to the heat transfer in the steam generator pool region. The proper calculation of heat transfer is demonstrated by the agreement between the plant data and the predicted values of primary system temperatures, steam generator liquid levels and secondary system pressure. The accurate prediction of primary system temperatures allowed primary system pressure and pressurizer level predictions that were close to the plant data, although there was some deviation caused by uncertainties in high pressure injection flow. This benchmark demonstrates that RELAP5/MOD2-B&W properly predicts the phenomena exhibited by the once-through steam generator and that it is a valid means for predicting secondary system-initiated events on B&W-designed pressurized water reactors.

5.4 Benchmark of Flow Coastdown Data

Certain tests are performed when any pressurized water reactor is started up for the first time. One such test is a coastdown of the reactor coolant pumps to verify the flow response. This comparison of reactor coolant pump speed following a four-pump coastdown demonstrates that the pump inertias, pump frictional torque values, and reactor coolant loop flow resistances input to the RELAP5/MOD2-B&W plant model yield an accurate calculation of the system flow rate.

Pump speed and reactor coolant flow were recorded for pump coastdown tests performed at Oconee Unit 1 and Crystal River Unit 3. Both units are of the B&W lowered-loop 177-fuel assembly design. The tests were performed from hot, full pressure, zero power conditions. The flow and/or pump speed were recorded during each test (Figure 5-19). The generic 177 fuel assembly model described in 5.1 was used for this comparison. A four-pump coastdown from full power conditions was simulated. The predicted pump response essentially overlays the plant data. Consequently, this comparison shows that the pump inertia, pump frictional torque values, and reactor coolant loop flow resistances input to the RELAP5/MOD2-B&W plant model yield an accurate calculation of the system flow rate.

5.5. TMI-1 Natural Circulation Test

Natural circulation heat removal is attained in a B&W-designed pressurized water reactor (PWR) by maintaining the heat sink elevation above the core center line elevation. This provides a driving head sufficient to circulate coolant around each primary system loop. The rate of the natural circulation flow depends upon (1) the vertical distance between the source and sink thermal centers, (2) the hot and cold leg fluid densities, and (3) the system pressure losses.

A test was performed on the Three Mile Island Unit 1 plant that demonstrated the natural circulation heat removal capability of the B&W-designed PWR. On October 7,1985, a low-power natural circulation test was conducted at the TMI-1 plant. The unit was operating at approximately three percent power, with full RCS flow and steam generator liquid levels controlled to the natural circulation setpoints. The test was initiated by tripping the reactor coolant pumps. The power was maintained at about three percent while the plant attained steady natural circulation. Auxiliary feedwater was used to remove the heat from the reactor coolant system. The primary system pressure increase was minimized by the intermittent use of pressurizer spray during the pump coastdown and by adjustment of letdown flow. After natural circulation was established, only the pressurizer heaters and letdown flow were used to regulate the primary system pressure. Primary system pressure, pressurizer level, primary system fluid temperatures and steam generator liquid levels were recorded during the event.

5.5.1 Boundary Conditions

Boundary conditions modeled in the RELAP5/MOD2-B&W simulation include the steam generator pressures, the reactor power, steam generator liquid levels, pressurizer heaters, and primary system letdown flow. The steam generator pressure boundary conditions were taken directly from the test data. Throughout the test, steam generator pressures were controlled to within 11 psi of the initial value. The reactor power input to the model (Table 5-13) was equal to 1.12 times the measured power. The correction factor was required because the reactor vessel downcomer fluid temperature during the test was colder than the temperature at which the out-of-core neutron detectors were calibrated. A control system was used to maintain the steam generator levels at 50 percent on the operate range (Figure 5-20) using auxiliary feedwater. Pressurizer heaters were used to control the primary system pressure to 1970 psia. The pressurizer heater setpoints were increased at 550 seconds to increase the primary system pressure to the normal value. The setpoints are shown in Table 5-14. Letdown mass flow rate data was not taken during the test. Consequently, the letdown flow was increased to the maximum value (140 gpm) upon initiation of the test, and was later throttled back to control primary system pressure (Table 5-15). The total letdown mass was limited to a value estimated from the recorded plant parameters.

5.5.2 RELAP5/MOD2-B&W Prediction

The plant model was initialized to the conditions that existed at the beginning of the test (Table 5-16). The benchmark was initiated by tripping all four reactor coolant pumps. The predicted plant response is shown in Figures 5-21 through 5-23. The test results for each reactor coolant loop were very similar, so the figures presented in this calculation show only the loop A data.

As the pumps coasted down, the predicted hot leg temperature increased to 580 F and subsequently decreased with the decline in core power until it stabilized at approximately 575 F. The RELAP5/MOD2-B&W prediction is in good agreement with the data although the rate of temperature increase is overpredicted. Similarly, the calculated rates of increase of primary system pressure and pressurizer level are greater than those observed during the test. Despite this deviation, the calculated pressure and pressurizer responses are very good.

Furthermore, the equilibrium primary system fluid temperature difference was calculated to be 34 F as compared with 35 F recorded during the test. Therefore, the calculated primary side natural circulation flow is within three percent of the test result. It is expected that RELAP5/MOD2-B&W will slightly overpredict the natural circulation by some small amount using this plant model. For example, the core is modeled with three axial control volumes. This means the core thermal center in the model is [c] the mid-core elevation. Consequently, the steam generator-to-core thermal center difference in the model is greater than in the plant, yielding a primary system natural circulation flow slightly greater than occurs in the plant.

The significance of this observation is that RELAP5/MOD2-B&W properly predicts the steam generator thermal center during natural circulation. An accurate calculation of the thermal center requires accurate calculation of the heat transfer in the tube region wetted by auxiliary feedwater as well as accurate calculation of the heat transfer to the secondary pool.

The code predictions of primary system fluid temperatures, primary system pressure, and pressurizer liquid level during the TMI-1 natural circulation test of October 7,1985 are very good. The equilibrium primary side temperature difference calculated by RELAP5/MOD2-B&W is in good agreement with the plant data, indicating that the code properly predicts the primary system flow and steam generator thermal center during natural circulation heat removal. Consequently, this benchmark demonstrates that RELAP5/MOD2-B&W is a good tool for predicting the response of B&W-designed PWRs during natural circulation events.

5.6 Conclusions

Four plant transients were benchmarked using the RELAP5/MOD2-B&W computer code. These transients exhibit the phenomena encountered by a B&W-designed PWR during loss-of-feedwater, turbine trip, severe overcooling, reactor coolant pump trips, and primary system natural circulation cooling. In each case, the plant response predicted by RELAP5/MOD2-B&W was in good agreement with the plant data. Consequently, RELAP5/MOD2-B&W is a good tool for predicting the response of B&W-designed PWRs during non-LOCA events.

Table 5-1. Reactor Kinetics Parameters Used in the TMI-2 LOFW Benchmark

Parameter	Value
c	

Table 5-2. Calculated Core Decay Heat for the TMI-2 LOFW Event

Time After Trip (Seconds)	Decay Heat (MW)	Time After Trip (Seconds)	Decay Heat (MW)
0	178.4	20	114.3
1	168.4	25	110.8
2	159.1	30	107.4
3	152.8	35	104.1
4	147.0	40	100.7
5	142.7	50	96.8
7	135.8	60	92.9
10	128.4	70	90.2
13	124.0	90	85.5
15	121.2	110	82.4
17	118.4	130	80.1

Table 5-3. Main Feedwater Flow During the TMI-2 LOFW Event

Time (sec)	Loop A (lbm/s)	Loop B (lbm/s)
0	1609	1609
1	1391	1492
4	430	872
7	197	430
10	[	]
13		
16		
19	c	
22		
25		
30		
35		

Table 5-4. Main Steam Safety Valve Characteristics Used in the TMI-2 LOFW Benchmark

Number Per SG	Lift Pressure (psig)	Reseat Pressure (psig)	Capacity (lb/s/valve)
2	1050	985	207.3
2	1065	995	207.3
1	1075	1015	207.3
1	1075	1040	207.3

Table 5-5. Plant Conditions Prior to the TMI-2 LOFW Event

Parameter	Plant	RELAP5/MOD2-B&W
Power, percent of 2772 Mwt	97	97
Primary Pressure, psia	2163	2163
Hot Leg Temperature, F	606.0*	606.0
Cold Leg Temperature, F	557.9*	555.5
Pressurizer Level, inches	221	220
Secondary Pressure, psia	914.5*	925.0
Feedwater Flow, lbm/s	1588*	1609
Operate Range Level, percent	56.5	45.8

*Average of both loops.

Table 5-6. Sequence of Events for the TMI-2 LOFW Event

Event	Plant	RELAP5/MOD2-B&W
Both main feedwater pumps trip.	0	0*
Turbine trips.	~1	1*
PORV opens.	~4	3.3
Reactor trips on high RCS pressure of 2340 psia.	~9	8.8
Pressurizer heaters actuate.	14	15.6
Pressurizer level falls below 200 inches.	24	26.1
OTSG A startup range level off-scale low.	74	75
OTSG B startup range level off-scale low.	77	74

*Specified

Table 5-7. Post-Trip Core Power for the Rancho-Seco Loss-of-ICS Power Event

Time After Scram (s)*	Power (MW)	Time After Scram (s)*	Power (MW)
0.0	2106.72	67.2	38.25
1.2	331.73	77.2	35.55
3.2	140.03	87.2	33.33
5.2	119.41	97.2	31.49
7.2	105.84	107.2	30.18
9.2	95.78	117.2	29.21
11.2	87.93	200.0	26.28
13.2	81.84	300.0	24.38
15.2	76.97	400.0	23.15
17.2	72.91	600.0	21.32
22.2	65.37	800.0	19.97
27.2	60.15	1000.0	18.88
37.2	52.14	2000.0	15.25
47.2	45.94	5000.0	10.08
57.2	41.56		

*Time after control rods begin to insert.

Table 5-8. Main Feedwater Flow During the Rancho-Seco Loss-of-ICS Power Event

Time (sec)	Loop A (lbm/s)	Time (sec)	Loop B (lbm/s)
0	1202.8	0	1202.8
4	652.8	4	652.8
9	0.0	9	0.0
420	0.0	376	0.0
436	75.0	390	36.1
466	125.0	420	97.2
480	133.3	450	136.1
496	152.8	466	144.4
510	163.9	480	166.7
526	175.0	526	197.2
540	0.0	540	225.0
		556	233.3
		570	0.0

Table 5-9. AFW Flow During Rancho-Seco Loss-of-ICS Power Event

Time (sec)	Loop A (gpm)	Time (sec)	Loop B (gpm)
0	0	0	0
23	0	23	0
24	536	24	362
50	625	50	361
70	587	60	425
80	700	80	651
90	943	100	985
100	1053	120	1016
120	1091	150	1060
150	1142	180	1096
180	1173	190	1103
190	1194	192	618
192	662	200	670
210	689	210	648
220	903	220	436
224	1300	223	546
230	1224	225	1026
240	1231	230	1139
300	1289	240	1151
316	1298	300	1202
390	**1365**	360	1251
420	**1405**	466	1216
466	**1402**	496	697
480	**1665**	570	693
570	**1932**	600	536
600	**2090**	660	753
660	**1874**	720	975
690	478	780	1007
720	211	946	1012
750	82	960	684
780	0	1020	666
960	0	1140	650
976	**1744**	1200	650
1152	**1750**	1216	380
1212	**1960**	1246	102
1332	**1989**	1486	94
1392	**1930**	1500	0
1452	**1950**		
1512	**2035**		
1546	**2035**		
1560	1026		
1576	0		

Values in bold type-face are estimated based on system conditions and pump curves.

Table 5-10. Main Steam Safety Valve Characteristics Used to Benchmark the Rancho-Seco Loss-of-ICS Power Event

SG Pressure (psia)	Safety Valve Flow (lbm/s/SG)
1064.9	0
1065.0	470
1084.9	470
1085.0	940
1104.9	940
1105.0	1426
1117.4	1426
1117.5	1896

Table 5-11. Plant Conditions Prior to the Rancho-Seco Loss-of-ICS Power Event

Parameter	Plant	RELAP5/MOD2-B&W
Power, percent of 2772 Mwt	76	76
Primary Pressure, psia	2185	2186
Hot Leg Temperature, F	604.4*	603.2
Cold Leg Temperature, F	566.8*	565.5
Pressurizer Level, in std H_2O	141	140
Secondary Pressure, psia	898*	900
Feedwater Flow, lbm/s	1186*	1203
Operate Range Level, percent	42*	39

*Average of both loops.

Table 5-12. Sequence of Events for the Rancho-Seco Loss-of-ICS Power Event

Event	Plant	RELAP5/MOD2-B&W
Loss of ICS power.	0	0*
Main feedwater goes to zero.	9	9*
Reactor trip on high RCS pressure.	15	15.2
Turbine trip.	15	15.2
Auxiliary feedwater flow initiated.	24	24*
HPI flow initiated through the "D" nozzle.	38	38*
HPI flow initiated through the "A" nozzle.	48	48*
Safety Feature Actuation System pressure setpoint reached (1627 psia).	200	222
HPI flow initiated through all nozzles.	201	201*
Minimum RCS pressure reached. (Value in psia)	466 (1064)	490 (1077)
Reactor coolant pump 2B tripped.	853	853*
AFW flow to loop-B terminated.	1500	1500*
AFW flow to loop-A terminated.	1560	1560*

*Specified

Table 5-13. Corrected Core Power During the TMI-1 Natural Circulation Test

Time (sec)	Core Power (MW)
0	95.5
7	96.0
21	95.5
71	87.3
125	86.8
146	90.1
167	90.9
208	89.1
232	89.1
244	89.3
409	88.1
450	87.3
668	76.5
765	69.6
836	69.1
915	65.7
1104	62.1
1413	60.3
2000	60.3

Table 5-14. Pressurizer Heater Setpoints After 550 Seconds in the TMI-1 Natural Circulation Test

Bank	Plant ON Setpoint (psia)	RELAP5 ON Setpoint (psia)
1	2150	2150
2	2150	2150
3	2150	2135
4	2135	2120

Table 5-15. Primary System Letdown Flow Used to Benchmark the TMI-1 Natural Circulation Test

Time (sec)	Letdown Flow (gpm)
0	0
80	0
85	140
370	140
375	65
550	0

Table 5-16. Initial Plant Conditions for the TMI-1 Natural Circulation Test

Parameter	Plant	RELAP5
Reactor Power, MW	95.5**	95.5
Primary System Pressure, psia	1970	1970
Pressurizer Level, inches	148	152
Reactor Coolant Flow Per Loop, Mlbm/hr	73.5*	73.6
Hot Leg Temperature, F	543.1*	544.0
Cold Leg Temperature, F	541.4*	542.1
Steam Generator Pressure, psia	962.5*	959
Steam Generator Liquid Level, % operate range	49.5*	50

*Average of both loops.
**Corrected for shielding of out-of-core detectors.

FIGURE 5-1. RELAP5/MOD2-B&W CONTROL VOLUME DIAGRAM OF A B&W-DESIGNED 177-FA LOWERED-LOOP PLANT

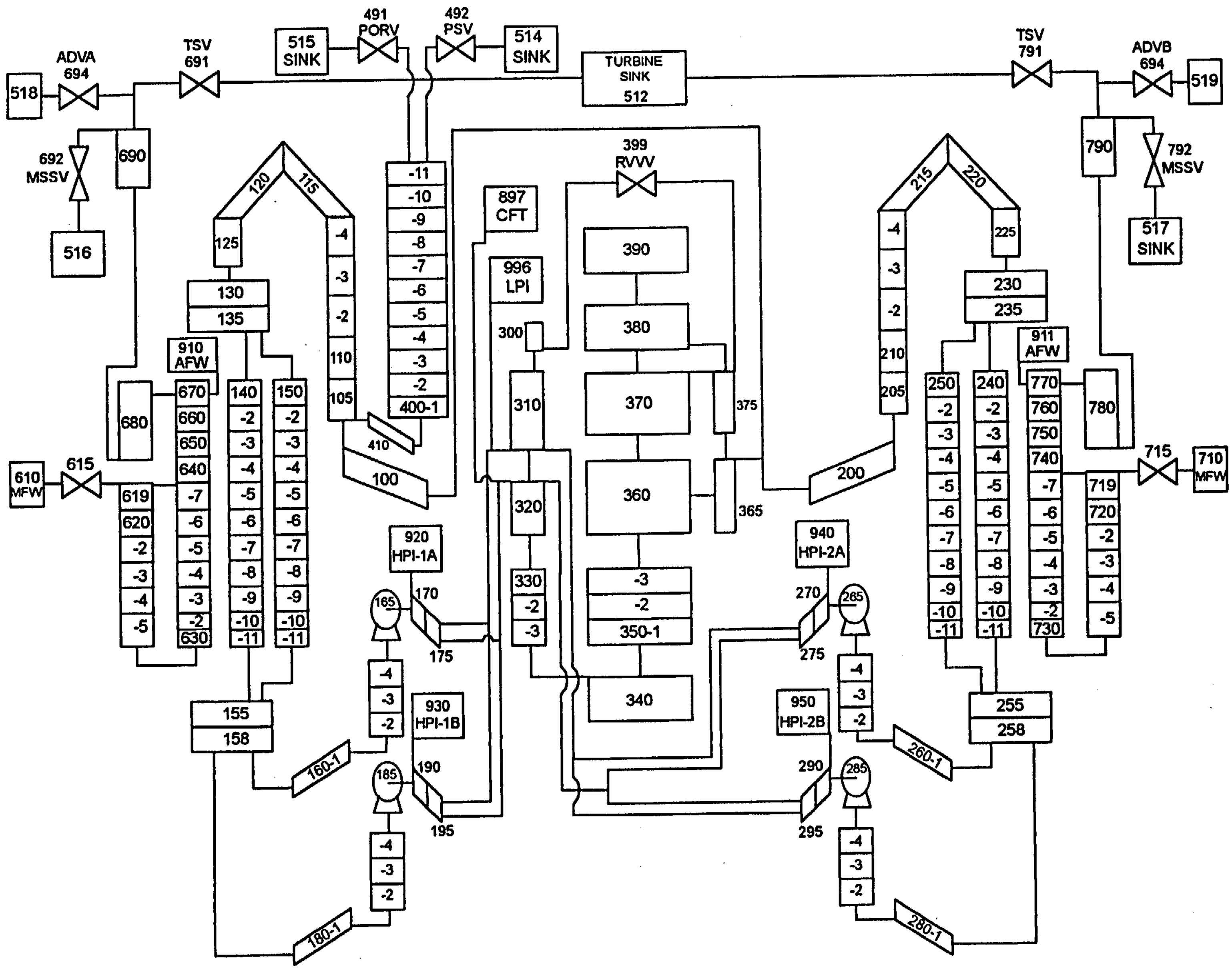


FIGURE 5-2. MAIN FEEDWATER FLOW COASTDOWN FOR THE TMI-2 LOFW EVENT.

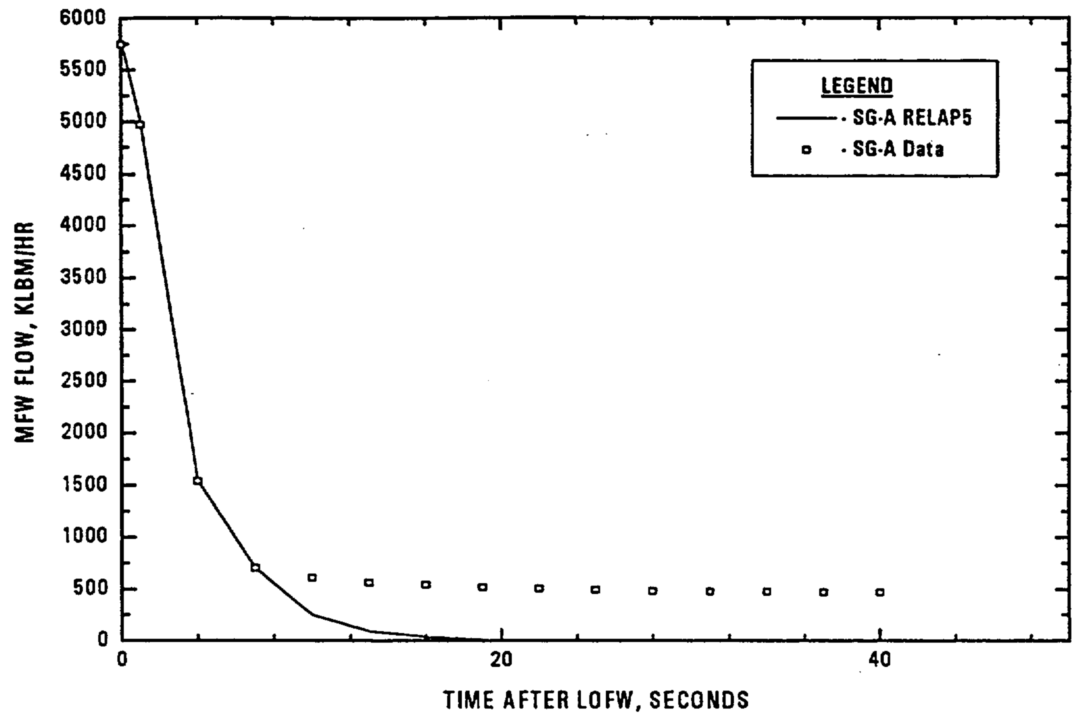


FIGURE 5-3. MAIN FEEDWATER FLOW COASTDOWN FOR THE TMI-2 LOFW EVENT.

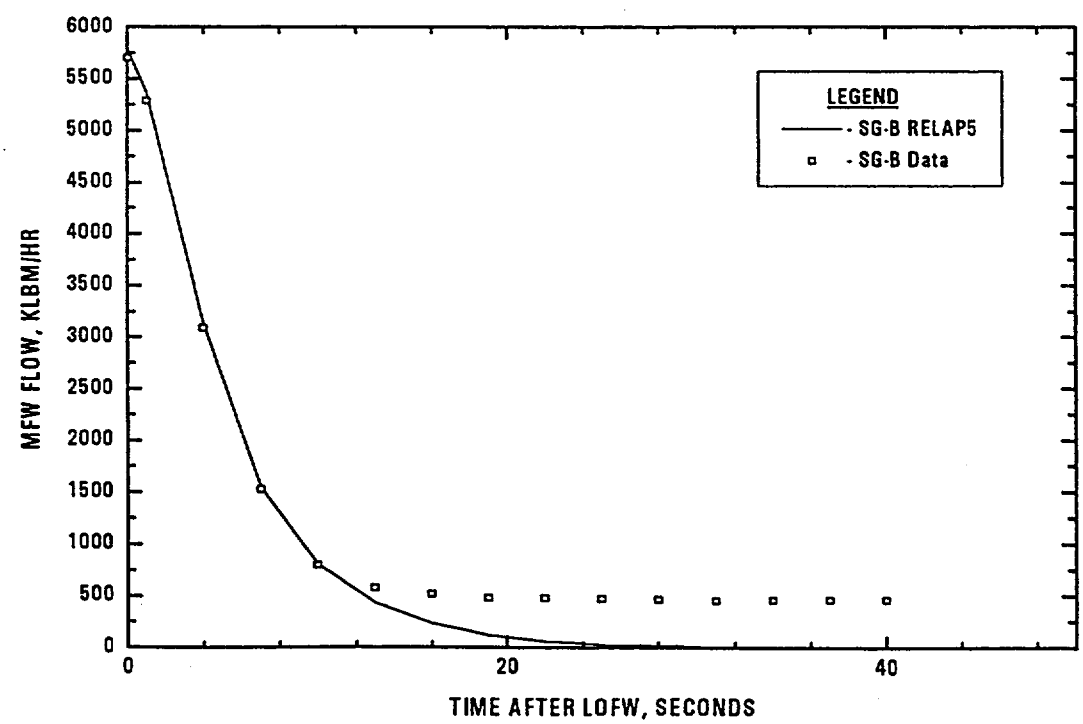


FIGURE 5-4. STEAM GENERATOR PRESSURES FOR THE TMI-2 LOFW EVENT.

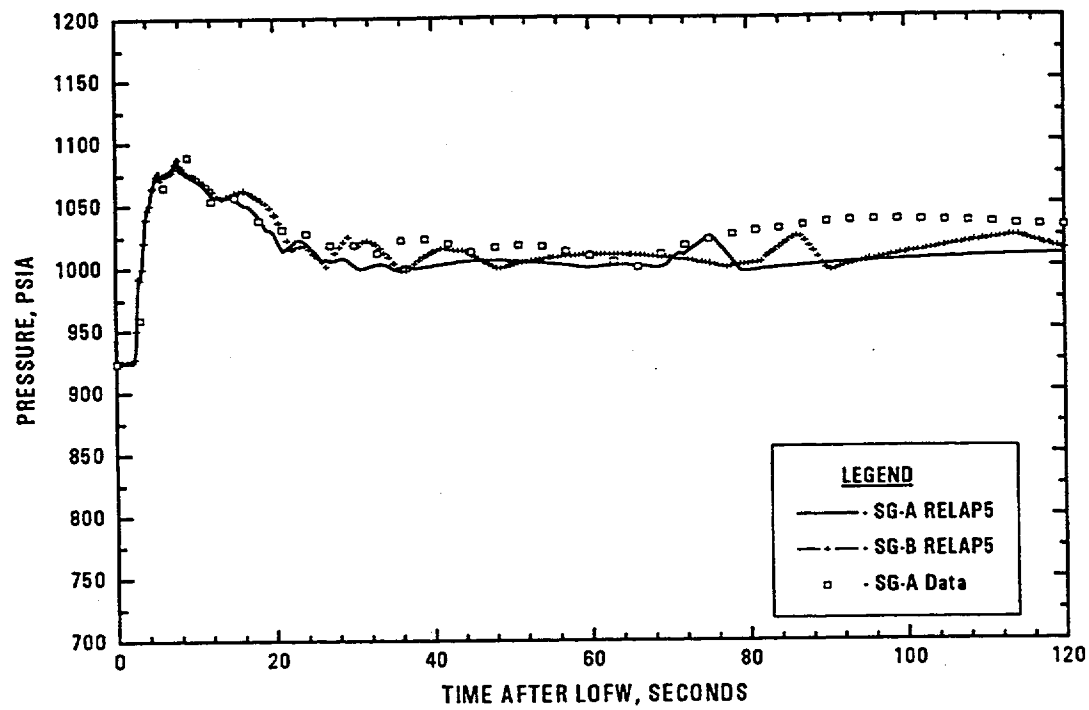


FIGURE 5-5. LOOP-A PRIMARY TEMPERATURES FOR THE TMI-2 LOFW EVENT.

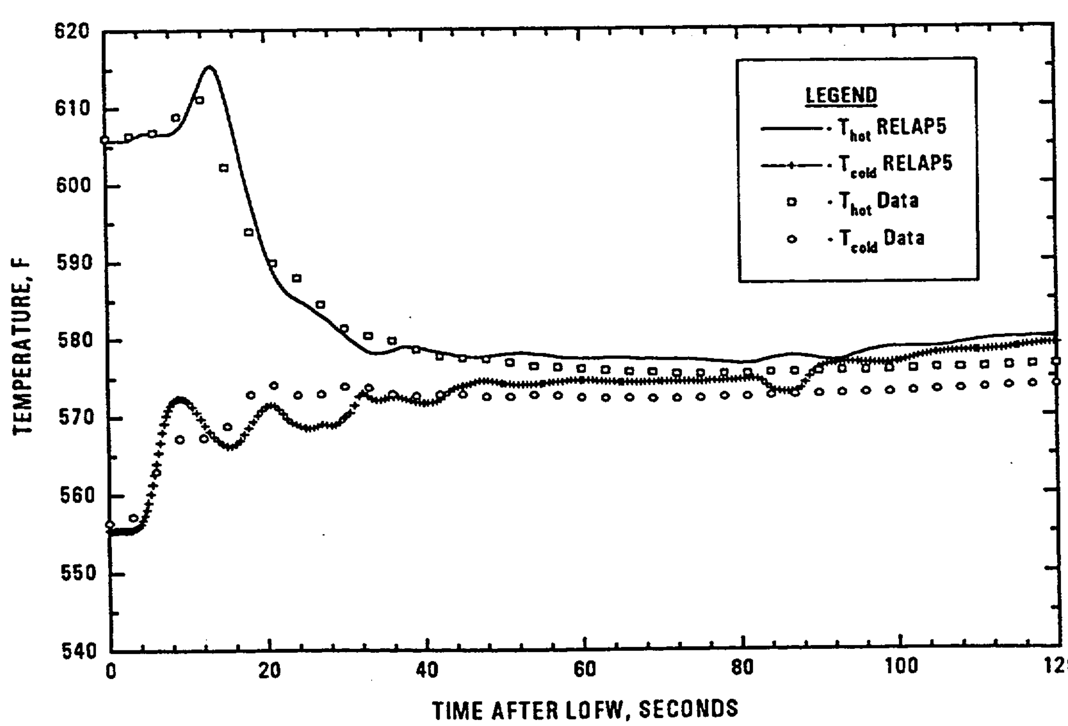


FIGURE 5-6. LOOP B PRIMARY TEMPERATURES FOR THE TMI-2 LOFW EVENT.

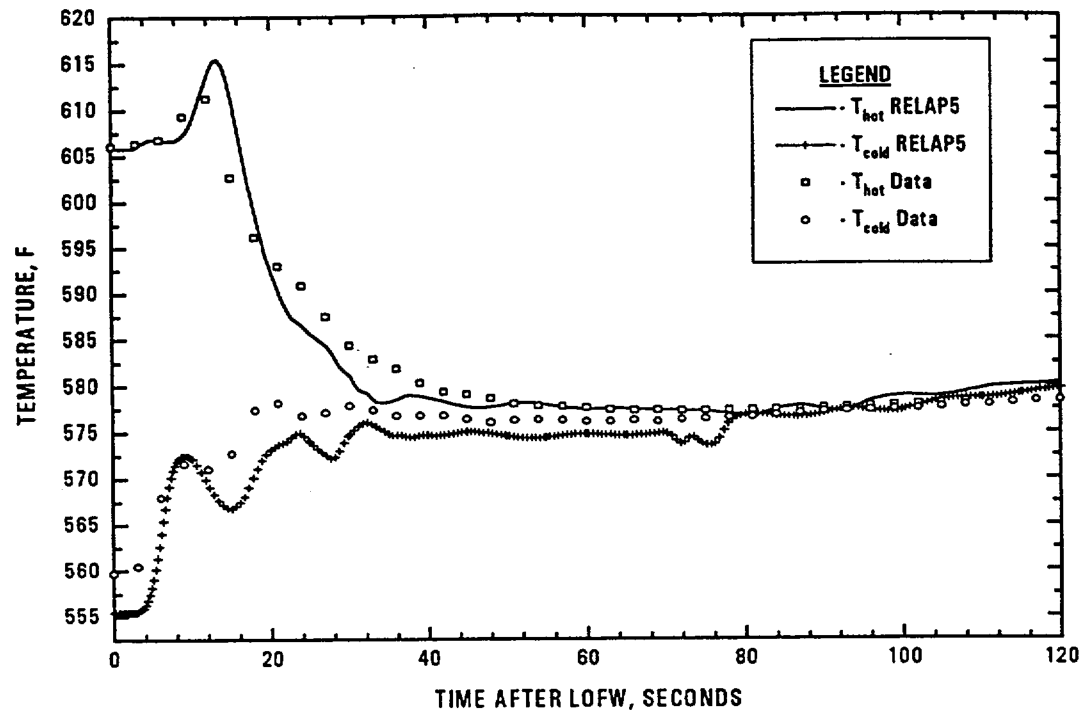


FIGURE 5-7. PRIMARY SYSTEM PRESSURE FOR THE TMI-2 LOFW EVENT.

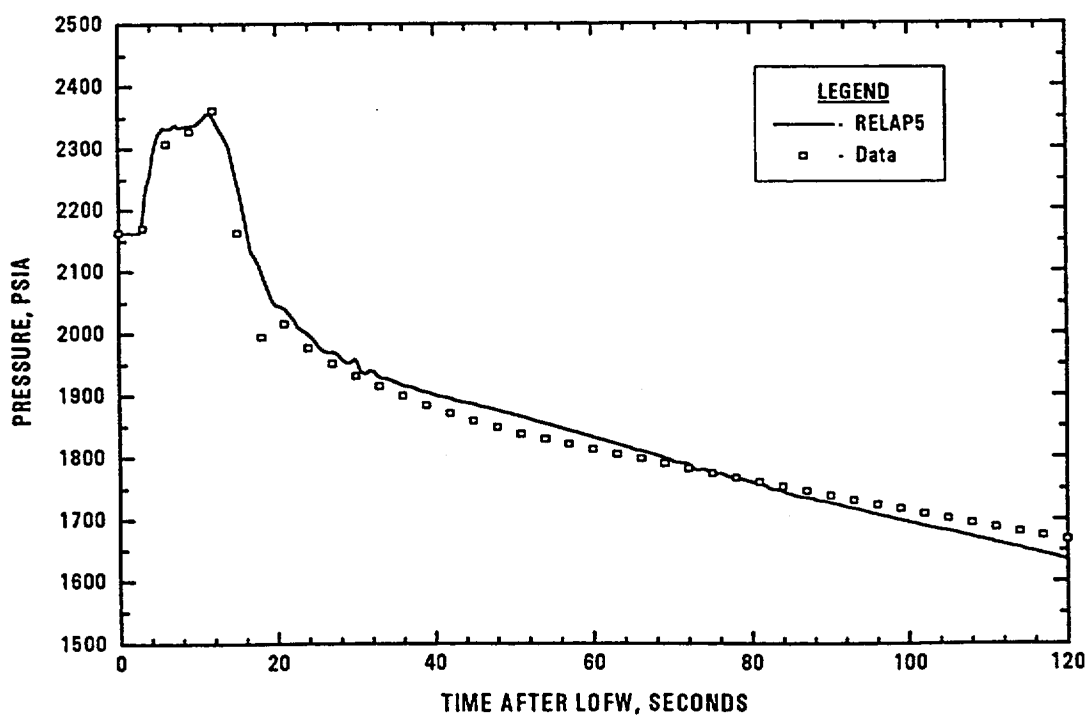


FIGURE 5-8. STEAM GENERATOR-A START-UP RANGE LEVEL FOR THE TMI-2 LOFW EVENT.

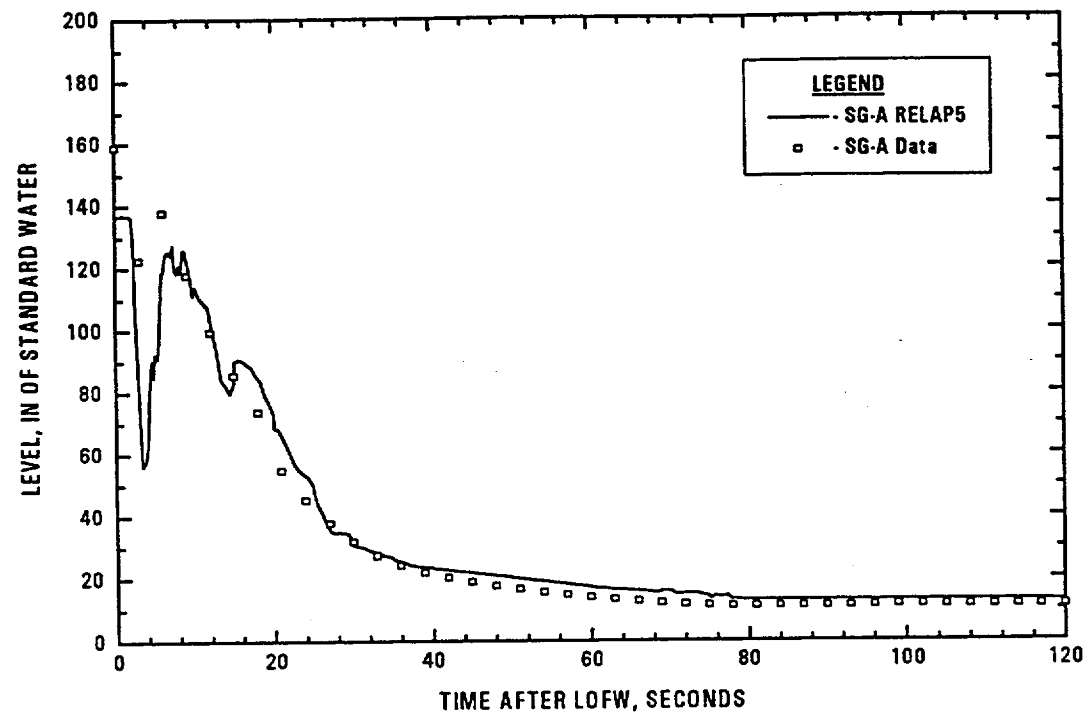


FIGURE 5-9. STEAM GENERATOR-B START-UP RANGE LEVEL FOR THE TMI-2 LOFW EVENT.

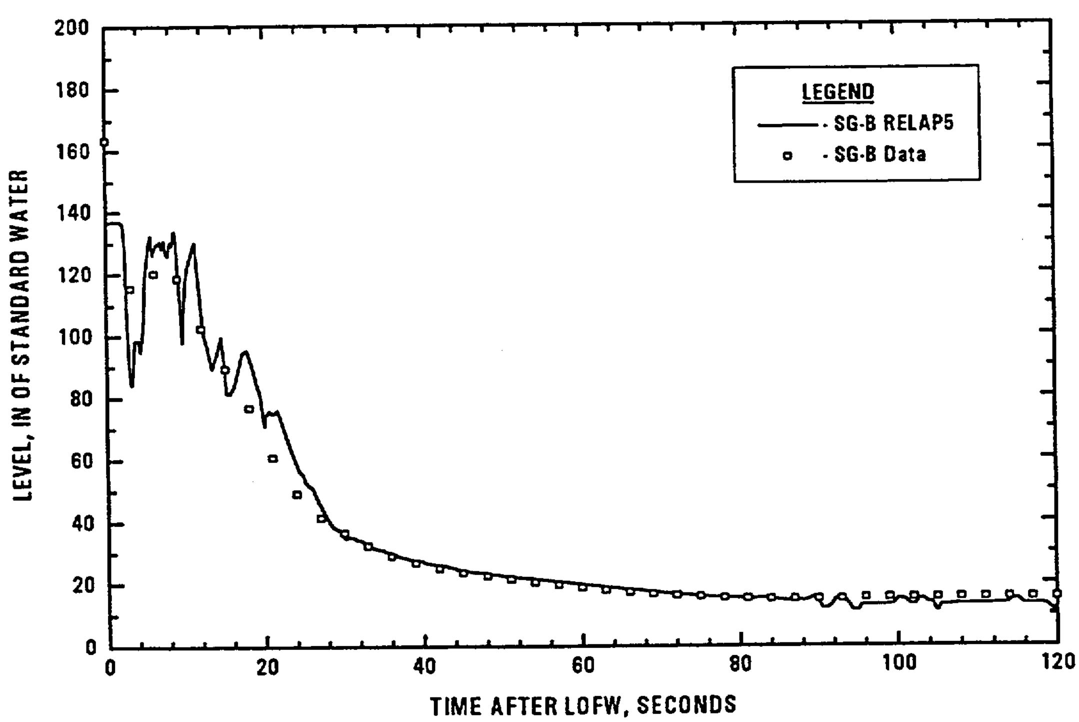


FIGURE 5-10. TEMPERATURE COMPENSATED PRESSURIZER LEVEL FOR THE TMI-2 LOFW EVENT.

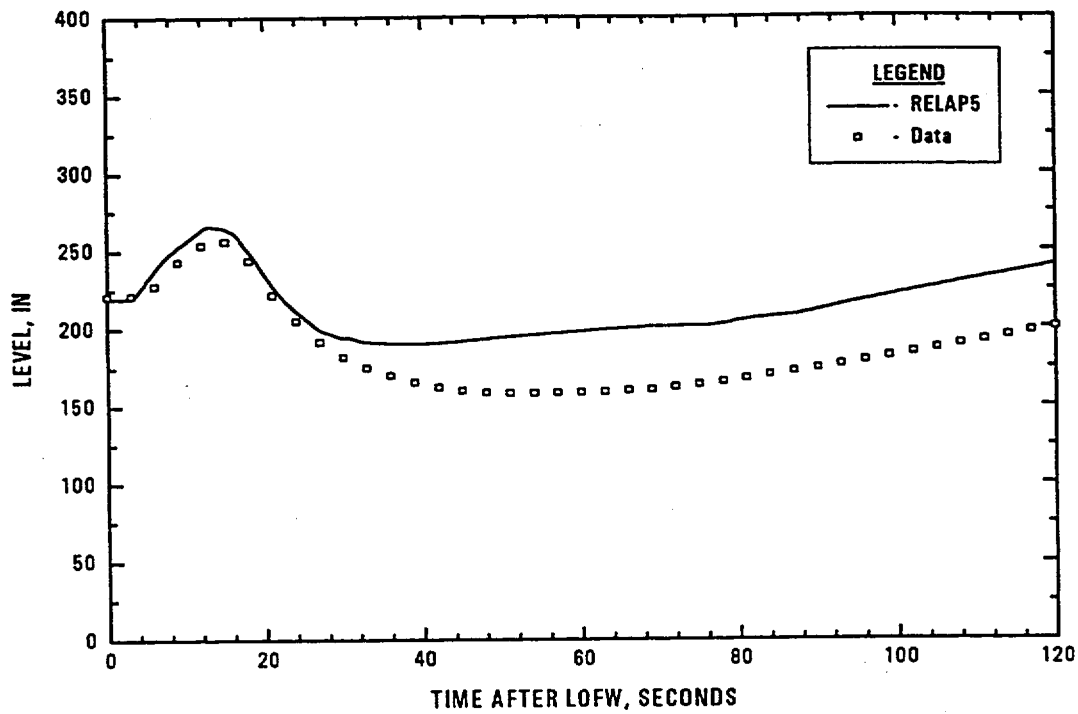


FIGURE 5-11. HIGH PRESSURE INJECTION FLOW DURING THE RANCHO-SECO LOSS-OF-ICS POWER EVENT.

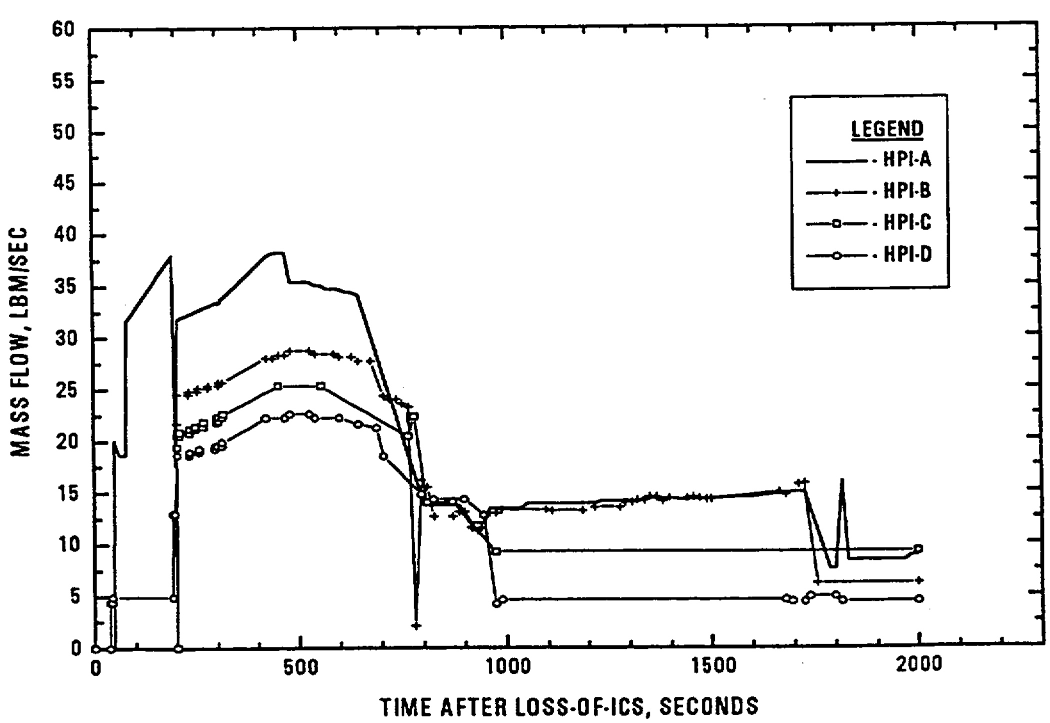


FIGURE 5-12. LOOP-A TEMPERATURES FOR THE RANCHO-SECO LOSS-OF-ICS POWER EVENT.

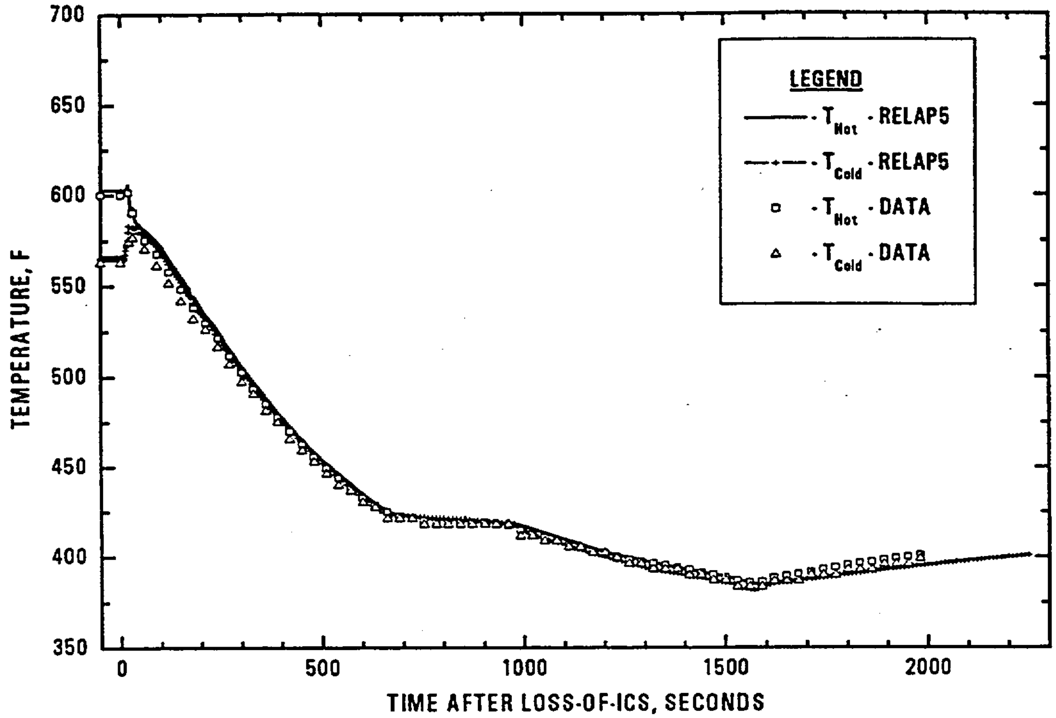


FIGURE 5-13. LOOP-B TEMPERATURES FOR THE RANCHO-SECO LOSS-OF-ICS POWER EVENT.

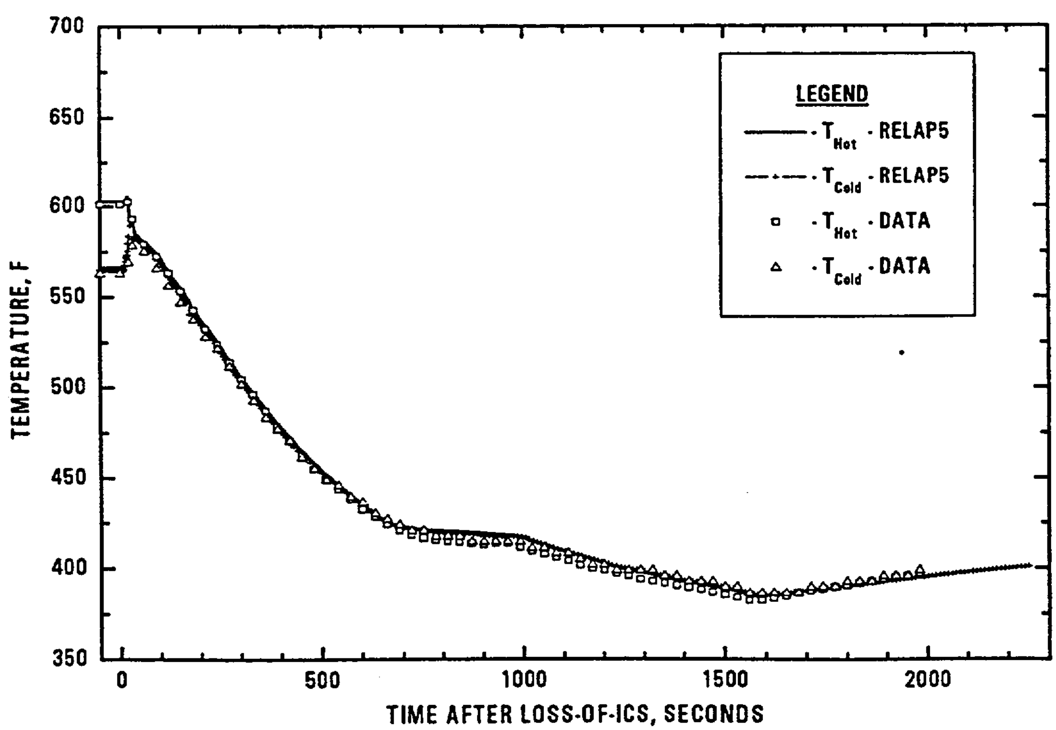


FIGURE 5-14. PRIMARY SYSTEM PRESSURE FOR THE RANCHO-SECO LOSS-OF-ICS POWER EVENT.

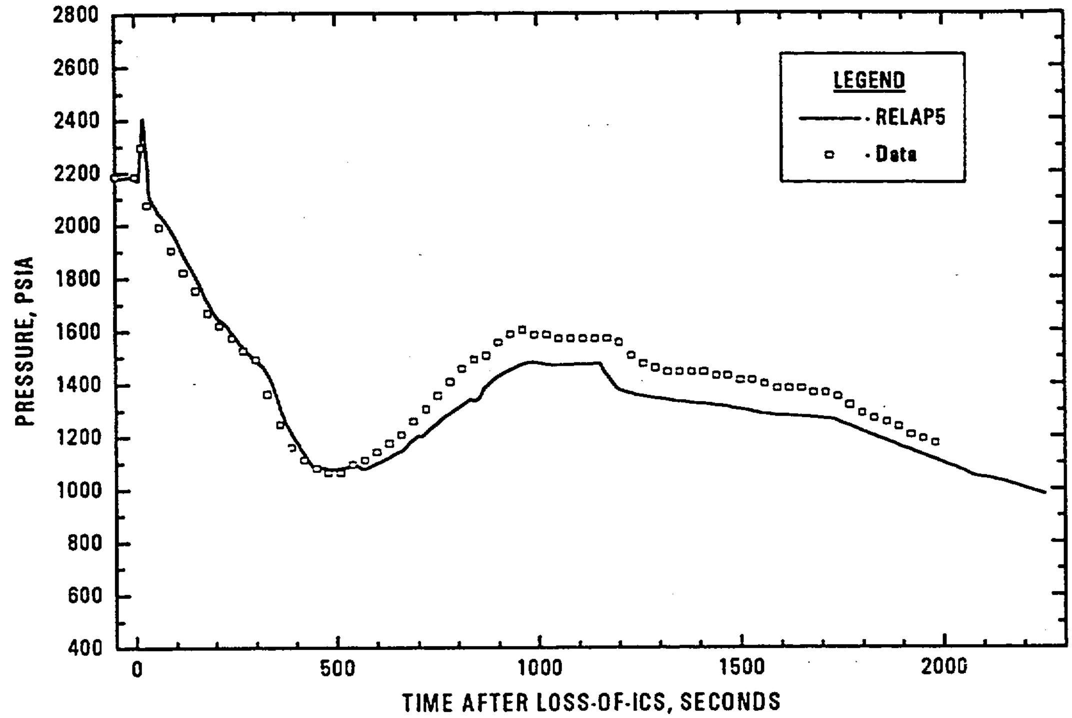


FIGURE 5-15. SECONDARY PRESSURES FOR THE RANCHO-SECO LOSS-OF-ICS POWER EVENT.

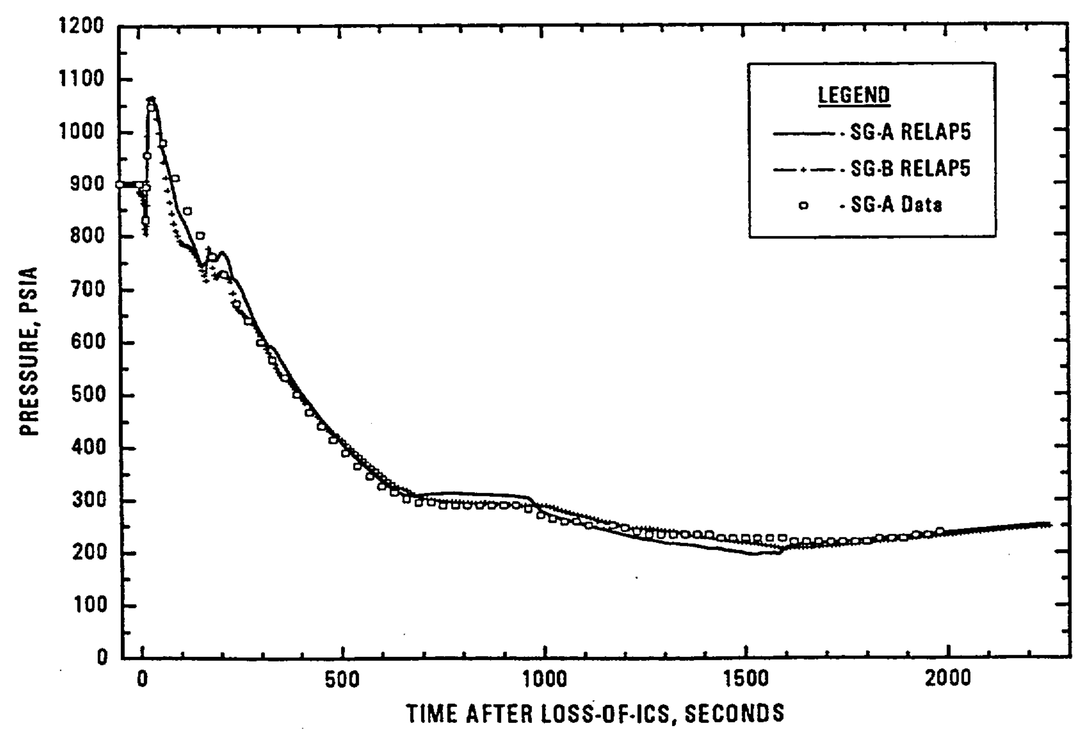


FIGURE 5-16. STEAM GENERATOR-A LIQUID LEVEL FOR THE RANCHO-SECO LOSS-OF-ICS POWER EVENT.

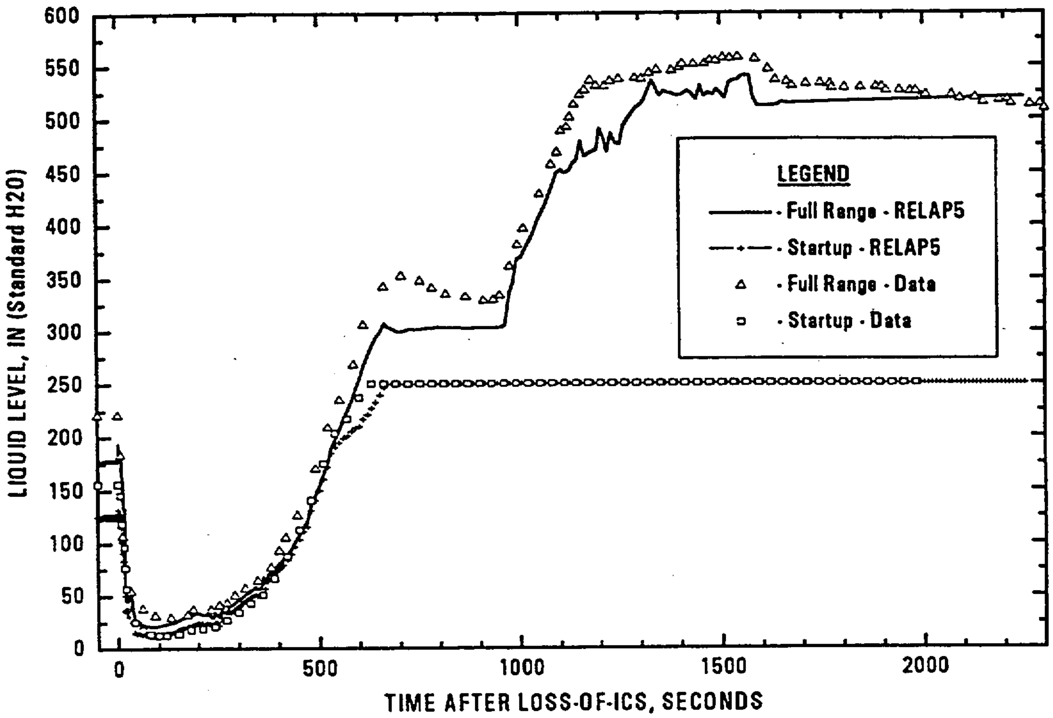


FIGURE 5-17. STEAM GENERATOR-B LIQUID LEVELS FOR THE RANCHO-SECO LOSS-OF-ICS POWER EVENT.

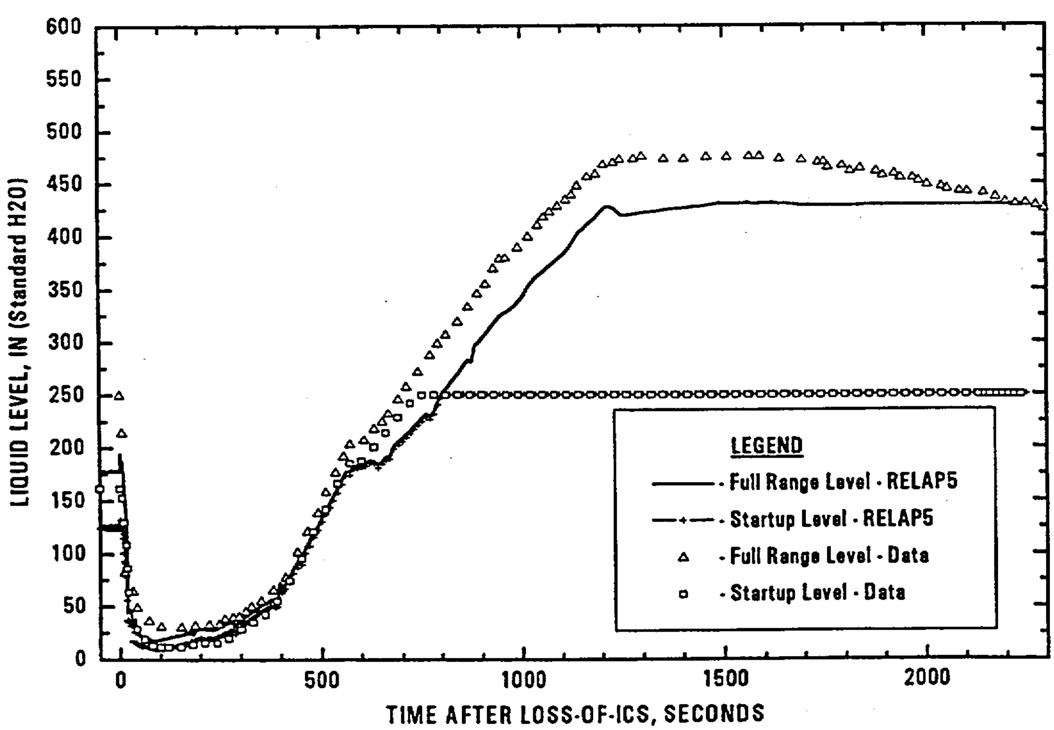


FIGURE 5-18. PRESSURIZER LEVEL FOR THE RANCHO-SECO LOSS-OF-ICS POWER EVENT.

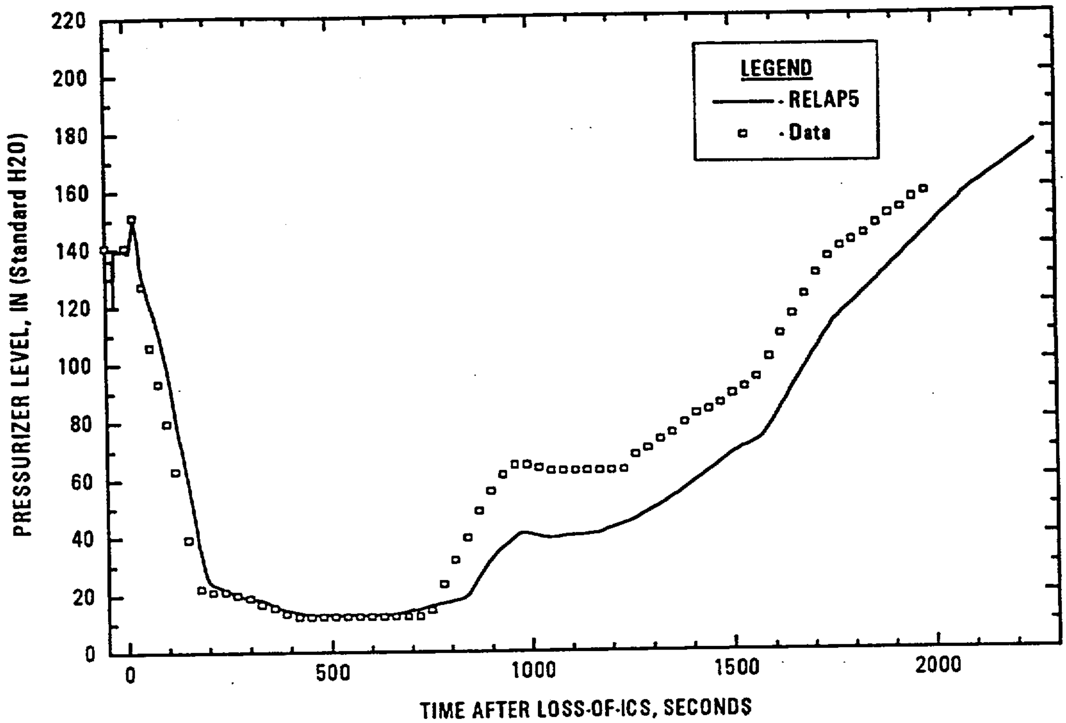


FIGURE 5-19. COMPARISON TO OCONEE AND CR-3 FLOW COASTDOWN DATA.

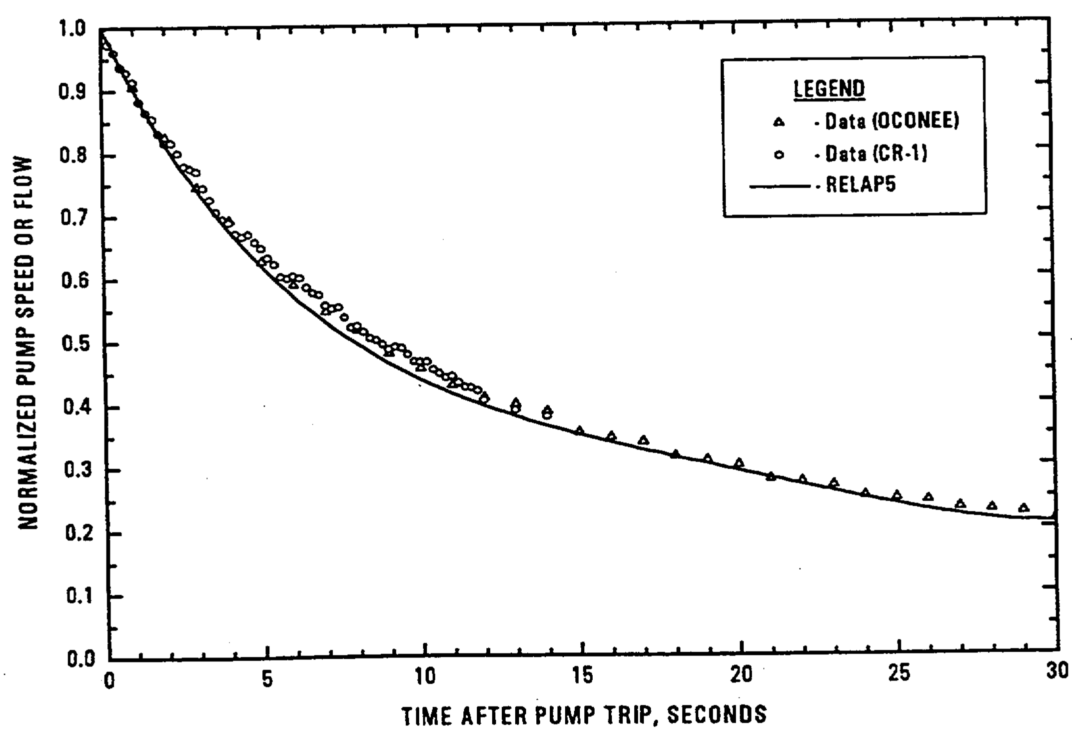


FIGURE 5-20. STEAM GENERATOR OPERATE RANGE LEVEL FOR THE TMI-1 NATURAL CIRCULATION TEST.

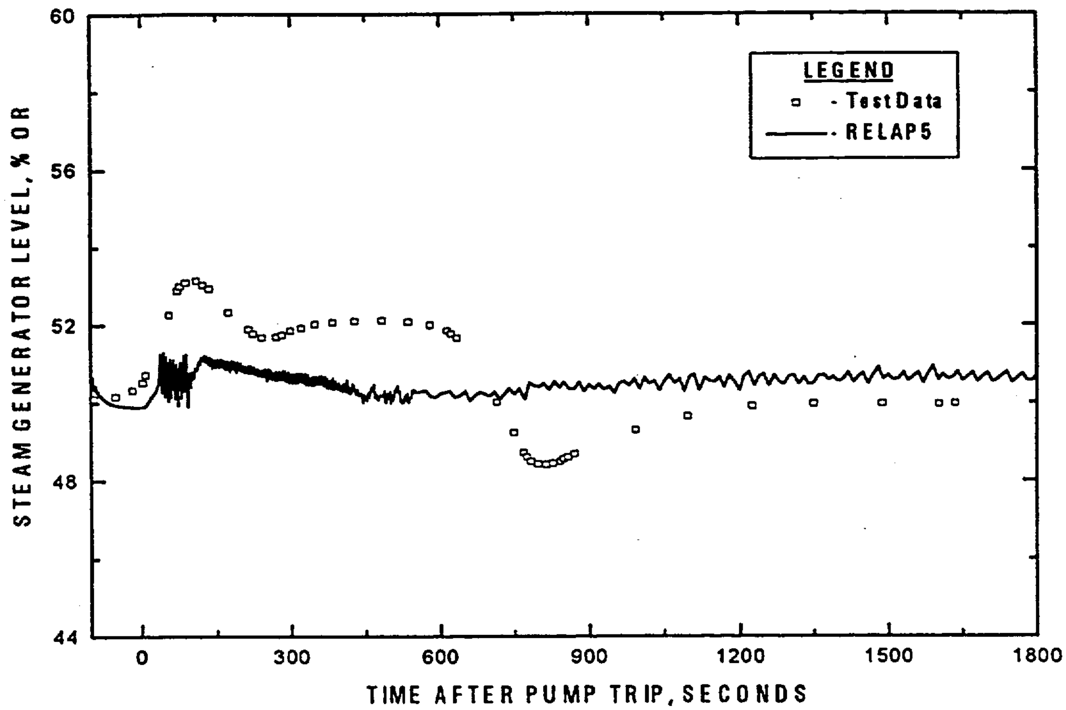


FIGURE 5-21. RCS HOT- AND COLD-LEG TEMPERATURES FOR THE TMI-1 NATURAL CIRCULATION TEST.

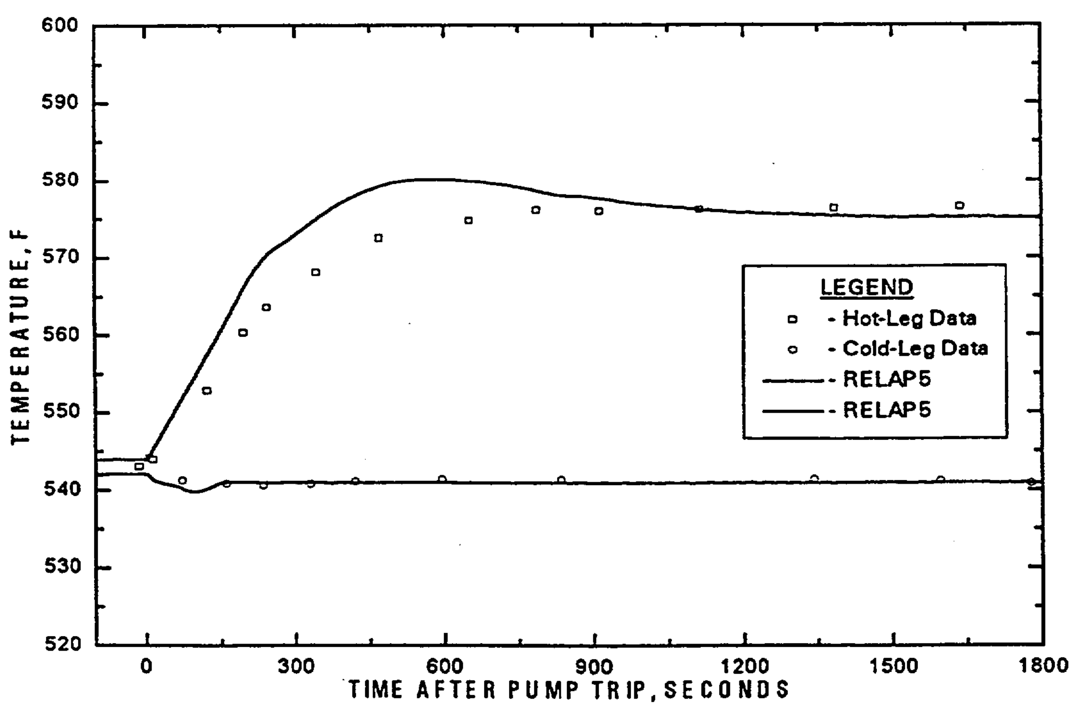


FIGURE 5-22. RCS HOT-LEG PRESSURE FOR THE TMI-1 NATURAL CIRCULATION TEST.

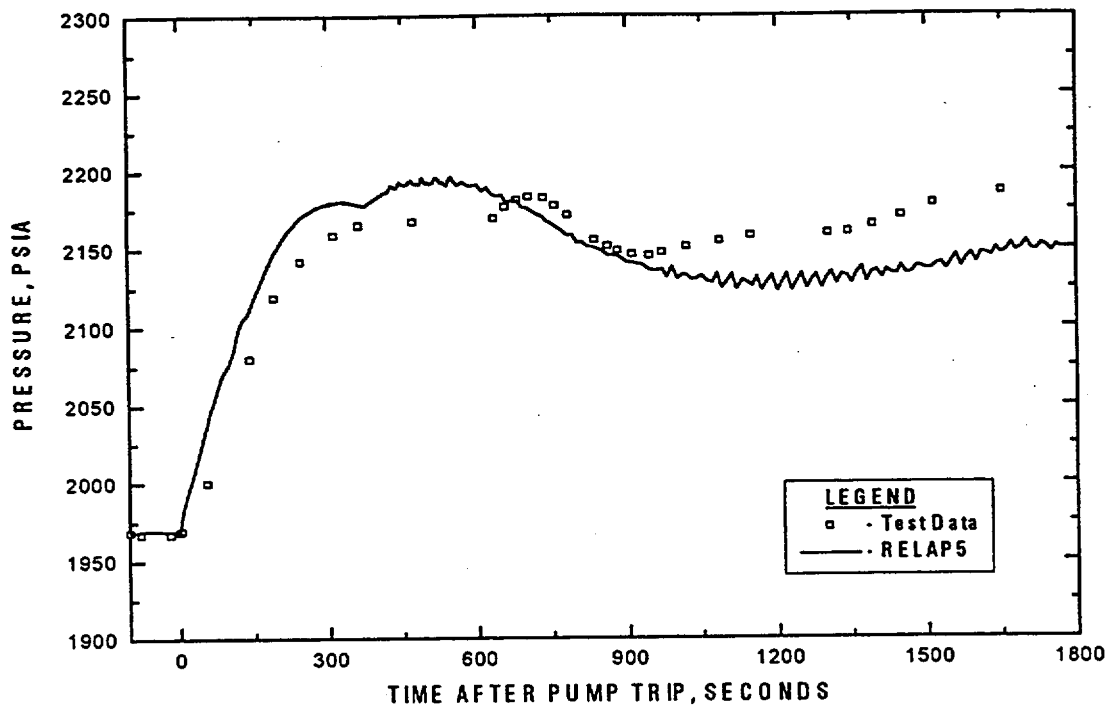
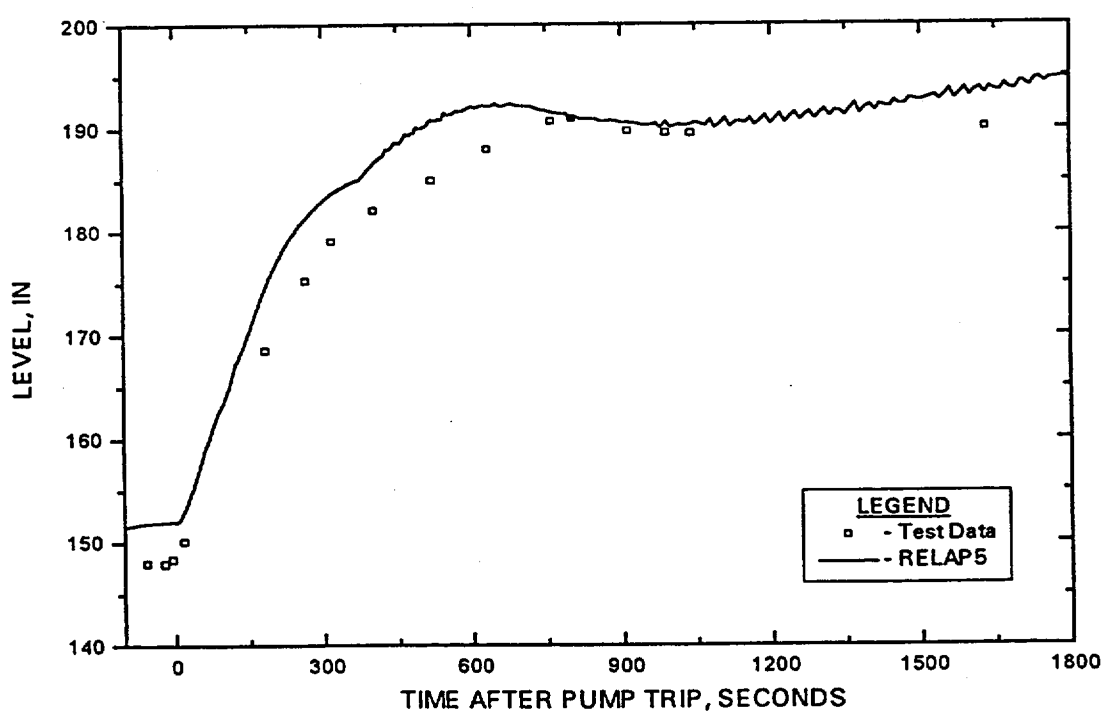


FIGURE 5-23. RCS PRESSURIZER LEVEL FOR THE TMI-1 NATURAL CIRCULATION TEST.



6. COMPARISON OF RELAP5/MOD2-B&W WITH CADDS AND TRAP2

The CADDS computer code (Computer Application to Direct Digital Simulation of Transients in Water Reactors With and Without Scram) was developed to analyze the transient response of B&W-designed pressurized water reactors for a variety of accidents. Although the code was approved by the Nuclear Regulatory Commission for simulation of a wide range of events, CADDS is primarily used to analyze control rod withdrawal, control rod ejection, loss of primary flow, reactor coolant pump startup, boron dilution and anticipated transients without scram. The TRAP2 (Transient Reactor Analysis Program) computer code is used to calculate the core power and system responses to steam line break, turbine trip, loss of feedwater, feedwater line break and steam generator tube rupture accidents.

It is the intent of FTG to replace these computer codes with RELAP5/MOD2-B&W. The ability of RELAP5/MOD2-B&W to predict the phenomena exhibited by the B&W-designed PWR was previously illustrated through benchmarks to test facility and plant transient data. The objective of these additional comparisons is to show that, by applying the conservative boundary and initial conditions used in safety analyses, RELAP5/MOD2-B&W predicts results similar to the NRC approved computer codes, CADDS and TRAP2. On that basis, it is concluded that RELAP5/MOD2-B&W is an acceptable replacement for the CADDS and TRAP2 computer codes for performing safety analyses on B&W-designed plants.

6.1 Comparison With CADDS Predictions of the Startup Event

The code comparison was performed for control rod assembly bank withdrawal from a low power condition (start-up event) because this reactivity insertion event is the limiting primary system overpressure event for the B&W 177-FA plant design. A spectrum of reactivity insertion rates was analyzed with RELAP5/MOD2-B&W for comparison with CADDS predictions. The time-dependent plant responses of a single bank withdrawal and withdrawal of all control rods are presented. In addition, comparisons of peak neutron power and peak thermal power, as a function of reactivity insertion rate, are also shown.

6.1.1 Description of the CADDS Computer Code

CADDS was developed to simulate the primary system responses to reactivity events, loss-of-flow events and anticipated transients without scram. It uses lumped single-loop modeling to simulate the transient effects of the reactor coolant system. The model includes only five nodes: the hot leg, the steam generator, the cold leg, the reactor core and core by-pass, and pressurizer. The primary system fluid conditions are assumed to remain subcooled. System flow around the single-loop is simulated via transport time delays.

The system pressure response is calculated using a detailed, non-equilibrium pressurizer model that includes the surge-line, heaters, sprays, and safety valve simulations. The pressure calculation neglects the effect of momentum in that primary system fluid volume changes are instantaneously transferred to or from the pressurizer. The surge line flow rate is used to determine the dynamic pressure drop, which is applied to the pressurizer pressure to yield reactor coolant system pressure.

The core power response is calculated using the point kinetics equations with separable reactivity feedback contributions by moderator density, moderator temperature and fuel Doppler effects. CADDS contains a fuel pin model with multiple radial nodes to calculate fuel and cladding temperatures using a one-dimensional radial heat transfer model. Multiple axial nodes can be used to simulate a flux shape and provide weighted reactivity feedback.

6.1.2 RELAP5/MOD2-B&W Model Used For Startup Event Analyses

A generic model of the B&W lowered-loop 177 fuel-assembly plant was used to perform the startup event comparisons to CADDS. This model was composed of:

a. Two hot legs.

b. Four cold legs with reactor coolant pumps.

c. Reactor vessel with core.

d. Two steam generators.

e. A pressurizer including pressurizer safety valves.

Unlike the CADDS model formulation, the RELAP5/MOD2-B&W model contains multiple control volumes to represent the hot legs and cold legs. In addition, reactor coolant pumps are explicitly modeled as compared to the CADDS model that used a specified flow rate and loop time delays. These differences are implicit to the two different model formulations and have little impact on the calculated results.

Certain features of the RELAP5/MOD2-B&W model were purposely altered to match the CADDS model formulation. For instance, the primary system fluid volume in the CADDS calculations did not include the fluid in the upper reactor vessel head region. Consequently, this region was deleted from the RELAP5/MOD2-B&W model. Also, only the fuel pin, steam generator tubing and pressurizer shell metal heat structures are included in the CADDS model formulation. Therefore, all other heat structures were deleted from the RELAP5/MOD2-B&W model. Similarly, the core region was modeled as a single control volume and heat structure to match the CADDS modeling. The steam generator heat transfer was modeled as a constant heat demand to mimic the CADDS simulation of the startup event.

A representation of the RELAP5/MOD2-B&W model is shown Figure 6-1. The reactor coolant system initial and boundary conditions are shown in Table 6-1. The reactor core neutron kinetics parameters and reactor protection trip setpoints are shown in Table 6-2.

6.1.3 Comparison of the Single Bank Withdrawal (SBW) Transient

A withdrawal of a single control bank from hot zero power was simulated by inserting 1.42×10^{-4} $\Delta k/k/s$ reactivity into the core. Doppler reactivity feedback terminated the power excursion before a high flux trip was reached (Figures 6-2 and 6-3). The heat addition to the primary system caused an increase in pressure (Figures 6-4 and 6-5) and subsequent reactor trip on high RCS pressure. Control rod insertion sharply reduced the reactor power and the rate of primary system pressurization. The overpressure transient was terminated following lift of the pressurizer safety valves. The RELAP5/MOD2-B&W predictions of system parameters are compared with the CADDS predictions in Figures 6-2 through 6-8. A sequence of events is provided in Table 6-3.

The neutron power, thermal power and fuel temperature responses predicted by RELAP5/MOD2-B&W are in good agreement with the CADDS predictions. The primary difference between the predictions is caused by a later reactor trip prediction by RELAP5/MOD2-B&W on high RCS pressure.

RELAP5/MOD2-B&W predicts a later reactor trip than CADDS because RELAP5/MOD2-B&W predicts a slower system pressurization. The pressurizer pressure response predicted by CADDS closely approximates an adiabatic compression of the pressurizer steam region. The pressurization rate predicted by RELAP5/MOD2-B&W, however, includes the real effects of condensation at the steam-liquid interface and on the surface of the pressurizer shell metal. Because RELAP5/MOD2-B&W predicts a later reactor trip than CADDS, the prediction of peak thermal power is greater than that predicted by CADDS. For the same reason, the RELAP5/MOD2-B&W prediction of peak primary system pressure is lower than the value predicted by CADDS. CADDS overpredicts the system pressure because it underpredicts steam condensation and because the pressurizer insurge rate is overpredicted after reactor trip.

After reactor trip, the core power decreases, which reduces the fluid expansion (heatup) rate. The mass flow rate into the pressurizer should reduce significantly, resulting in a decrease in the pressurization rate. RELAP5/MOD2-B&W properly predicted this sequence of events. However, in the CADDS prediction, the rate of primary system pressurization remained unchanged following reactor trip, and the pressure transient was terminated by safety valve lift. Since CADDS overpredicted the system pressurization rate at the time of safety valve lift, CADDS overpredicted the peak system pressure.

6.1.4 Comparison of the All Rods Withdrawal (ARW) Transient

A withdrawal of all control rods from hot zero power was simulated by inserting 8.89×10^{-4} $\Delta k/k/s$ reactivity into the core. The increase in neutron power caused a reactor trip on high flux (Figure 6-9). Doppler reactivity feedback terminated the power excursion prior to control rod insertion. Control rod insertion provided long term core shutdown. The event terminated quickly because of the reactor trip on high flux, such that the thermal power did not exceed 55 percent (Figure 6-10). The heat addition caused an increase in primary system pressure (Figures 6-11 and 6-12), but, the pressure excursion was much less severe than that for the single bank withdrawal from hot zero power. The RELAP5/MOD2-B&W predictions of system parameters are compared to the CADDS predictions in Figures 6-9 through 6-15. A sequence of events is provided in Table 6-4.

Similar to the SBW transient, the RELAP5/MOD2-B&W predictions of neutron power (Figure 6-9), thermal power (Figure 6-10) and fuel temperature (Figure 6-13) are in good agreement with the CADDS predictions. In addition, the CADDS predictions of primary system and pressurizer pressures (Figures 6-11 and 6-12) are greater than the RELAP5/MOD2-B&W predictions for the reasons previously stated. Specifically, the primary system pressurization rate predicted by CADDS is not consistent with the reduction in thermal power following reactor trip.

6.1.5 Comparison of Intermediate Rod Withdrawal Results

The methodology for analysis of the startup event requires that a spectrum of reactivity insertion rates be analyzed to determine the worse case system response. The RELAP5/MOD2-B&W predictions of peak neutron power versus reactivity insertion rate are compared to the CADDS predictions in Figure 6-16. Similarly, the RELAP5/MOD2-B&W predictions of peak thermal power versus reactivity insertion rate are compared to the CADDS predictions in Figure 6-17. These figures demonstrate that the RELAP5/MOD2-B&W predictions of peak neutron and thermal power are similar to, but greater than, the CADDS predictions. In addition, the reactivity insertion that causes coincident high RCS pressure and high flux reactor trips is the same for both code calculations.

6.1.6 Conclusions

The RELAP5/MOD2-B&W predictions of core response for the startup accident are similar to the CADDS predictions for a range of reactivity insertion rates. In fact, RELAP5/MOD2-B&W conservatively predicts peak core thermal power as compared with CADDS.

RELAP5/MOD2-B&W predicts a lower peak reactor coolant system pressure as compared to CADDS because the CADDS pressure prediction is overly conservative. The RELAP5/MOD2-B&W prediction of system pressure properly reflects the effects of changes in primary system fluid expansion rates and includes pressurizer steam condensation effects. Since RELAP5/MOD2-B&W conservatively predicts the core thermal response during the startup event, the peak system pressure predicted by the code is conservative. Consequently, RELAP5/MOD2-B&W is an adequate tool for analyzing the system response during reactivity transients on B&W-designed pressurized water reactors.

6.2. Comparison With TRAP2 Predictions of Main Steam Line Break

RELAP5/MOD2-B&W predictions of main steam line break (MSLB) accidents are compared with MSLB predictions made by TRAP2 for the Midland FSAR. TRAP2 predictions of MSLB response were selected for comparison with RELAP5/MOD2-B&W because this severe overcooling event sets the core design limit for end-of-cycle moderator temperature coefficient. Two cases were simulated. The first case is the double-ended rupture (DER) of a main steam line, yielding a break of 6.282 square feet. The second case is a single-ended rupture (SER), or split break, of 2.0 square feet.

6.2.1 Description of the TRAP2 Computer Code and Model

TRAP2 was developed to simulate the plant response to secondary system upset conditions. A control volume approach is used to model the primary and secondary systems of the B&W-designed pressurized water reactor. Fluid conditions are maintained in equilibrium. Therefore, conservation of mass, energy and momentum are solved for a single fluid. A constant bubble rise velocity is typically input by the user to simulate phase slip. Primary-to-secondary heat transfer is simulated by solving a one-dimensional heat transfer equation. Convective heat transfer coefficients for forced single-phase flow, subcooled nucleate boiling, saturated nucleate boiling or film boiling are calculated based on the fluid properties.

Models are available to simulate reactor coolant pumps, main feedwater pumps, control and isolation valves, safety injection, accumulators, a non-equilibrium pressurizer, and the core. The core power response is calculated using the point kinetics equations with separable reactivity feedback contributions by moderator density and fuel Doppler effects. TRAP2 uses a simple fuel pin model to calculate fuel and cladding temperatures using a one-dimensional radial heat transfer model.

6.2.2 RELAP5/MOD2-B&W MSLB Model

A generic model of the B&W lowered-loop 177 fuel-assembly plant was used to perform the MSLB comparisons with TRAP2. This model was composed of:

a. Two hot legs.

b. Four cold legs with reactor coolant pumps.

c. Reactor vessel with core.

d. Two steam generators.

e. A pressurizer including pressurizer safety valves.

Certain features of the RELAP5/MOD2-B&W model were altered to match the TRAP2 model. These modifications were made in the areas of primary system fluid volume, primary system heat structures, core modeling, secondary system piping and fill systems. The primary system fluid volume in the TRAP2 calculations did not include the fluid in the upper reactor vessel head region. Consequently, this region was deleted from the RELAP5/MOD2-B&W model.

Also, only the fuel pin and steam generator tubing heat structures were included in the TRAP2 model of the Midland plant. Therefore, all other heat structures were deleted from the RELAP5/MOD2-B&W model. Similarly, the core region was modeled as a single control volume and heat structure to match the TRAP2 modeling.

Secondary steam piping, feedwater piping, and a feedwater pump simulation were added to match those in the TRAP2 model. The break geometry and critical flow model (Moody) were the same as those used in the TRAP2 model. The auxiliary feedwater flow table for the unaffected steam generator was set to match the TRAP2 input. This was also done for the high pressure injection flow versus pressure tables.

A representation of the RELAP5/MOD2-B&W model is shown Figure 6-18. The reactor coolant system initial and boundary conditions are shown in Table 6-5. The reactor core neutron kinetics parameters and reactor protection trip setpoints are shown in Table 6-6. The engineered safety features actuation system (ESFAS) setpoints and delays are shown in Table 6-7.

6.2.3 Comparison With TRAP2 Predictions of a DER MSLB

The TRAP2 prediction was initiated by opening a double-ended rupture of a 28 inch main steam line (Figure 6-19) and by instantaneous termination of steam flow to the turbine. Both steam generators depressurized in response to the break (Figure 6-20). The decrease in secondary saturation temperatures caused an increase in primary-to-secondary heat transfer. This caused the primary system pressure to decrease (Figure 6-21), causing a reactor trip on low primary system pressure. Shortly thereafter, an ESFAS trip occurred on low primary system pressure, initiating main steam isolation valve (MSIV) closure, main feedwater isolation valve (MFIV) closure, auxiliary feedwater flow to the unaffected steam generator and high pressure injection flow.

Following MSIV closure, the primary system depressurization slowed as the unaffected steam generator repressurized and became a heat source (Figure 6-22). The depressurization rate of the affected steam generator increased because it then provided steam to both sides of the break.

Following MFIV closure, the affected steam generator dried out as the unaffected steam generator continued to fill from the auxiliary feedwater flow (Figure 6-23). Affected steam generator dryout was delayed by approximately ten seconds as liquid in the feedwater piping began to flash, pushing liquid into the steam generator. Just prior to steam generator dryout, the minimum subcritical margin (Figure 6-24) and maximum subcritical multiplication (Figure 6-25) were observed.

The RELAP5/MOD2-B&W predictions are compared with those of TRAP2 in Figures 6-19 through 6-28 and in Table 6-8. The RELAP5/MOD2-B&W predictions of break flow, primary pressure, secondary pressure, primary system temperature, core reactivity and core power are in good agreement with those of TRAP2. The principle differences in the predictions are due to differences in the calculated phase-slip in the secondary system. The TRAP2 bubble rise velocities input on the secondary side were typically 0.5 ft/s. This means that the TRAP2 control volume fluid conditions were effectively homogeneous. Consequently, RELAP5/MOD2-B&W predicted the minimum cold leg temperature to occur earlier than the TRAP2 prediction. Because RELAP5/MOD2-B&W more accurately predicted the phase-slip in the steam generator, it calculated a shorter boiling length than did TRAP2, resulting in less heat transfer. More importantly, RELAP5/MOD2-B&W predicted steam generator dryout ten seconds earlier than did TRAP2. Because RELAP5/MOD2-B&W predicted phase-slip in the feedwater piping control volume, less liquid was transported into the steam generator as the piping liquid flashed. Consequently, RELAP5/MOD2-B&W predicted earlier steam generator dryout as compared with TRAP2, and the associated predictions of primary system temperature, core power, and core reactivity were less severe than the TRAP2 predictions.

6.2.4 Comparison With TRAP2 Predictions of a 2 ft^2 SER MSLB

The predicted plant response to a 2 ft^2 single ended rupture (split break) is similar to that for a double-ended rupture (DER). Because the break area is one third of the DER, the steam generator takes longer to blow down following the split break. The TRAP2 prediction was initiated by opening a 2 ft^2 split rupture of a main steam line (Figure 6-29) and by instantaneous termination of steam flow to the turbine. Both steam generators depressurized in response to the break (Figure 6-30). The decrease in secondary saturation temperatures caused an increase in primary-to-secondary heat transfer. Consequently, the primary system temperature decreased (Figure 6-31). When the colder water entered the core, the core fission power increased (Figure 6-32) causing a reactor trip on high flux. The overcooling caused the primary system pressure to decrease

(Figure 6-33), resulting in an ESFAS trip on low primary system pressure. ESFAS initiated main steam isolation valve (MSIV) closure, main feedwater isolation valve (MFIV) closure, auxiliary feedwater flow to the unaffected steam generator, and high pressure injection flow.

Following MSIV closure, the unaffected steam generator repressurized and became a heat source (Figure 6-34). The depressurization rate of the affected steam generator increased because it was then the sole source of steam for the break.

Following MFIV closure, the affected steam generator dried out as the unaffected steam generator continued to fill from the auxiliary feedwater flow (Figure 6-35). Affected steam generator dryout was delayed as liquid in the feedwater piping began to flash, pushing liquid into the steam generator. Just prior to steam generator dryout, the minimum subcritical margin (Figure 6-36) and maximum subcritical multiplication (Figure 6-32) were observed.

The RELAP5/MOD2-B&W predictions are compared with those of TRAP2 in Figures 6-29 through 6-38 and in Table 6-9. The RELAP5/MOD2-B&W predictions of break flow, primary pressure, secondary pressure, primary system temperature, core reactivity and core power are in good agreement with those of TRAP2. The principle differences in the predictions are due to differences in the calculated phase-slip in the secondary system.

As discussed in Section 6.2.3, the TRAP2 control volume fluid conditions were effectively homogeneous. Consequently, TRAP2 overpredicted the boiling length throughout the event resulting in a conservative prediction of the heat transfer. Furthermore, TRAP2 overpredicted the transport of liquid from the feedwater piping control volume as that liquid flashed, conservatively delaying dryout. At approximately 50 seconds, the phase-slip predicted by RELAP5/MOD2-B&W resulted in a boiling length that was insufficient to support the core power. The cold leg temperature increased momentarily. Flashing of the feedwater piping inventory, that started at 54 seconds, provided liquid to the steam generator to continue the cooldown. However, the average primary system temperature predicted by RELAP5/MOD2-B&W remained higher than what TRAP2 predicted for the duration of the event. As a result, the RELAP5/MOD2-B&W predictions of core power and core reactivity were less severe than the TRAP2 predictions.

6.2.5 Conclusions

Using the same initial and boundary conditions as were used in the TRAP2 calculations, RELAP5/MOD2-B&W was used to simulate the plant response to a double-ended MSLB and a two square foot split MSLB. The RELAP5/MOD2-B&W predictions of break flow, primary pressure, secondary pressure, primary system temperature, reactor trip time, ESFAS time, core reactivity, and core power were in good agreement with those of TRAP2.

The differences between the TRAP2 and RELAP5/MOD2-B&W predictions of minimum primary system temperature, minimum subcritical margin, and maximum subcritical multiplication were caused by differences in secondary side phase-slip modeling. RELAP5/MOD2-B&W mechanistically calculates the phase-slip in the steam generator and feedwater piping control volumes. The TRAP2 fluid conditions were effectively homogeneous resulting in an overprediction of the steam generator boiling length and delayed steam generator dryout.

These comparisons show that, given conservative boundary and initial conditions, the RELAP5/MOD2-B&W computer code provides conservative results, similar to those predicted by the TRAP2 computer code. Consequently, RELAP5/MOD2-B&W is appropriate for performing non-loss-of-coolant accident analyses on B&W-designed pressurized water reactors.

Table 6-1. RCS Initial and Boundary Conditions For Startup Event Comparison

PARAMETER	CADDS	RELAP5/MOD2
Total RCS Volume, ft^3	11268	11256
Pressurizer Liquid Volume, ft^3	800 31050	814 30562
RCS Temperature, F	532.1	532.1
Pressurizer Pressure, psia	2175	2170
RCS Pressure, psia	2182	2182
Reactor Power, W	2.501	2.501
Pressurizer Safety Valve Lift Setpoint, psia	2590	2590
Pressurizer Safety Valve Mass Flow Rate, lb/s/vlv @ 2590 psia	83.33	83.33
Number of Pressurizer Safety Valves	2	2

Table 6-2. Reactor and Reactor Protection System Parameters For The Startup Event Comparison

Moderator Temperature Coefficient, $\Delta k/k/F$	$+ 0.9 \times 10^{-4}$
Doppler Coefficient, $\Delta k/k/F$	$- 1.46 \times 10^{-5}$
Delayed Neutron Fraction, β_{eff}	0.007297
Prompt Neutron Generation Time, μs	33.5
High Flux Trip Setpoint, MW	2746.2
High RCS Pressure Trip Setpoint, psia	2406
High Flux Trip Delay, s	0.4
High RCS Pressure Trip Delay, s	0.7
Total Tripped Control Rod Worth, $\%\Delta k/k$	2.13

Table 6-3. Sequence of Events For The Single Bank Withdrawal From Hot Zero Power

Event	Time, Sec CADDS	Time, Sec RELAP5/MOD2
Rod Withdrawal Begins	0.0	0.0
Reactor Trip on High RCS Pressure	49.6	50.7
Control Rods Begin to Drop	50.3	51.4
Peak Thermal Power is Reached (Peak Thermal Power, percent)	51.0 (52.9)	52.1 (55.2)
Pressurizer Safety Valves Lift (Peak RCS Pressure, psia)	52.7 (2688.7)	55.5 (2627.6)

Table 6-4. Sequence of Events For The All Rod Withdrawal From Hot Zero Power

Event	Time, Sec CADDS	Time, Sec RELAP5/MOD2
Rod Withdrawal Begins	0.0	0.0
Reactor Trip on High Flux	8.9	8.9
Control Rods Begin to Drop	9.3	9.3
Peak Thermal Power is Reached (Peak Thermal Power, percent)	10.1 (50.4)	10.1 (50.0)
Pressurizer Safety Valves Lift (Peak RCS Pressure, psia)	20.4 (2688.7)	N/A (2608.7)

Table 6-5. Steady-State Conditions for the DER MSLB Benchmark

Parameter	TRAP2	RELAP5/MOD2
Core Thermal Power, Mw	2505.44	2505.44
Primary System Pressure, psia	2169.7	2172.8
Secondary System Pressure, psia	929.4	937.9*
Average Primary System Temperature, F	578.7	578.1
Cold Leg Temperature, F	554.4	554.2*
Primary System Flow Rate, lbm/s	36429	37545
Main Feedwater Flow Rate, lbm/s	2904	2880
Primary System Mass, lbm	486317	492364
Secondary System Mass, lbm/SG	44971	45155*

*Average of both loops.

Table 6-6. Reactor and Reactor Protection System Parameters For The MSLB Comparison

Moderator Temperature Coefficient, Δk/k/F	-3.0×10^{-4}
Doppler Coefficient, Δk/k/F	-1.85×10^{-5}
Delayed Neutron Fraction, β_{eff}	0.005401
Inverse Boron Worth, ppm/%Δk/k	96
Prompt Neutron Generation Time, μs	31.2
High Flux Trip Setpoint, percent	112
Low RCS Pressure Trip Setpoint, psia	1852
High Flux Trip Delay, s	0.4
High RCS Pressure Trip Delay, s	0.7
Scram Reactivity, %Δk/k	3.9

Table 6-7. Engineered Safety Feature Actuation System Setpoints and Delays Used in MSLB Comparisons

Low Primary System Pressure Setpoint, psia	1450
Low Steam Generator Pressure Setpoint, psia	600
High Pressure Injection Delay, s	26.4
Auxiliary Feedwater Flow Delay, s	15.0
Main Steam Isolation Valve Closure Delay, s	7.5
Main Steam Isolation Valve Closure Time, s	0
Main Feedwater Isolation Valve Closure Delay, s	0
Main Feedwater Isolation Valve Closure Time, s	17

Table 6-8. Sequence of Events for the DER MSLB Comparison

	Time, Sec	
Event	TRAP2	RELAP5/MOD2
Break Initiation	0.0	0.0
Reactor Trip on Low RCS Pressure	3.4	3.6
Control Rods Begin to Drop	4.1	4.3
ESFAS Trip on Low RCS Pressure	5.5	5.5
MSIVs Closed	13.0	13.0
AFW Flow to Unaffected SG Begins	20.5	20.5
MFIVs Closed	22.5	22.5
HPI Flow Begins	31.9	31.9

Table 6-9. Sequence of Events for the SER MSLB Comparison

Event	Time, Sec TRAP2	Time, Sec RELAP5/MOD2
Break Initiation	0.0	0.0
Reactor Trip on High Flux	6.4	6.4
Control Rods Begin to Drop	6.8	6.8
ESFAS Trip on Low RCS Pressure	21.8	20.7
MSIVs Closed	29.3	28.2
AFW Flow to Unaffected SG Begins	36.8	35.7
MFIVs Closed	38.8	37.7
HPI Flow Begins	48.2	47.1

FIGURE 6-1. RELAP5/MOD2-B&W MODEL USED FOR THE CADDS BENCHMARKS.

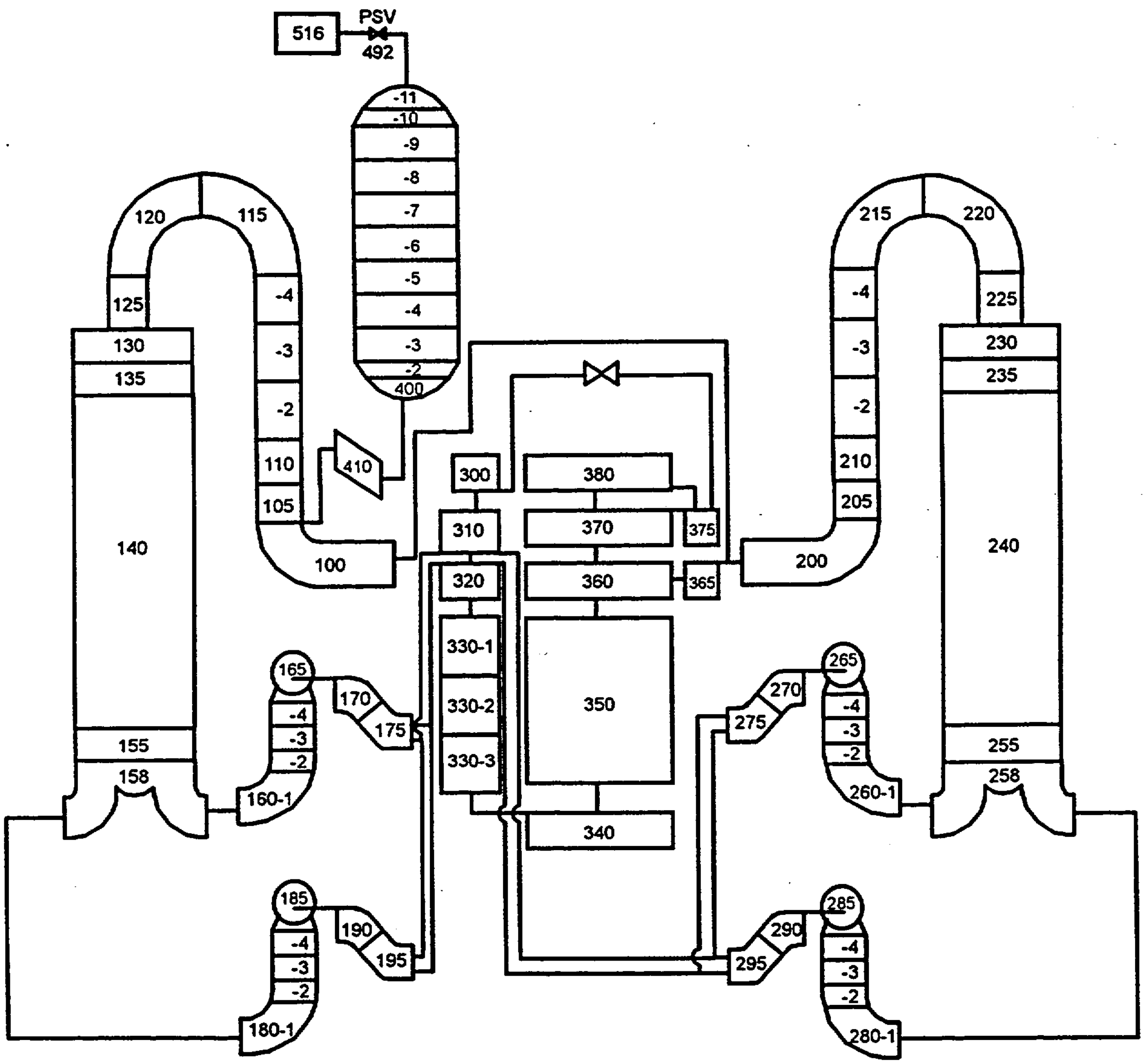


FIGURE 6-2. NEUTRON POWER PERCENTAGE OF RATED POWER FOR SBW EVENT.

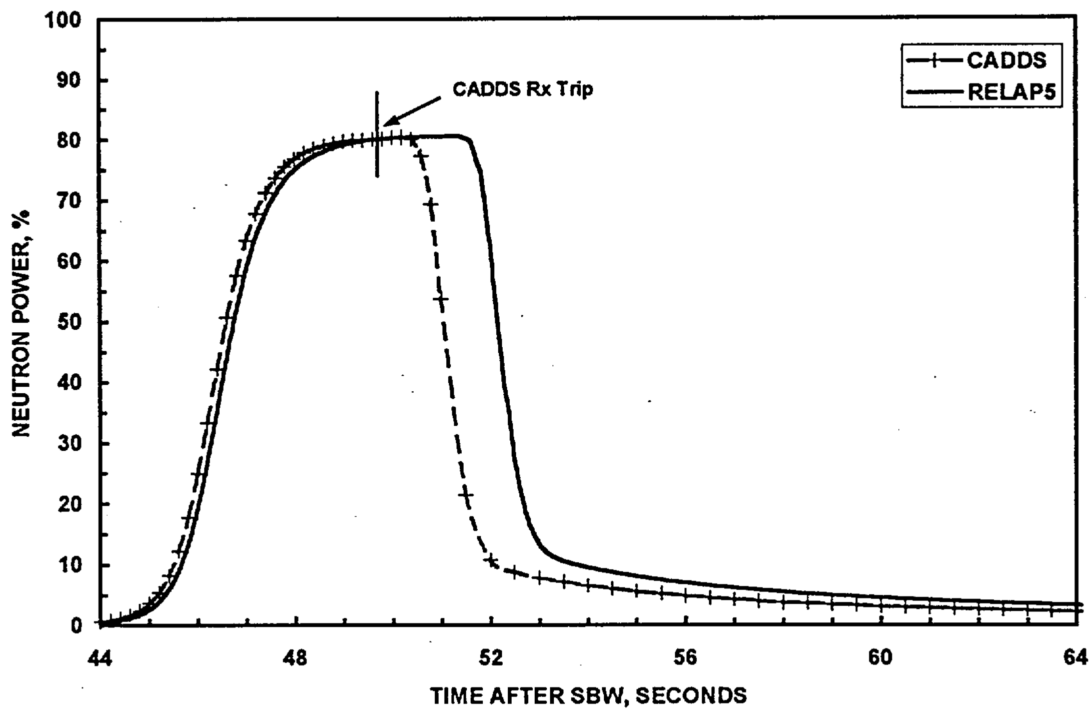


FIGURE 6-3. THERMAL POWER PERCENTAGE OF RATED POWER FOR SBW EVENT.

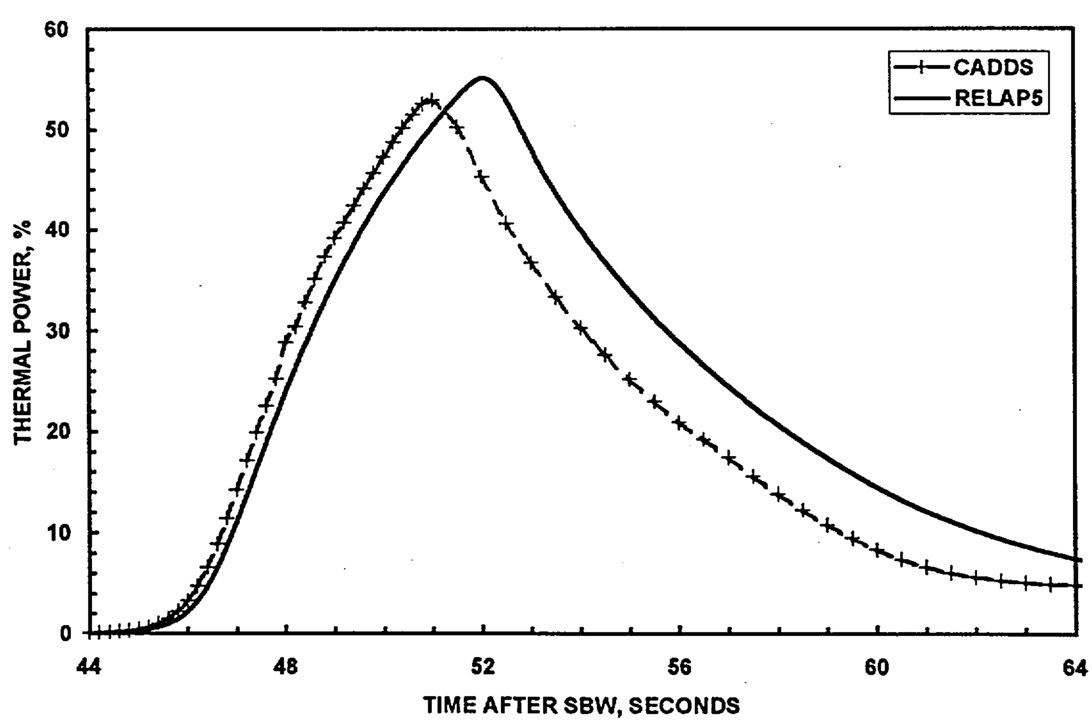


FIGURE 6-4. RCS SYSTEM PRESSURE FOR SBW EVENT.

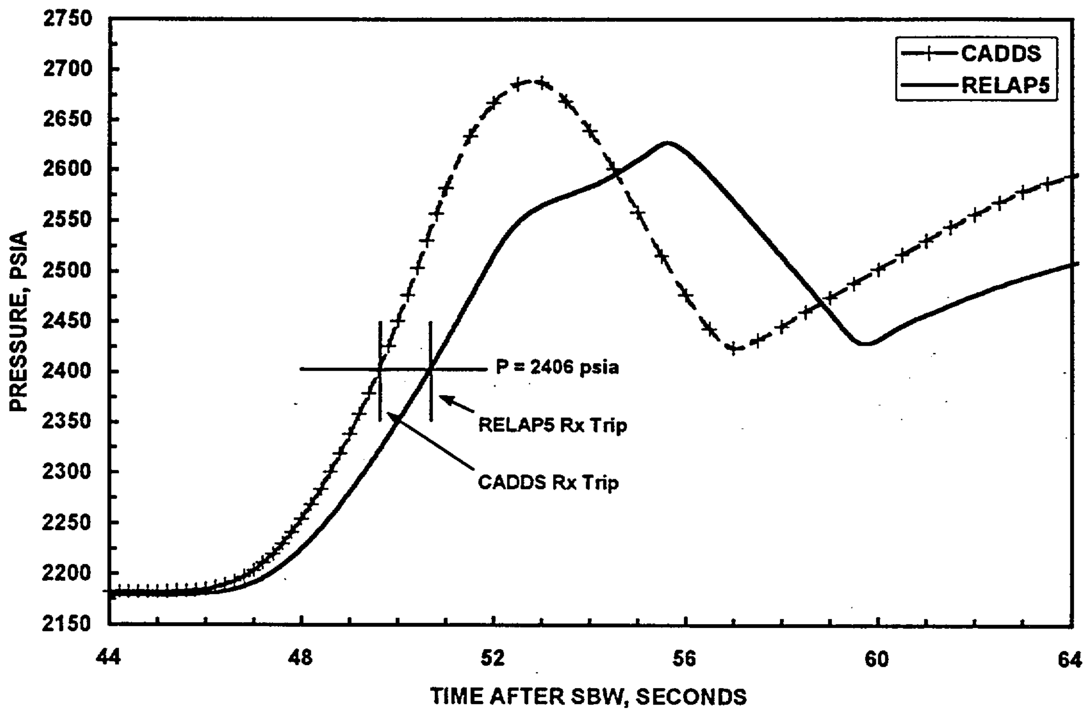


FIGURE 6-5. PRESSURIZER PRESSURE PREDICTIONS FOR SBW EVENT.

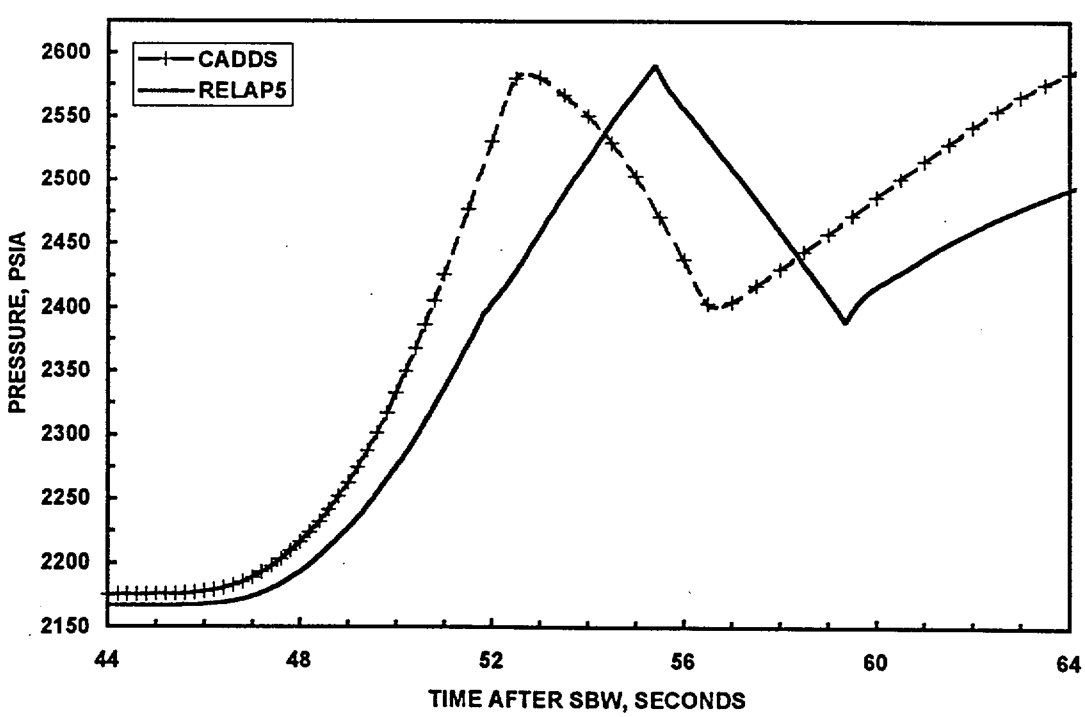


FIGURE 6-6. AVERAGE FUEL TEMPERATURE CHANGE FOR SBW EVENT.

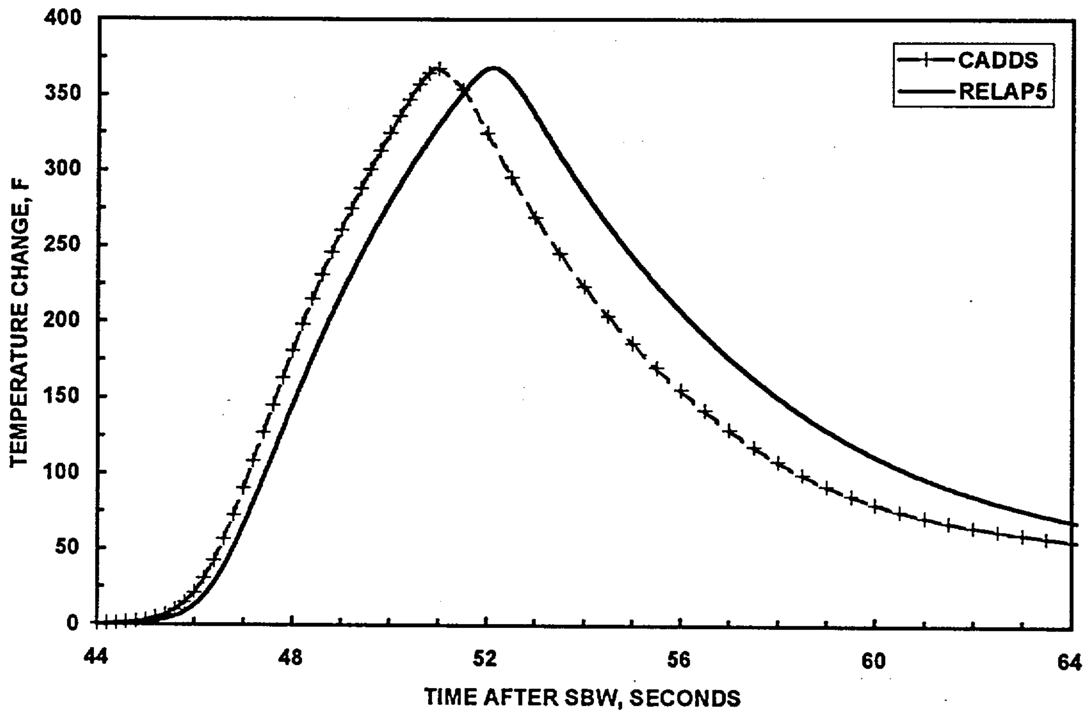


FIGURE 6-7. AVERAGE CORE MODERATOR TEMPERATURE CHANGE FOR SBW EVENT.

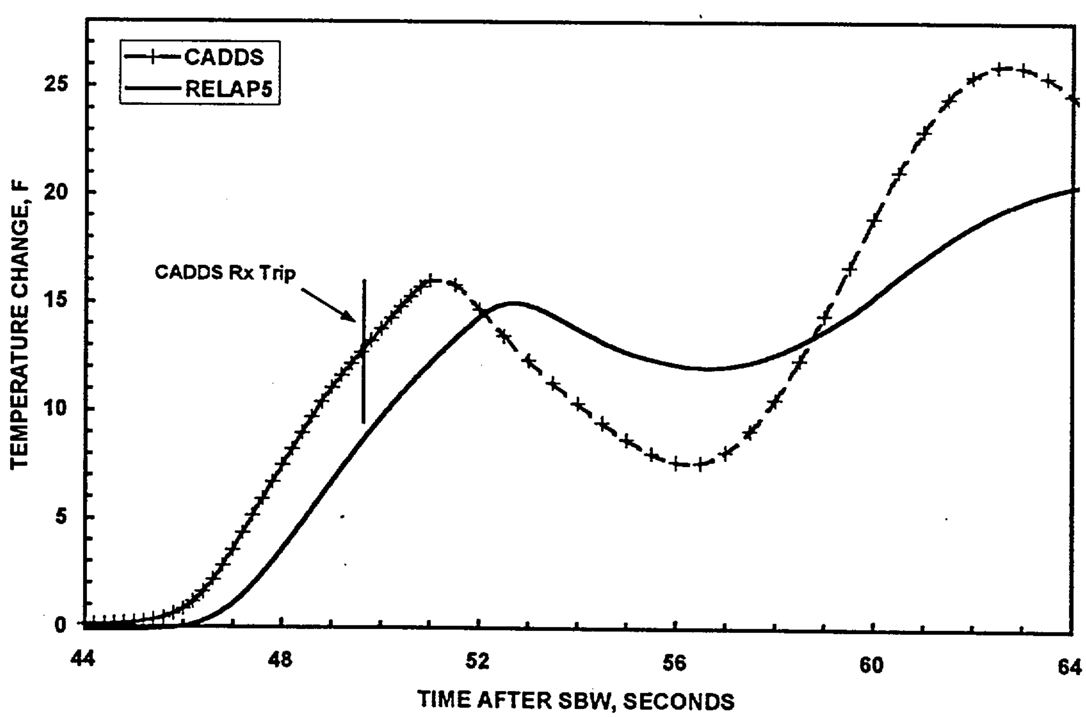


FIGURE 6-8. SURGELINE AND PRESSURIZER SAFETY VALVE MASS FLOW RATES FOR SBW EVENT.

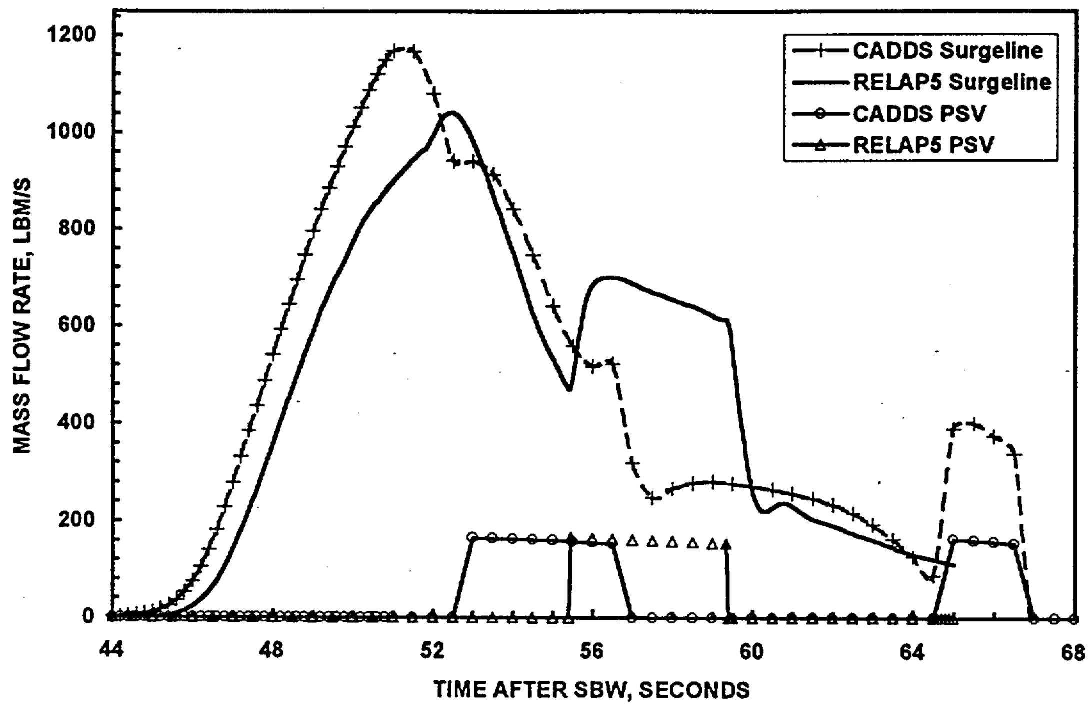


FIGURE 6-9. NEUTRON POWER PERCENTAGE OF RATED POWER FOR ARW EVENT.

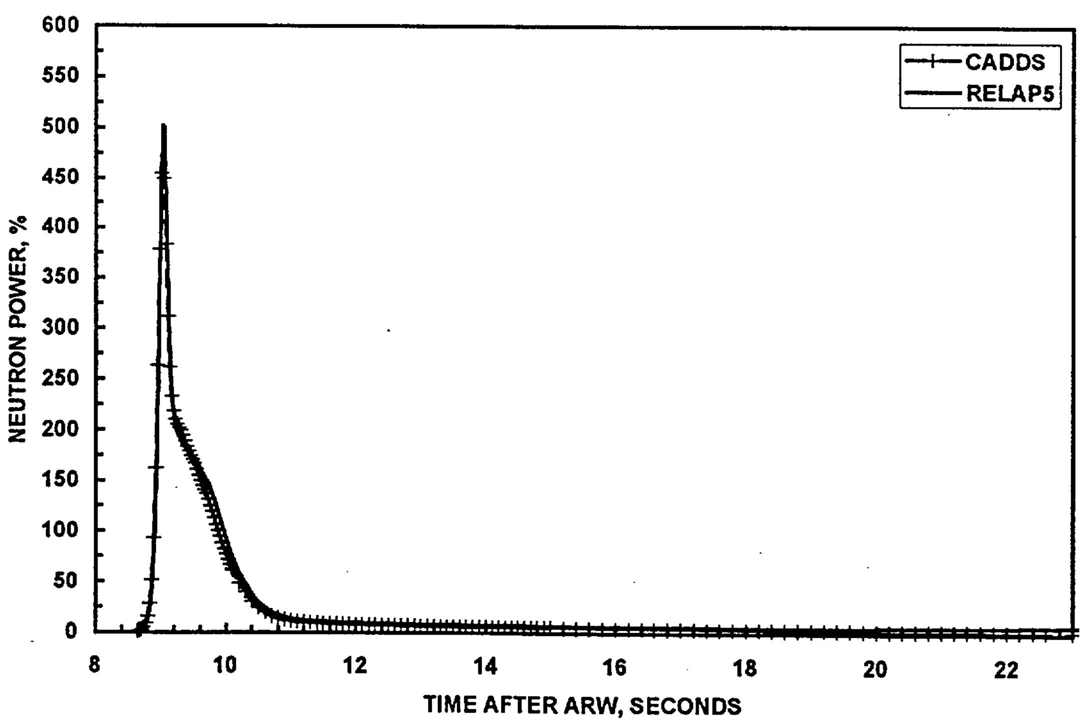


FIGURE 6-10. THERMAL POWER PERCENTAGE OF RATED POWER FOR ARW EVENT.

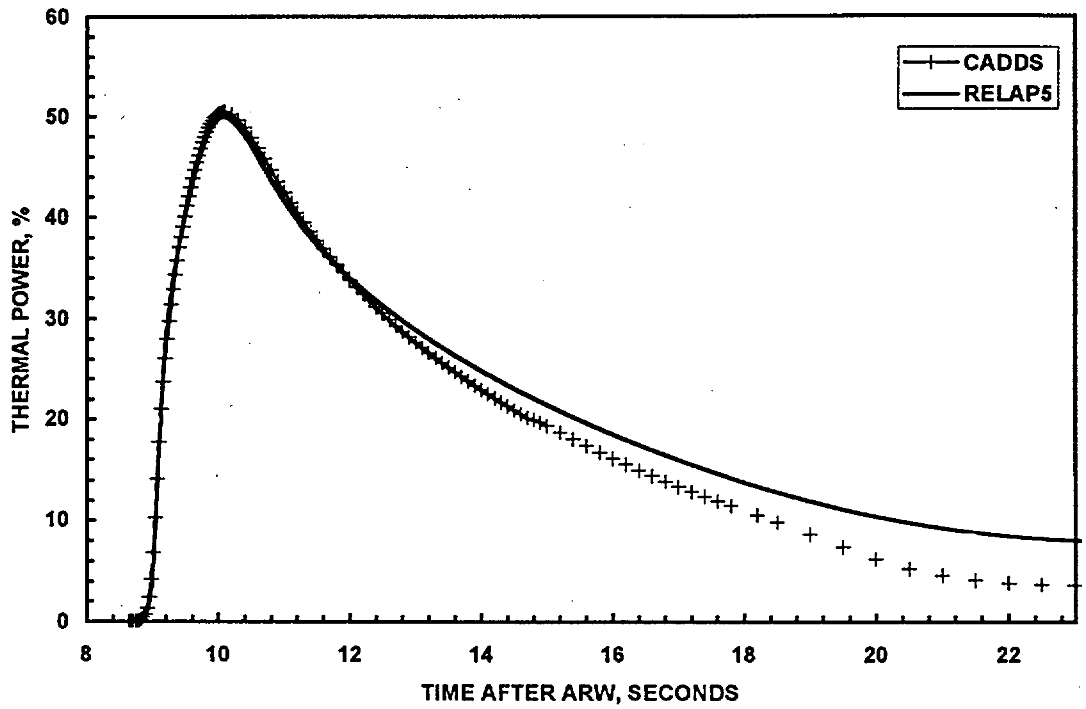


FIGURE 6-11. RCS SYSTEM PRESSURE FOR ARW EVENT.

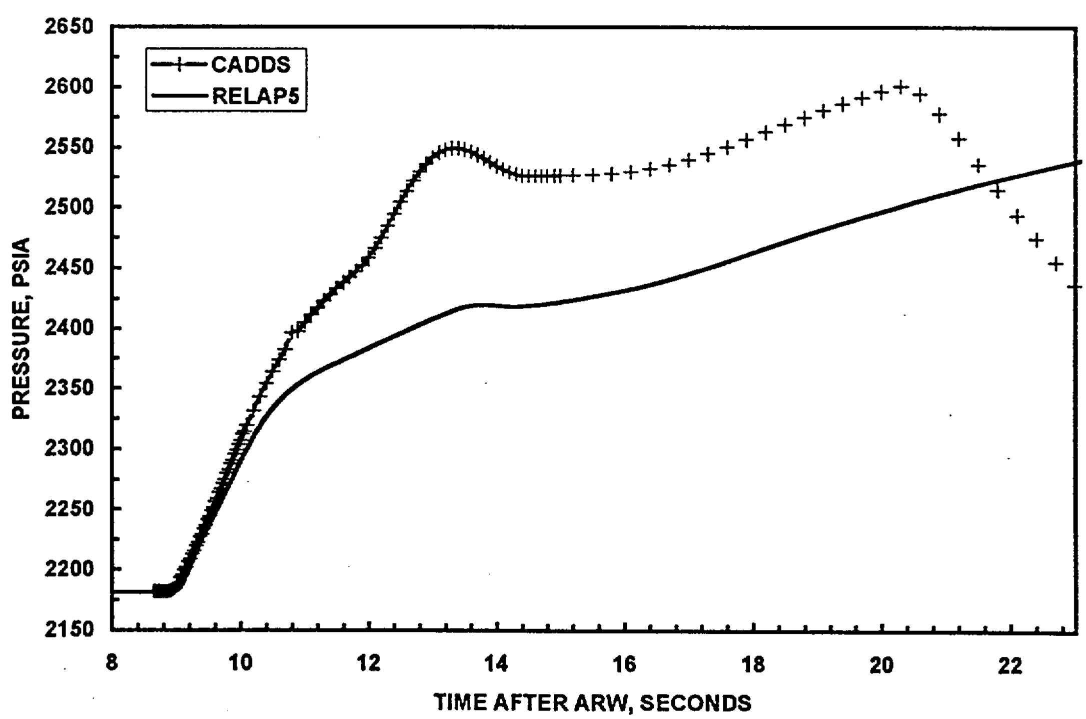


FIGURE 6-12. PRESSURIZER PRESSURE PREDICTIONS FOR ARW EVENT.

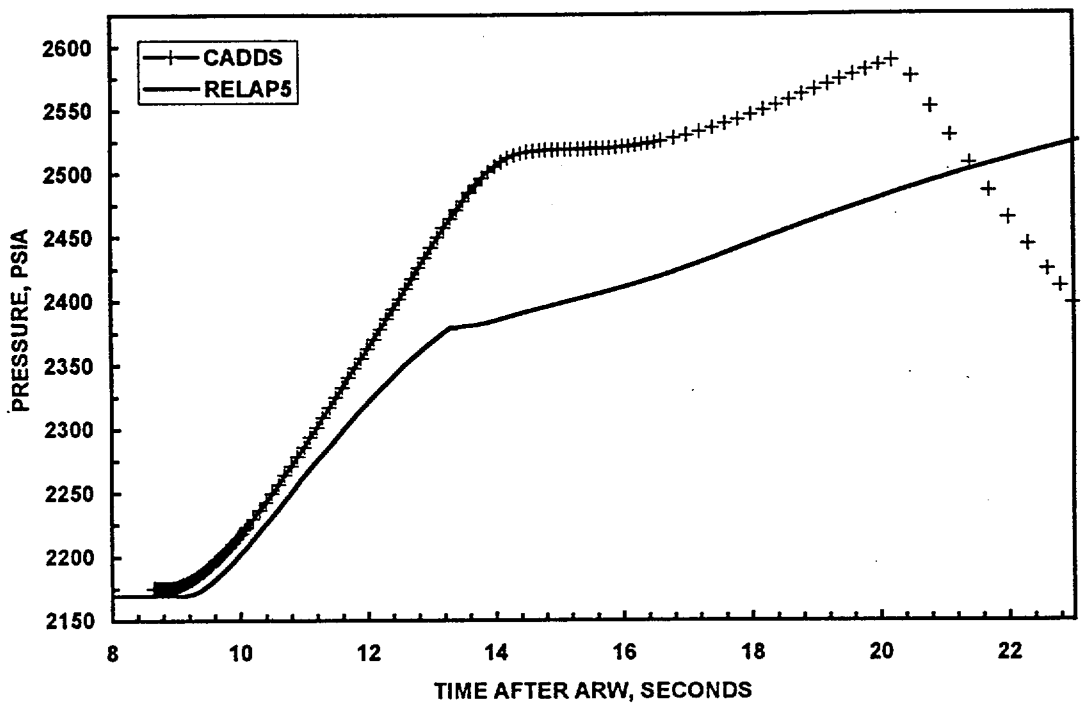


FIGURE 6-13. AVERAGE FUEL TEMPERATURE CHANGE FOR ARW EVENT.

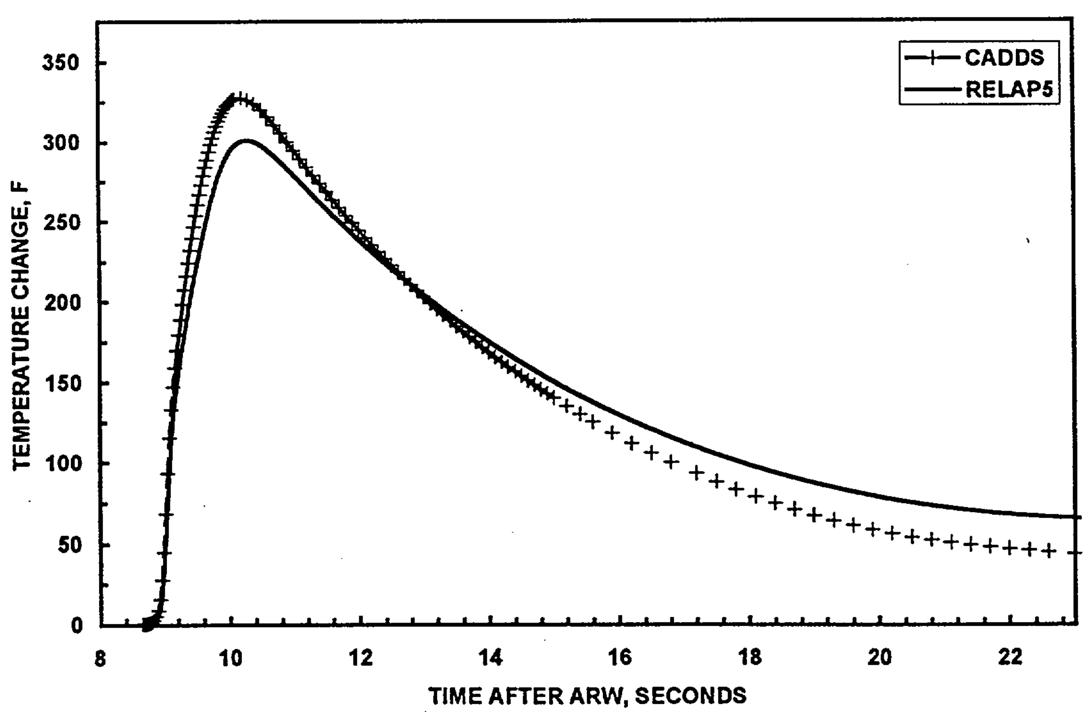


FIGURE 6-14. AVERAGE CORE MODERATOR TEMPERATURE CHANGE FOR ARW EVENT.

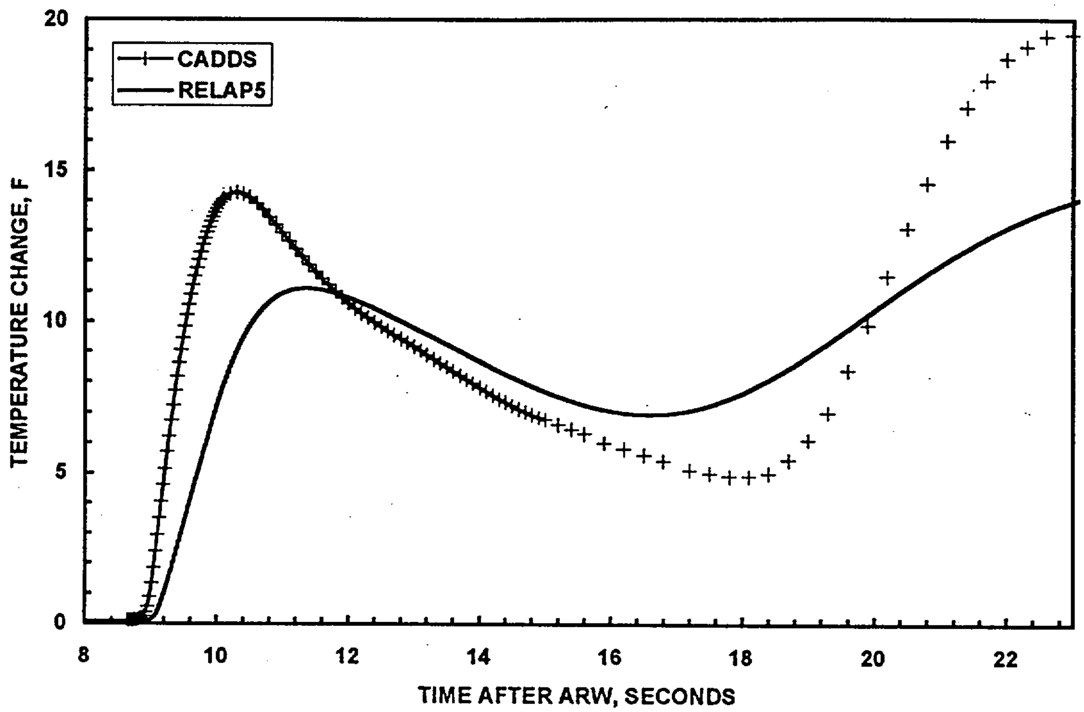


FIGURE 6-15. SURGELINE AND PRESSURIZER SAFETY VALVE MASS FLOW RATE FOR ARW EVENT.

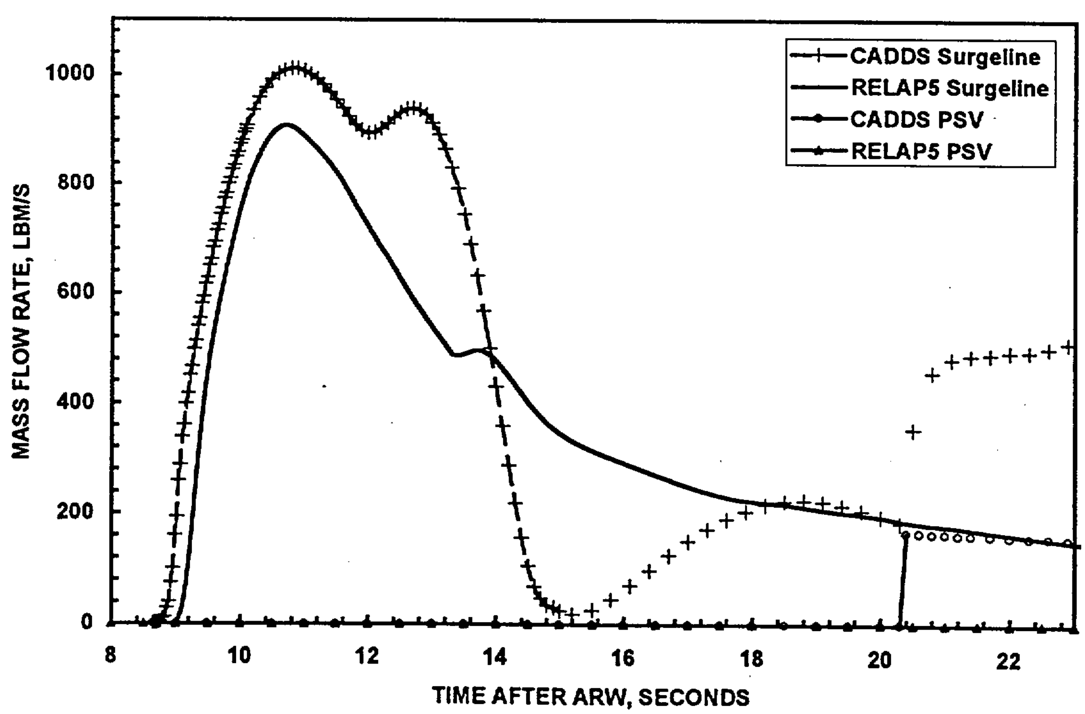


FIGURE 6-16. PEAK NEUTRON POWER VERSUS REACTIVITY INSERTION RATE FOR A ZERO POWER ROD WITHDRAWAL.

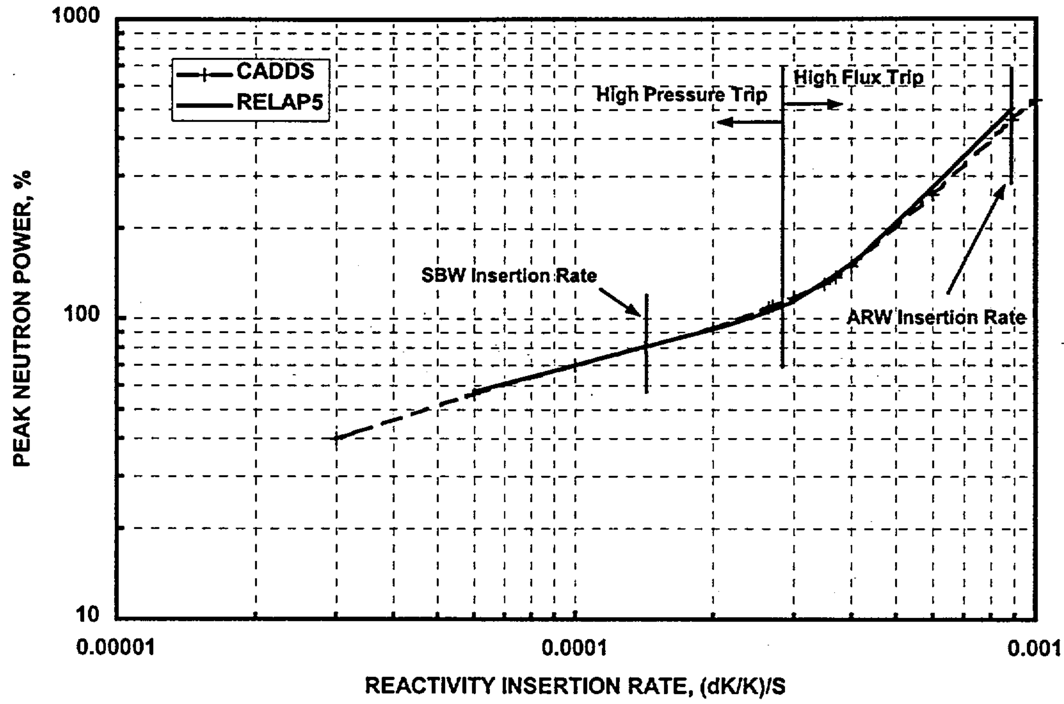


FIGURE 6-17. PEAK THERMAL POWER VERSUS REACTIVITY INSERTION RATE FOR A ZERO POWER ROD WITHDRAWAL.

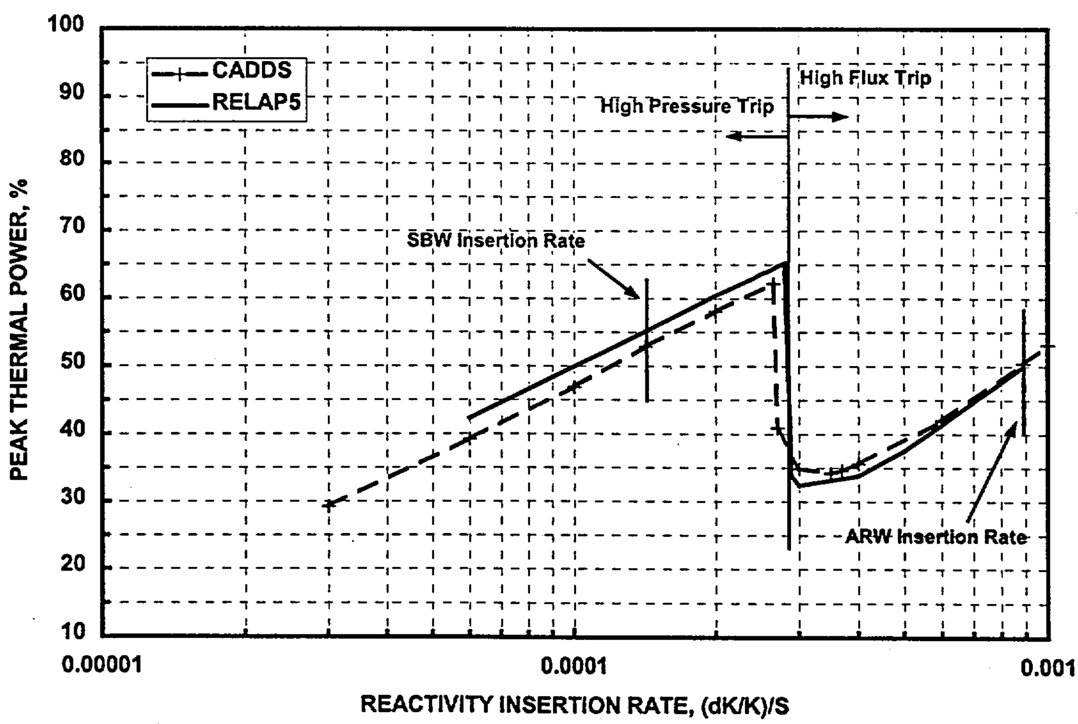


FIGURE 6-18. RELAP5/MOD2-B&W NODING ARRANGEMENT FOR THE TRAP2 MSLB BENCHMARKS.

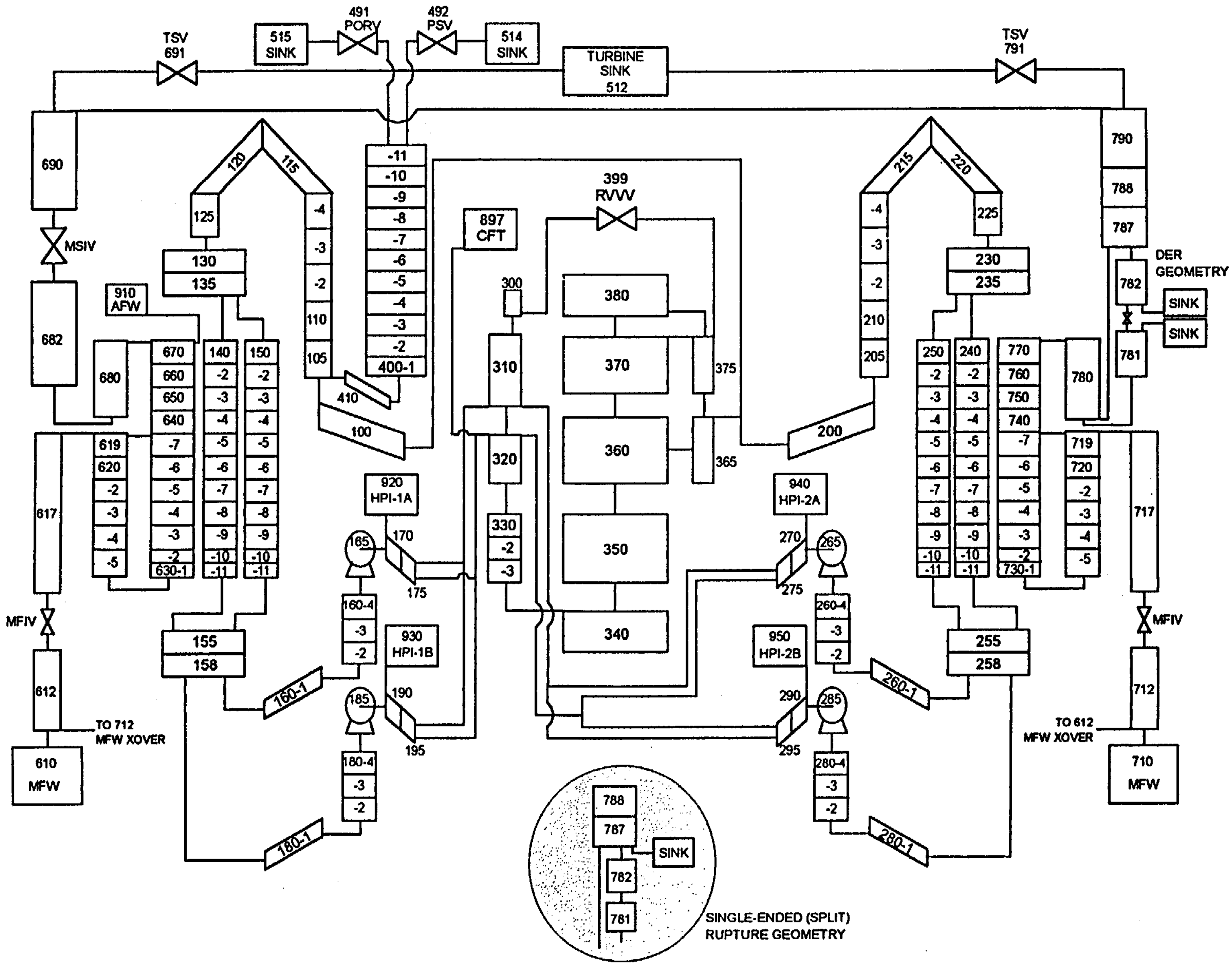


FIGURE 6-19. BREAK FLOW FROM A DER MSLB.

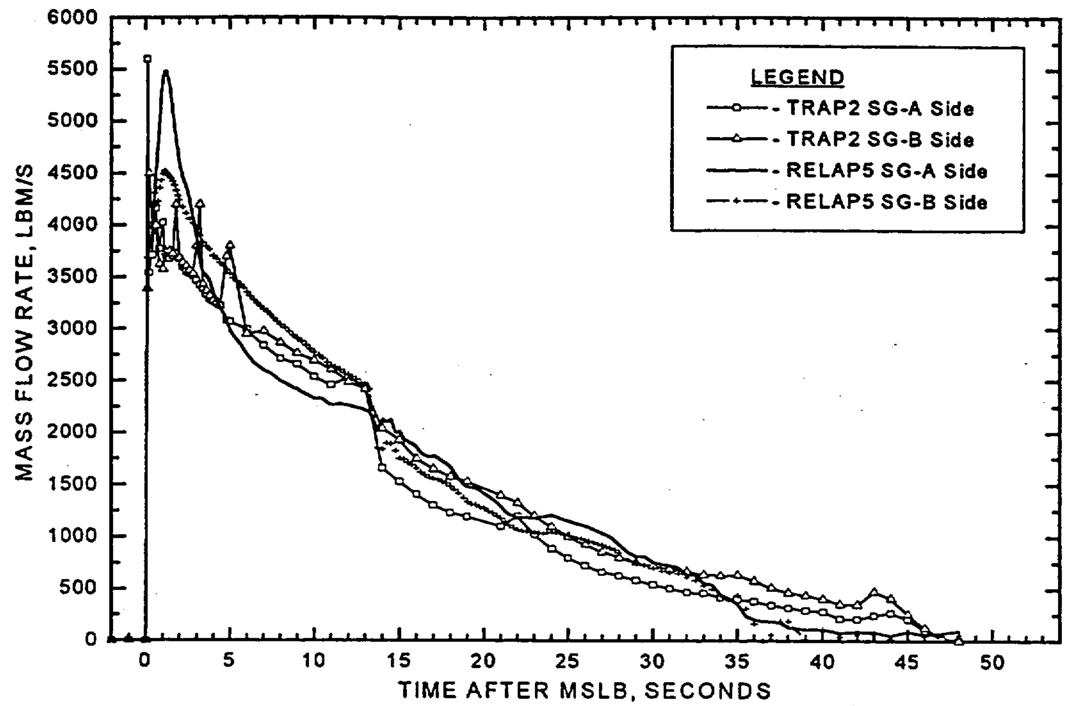


FIGURE 6-20. SECONDARY PRESSURES DURING A DER MSLB.

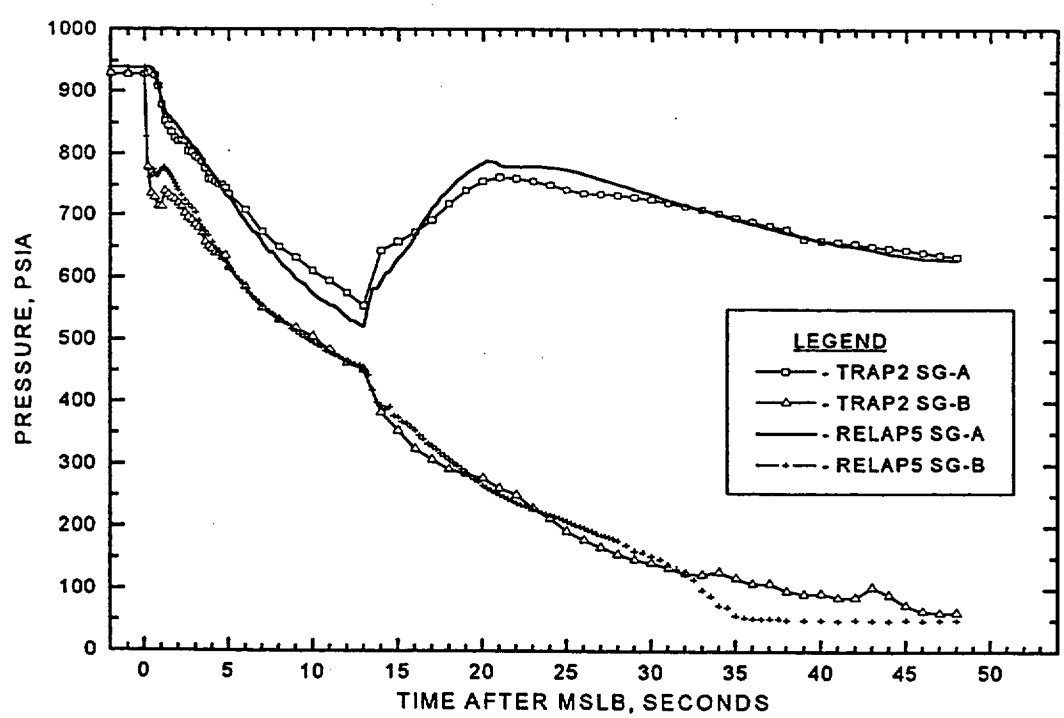


FIGURE 6-21. HOT LEG PRESSURE DURING A DER MSLB.

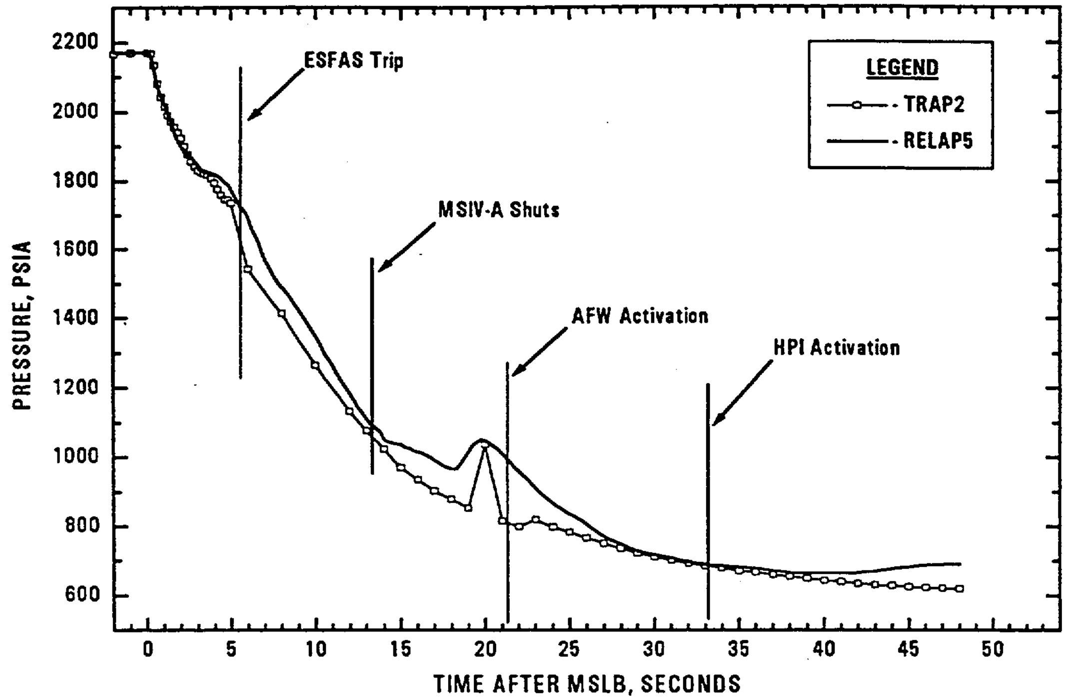


FIGURE 6-22. PRIMARY SG OUTLET TEMPERATURE DURING A DER MSLB.

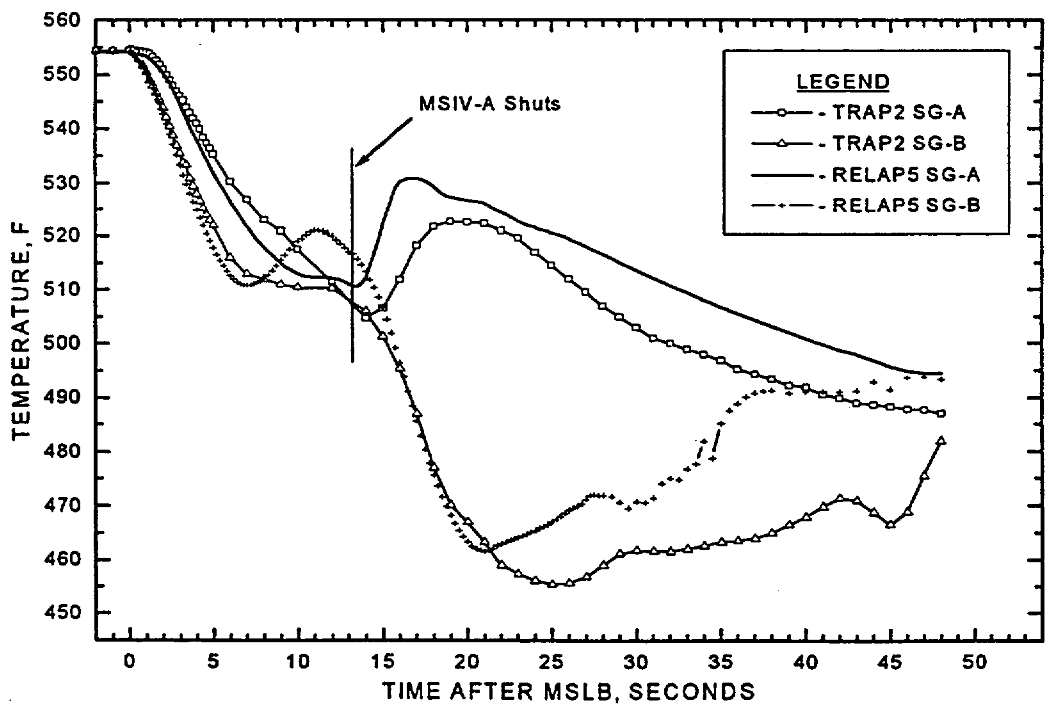


FIGURE 6-23. STEAM GENERATOR TOTAL SECONDARY MASS FOLLOWING A DER MSLB.

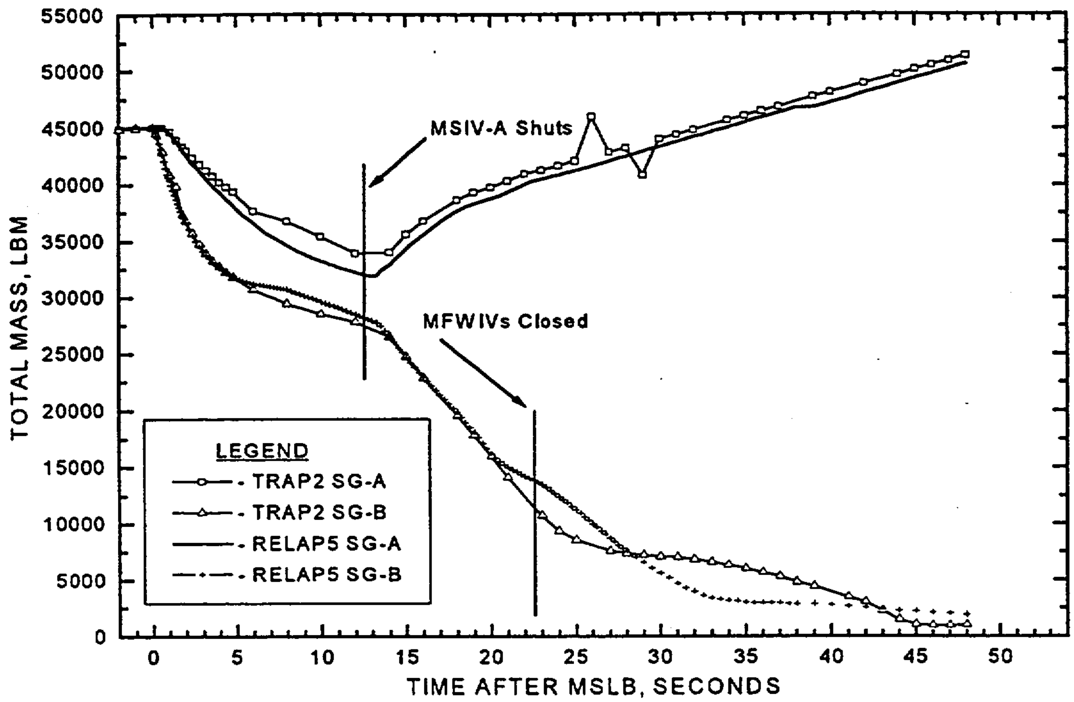


FIGURE 6-24. TOTAL REACTIVITY DURING A DER MSLB.

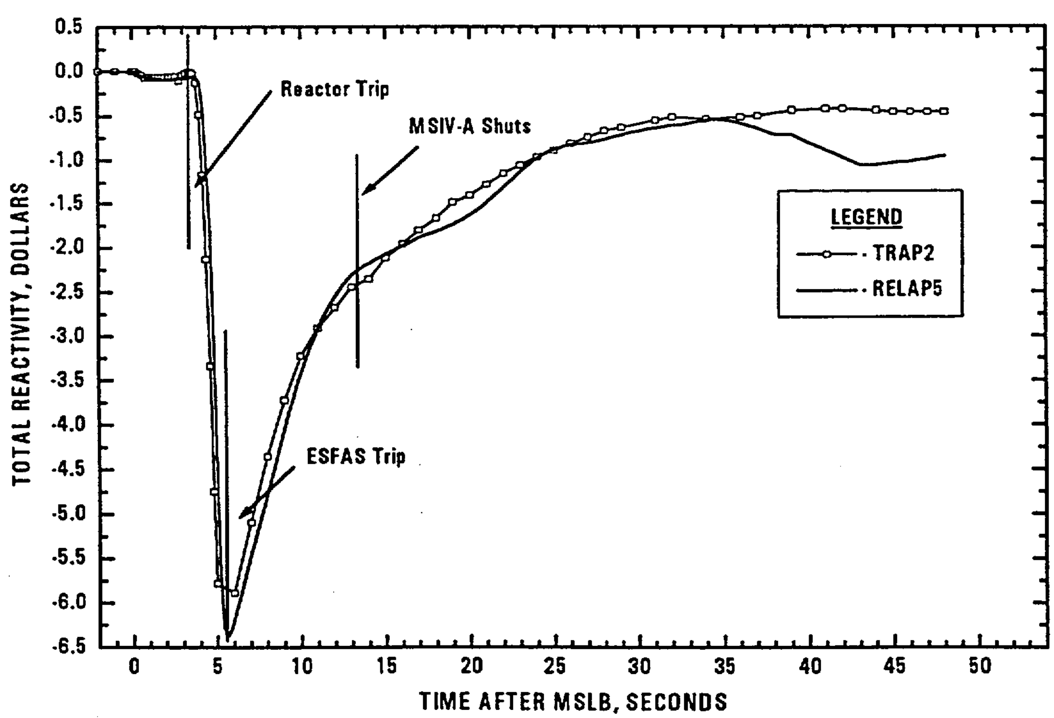


FIGURE 6-25. NORMALIZED FISSION POWER DURING A DER MSLB.

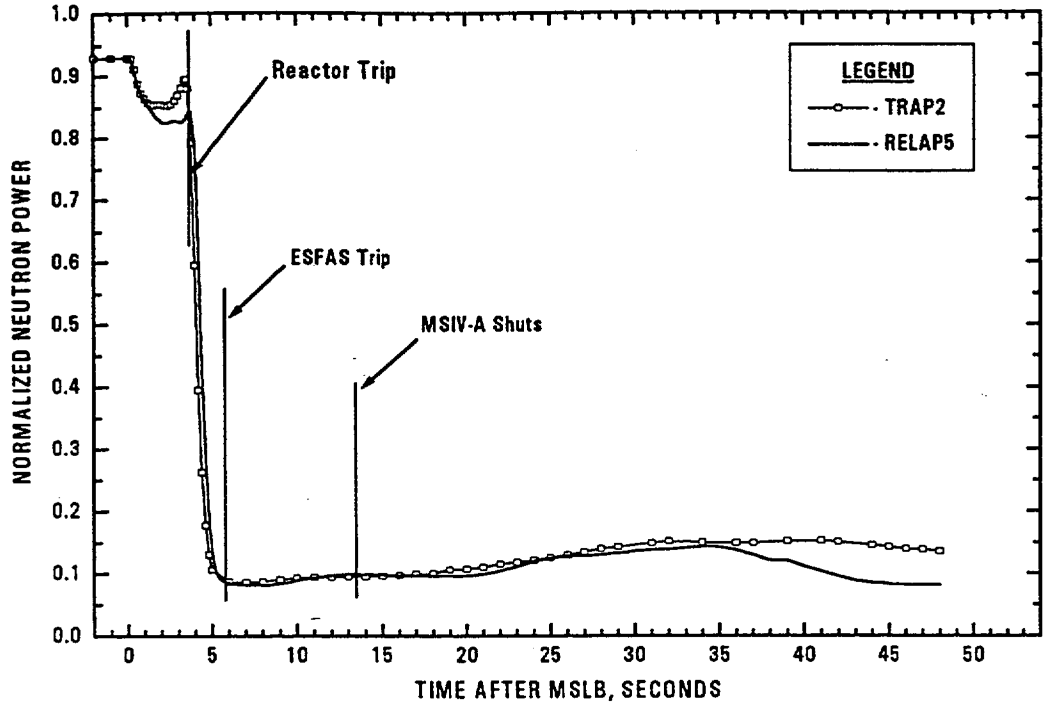


FIGURE 6-26. RCS AVERAGE TEMPERATURE DURING A DER MSLB.

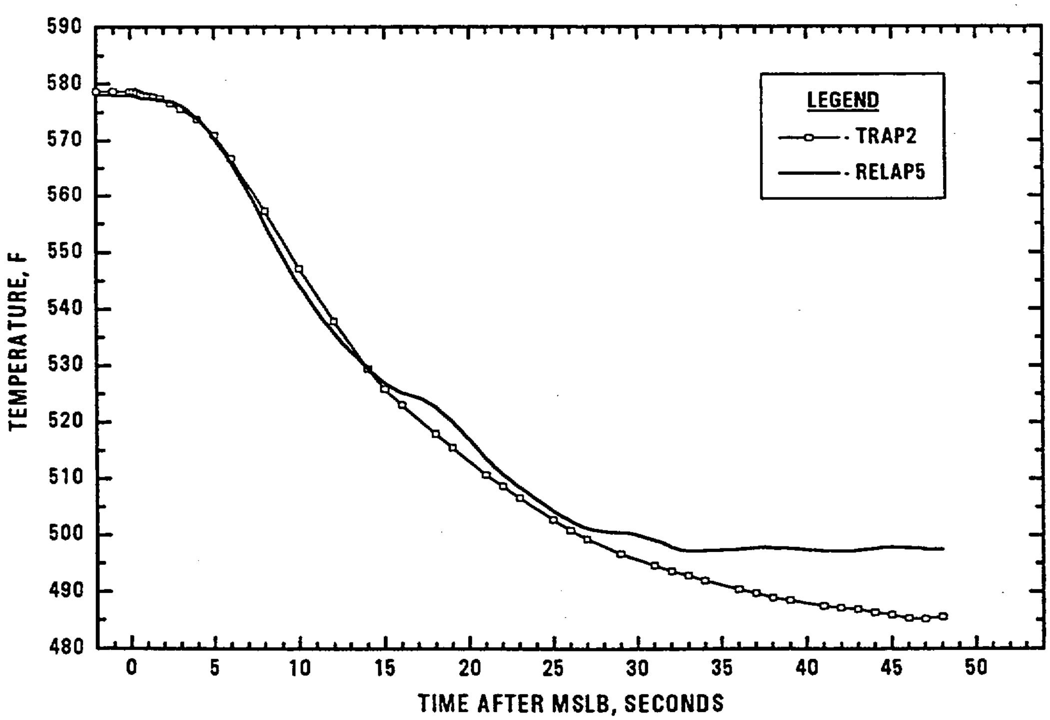


FIGURE 6-27. PRESSURIZER SURGELINE MASS FLOW DURING A DER MSLB.

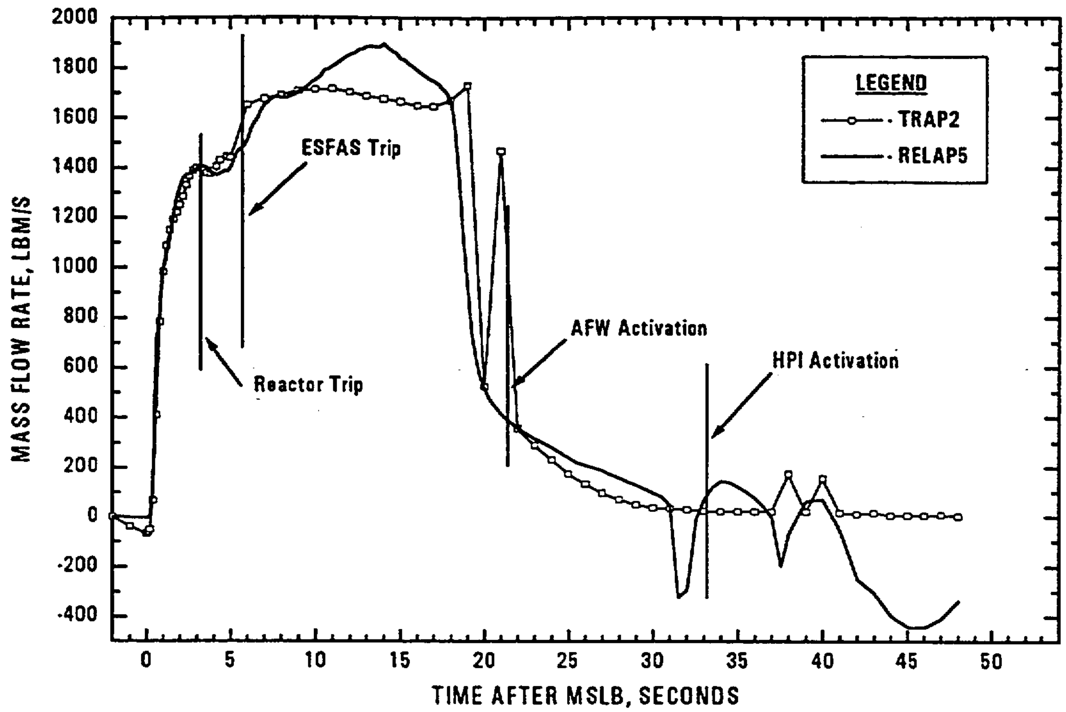


FIGURE 6-28. SG FEEDWATER MASS FLOW RATES DURING A DER MSLB.

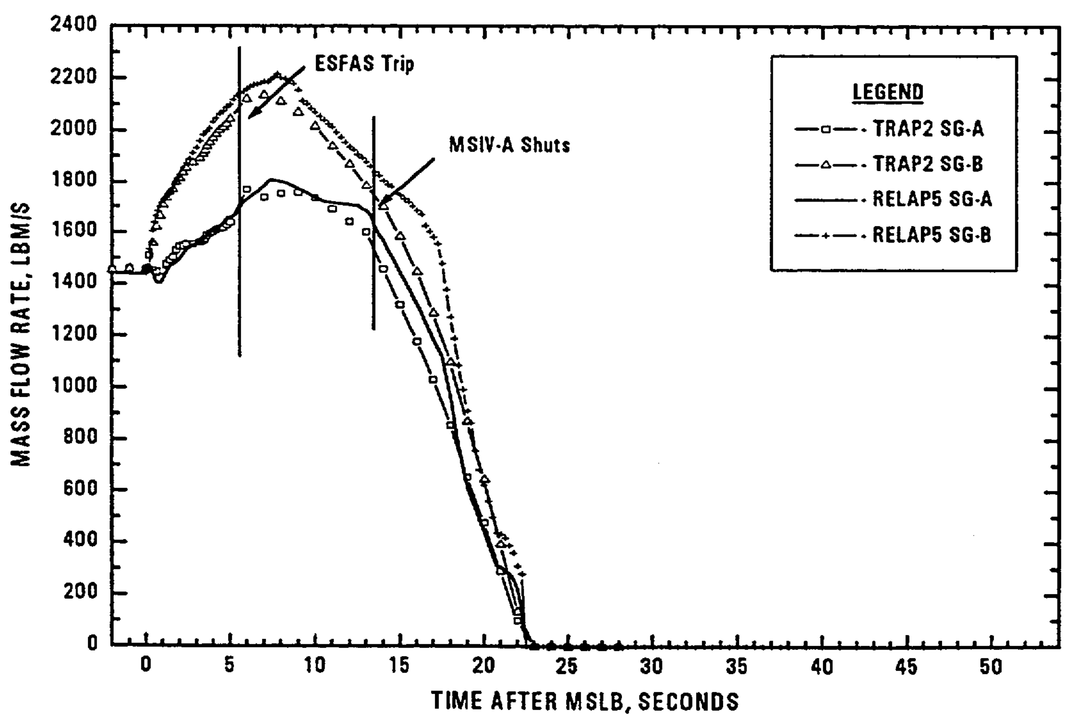


FIGURE 6-29. STEAMLINE BREAK MASS FLOW RATE FOR A SER MSLB.

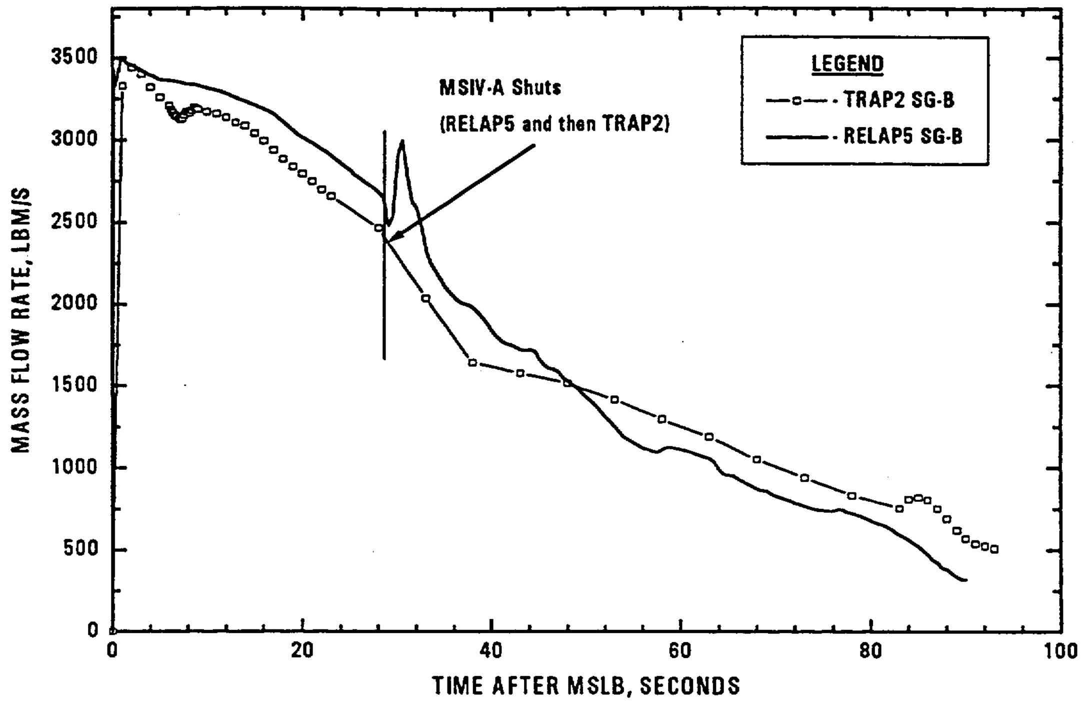


FIGURE 6-30. SECONDARY PRESSURES DURING A SER MSLB.

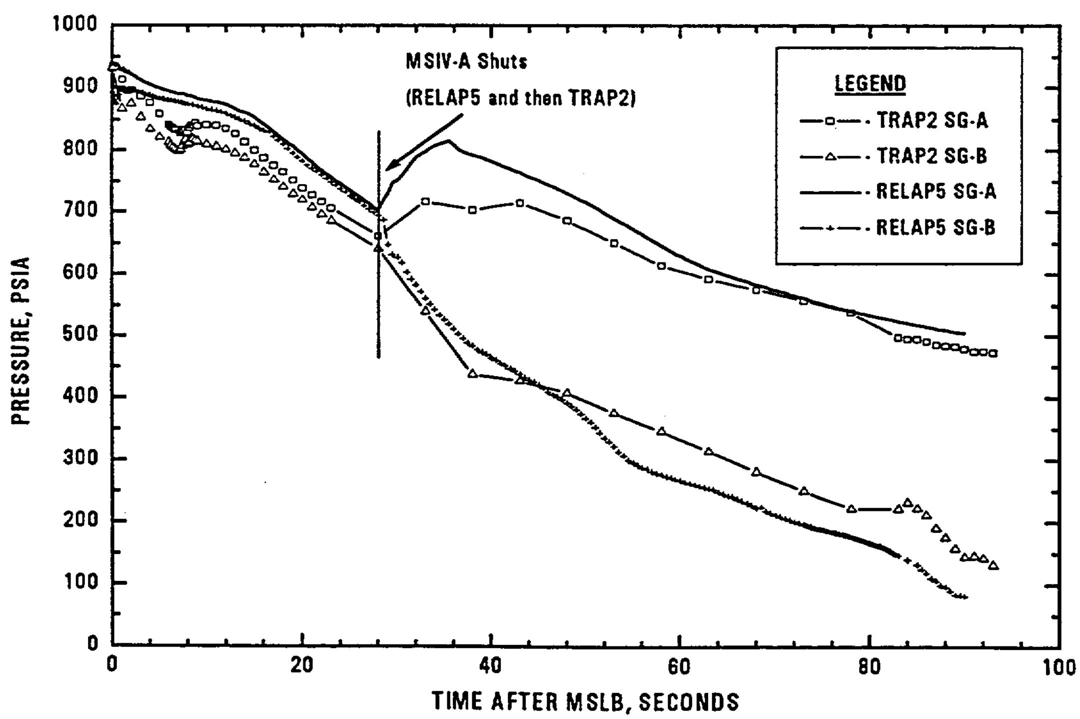


FIGURE 6-31. RCS AVERAGE TEMPERATURE DURING A SER MSLB.

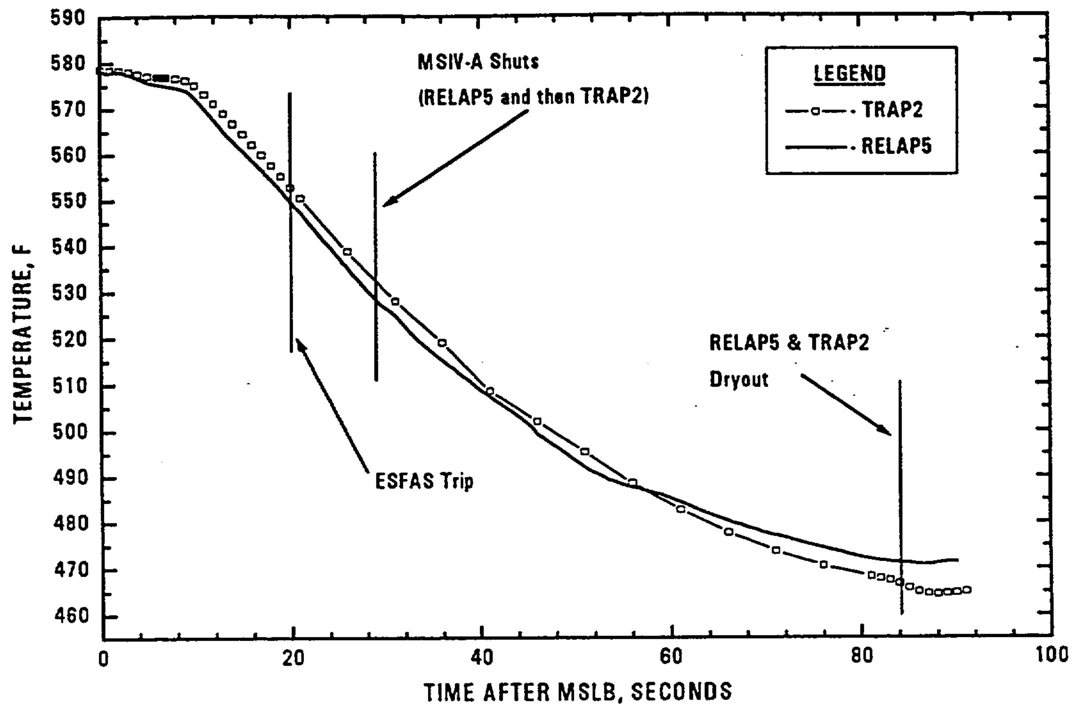


FIGURE 6-32. NORMALIZED FISSION POWER DURING A SER MSLB.

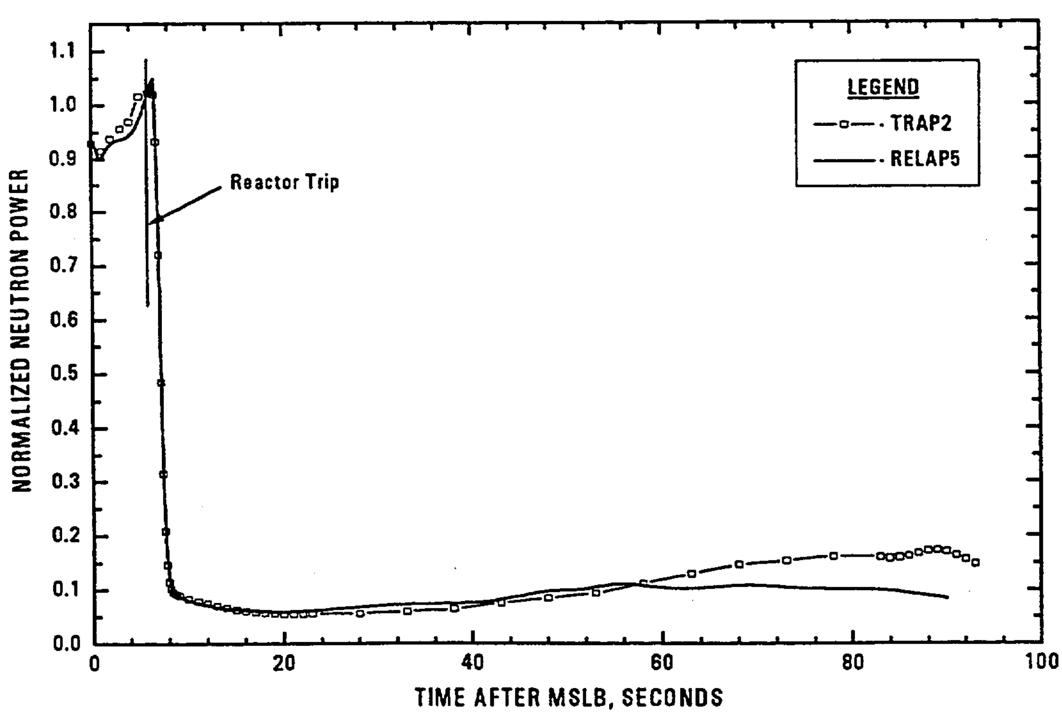


FIGURE 6-33. HOT LEG PRESSURE DURING A SER MSLB.

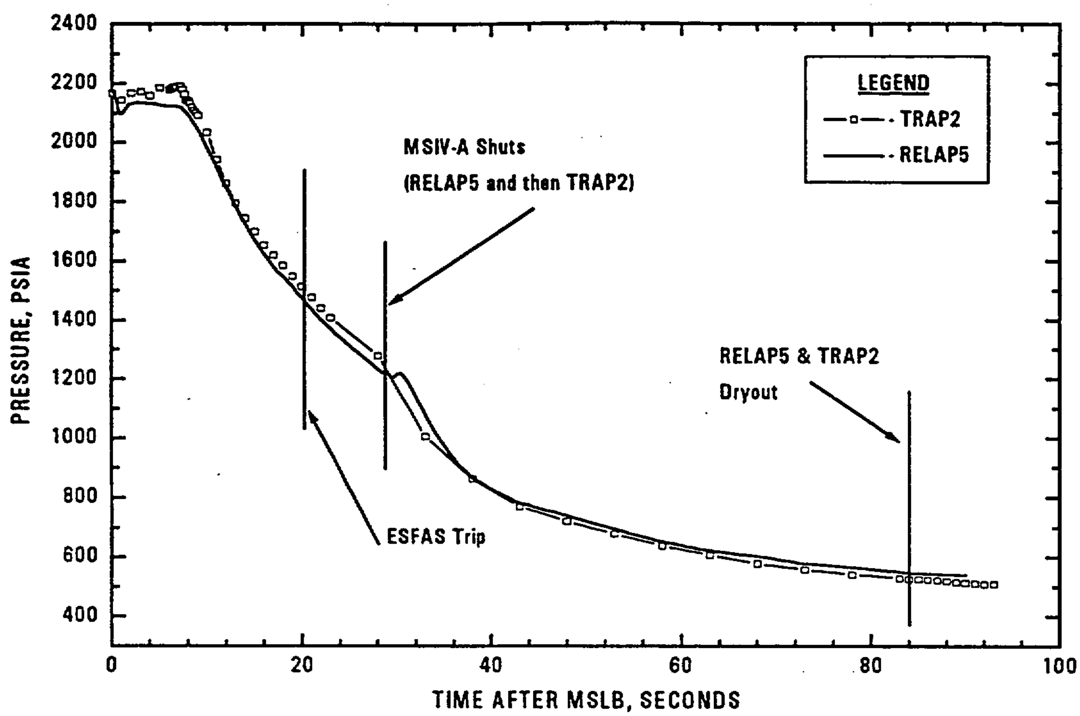


FIGURE 6-34. PRIMARY SG OUTLET TEMPERATURE DURING A SER MSLB.

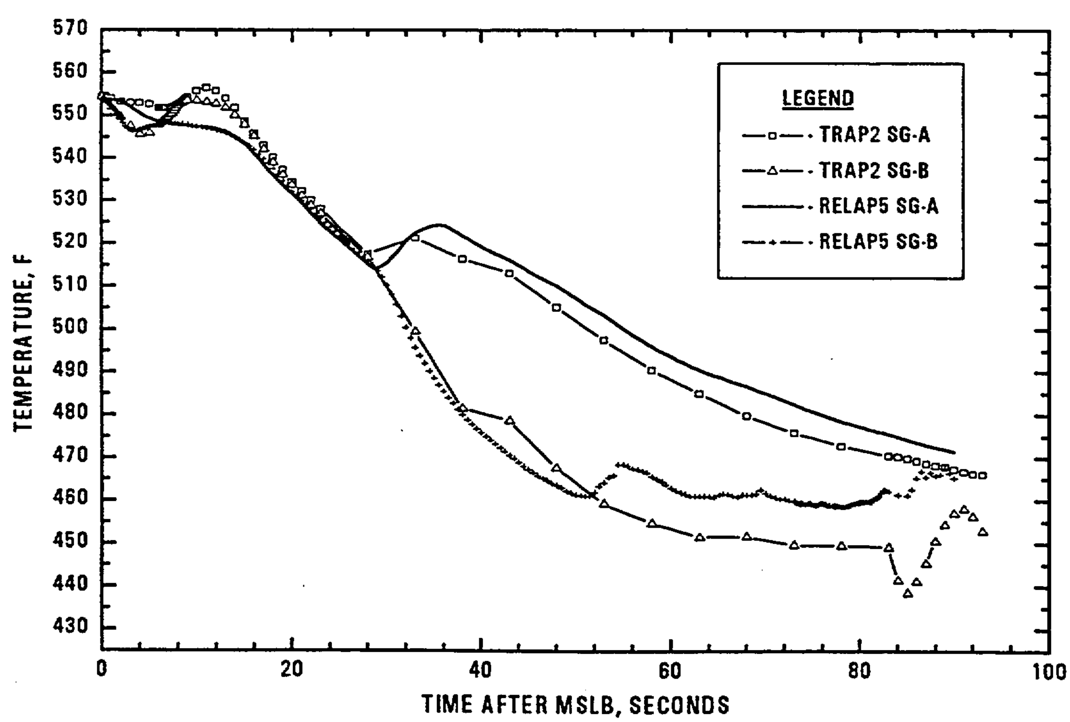


FIGURE 6-35. STEAM GENERATOR TOTAL SECONDARY MASS DURING A SER MSLB.

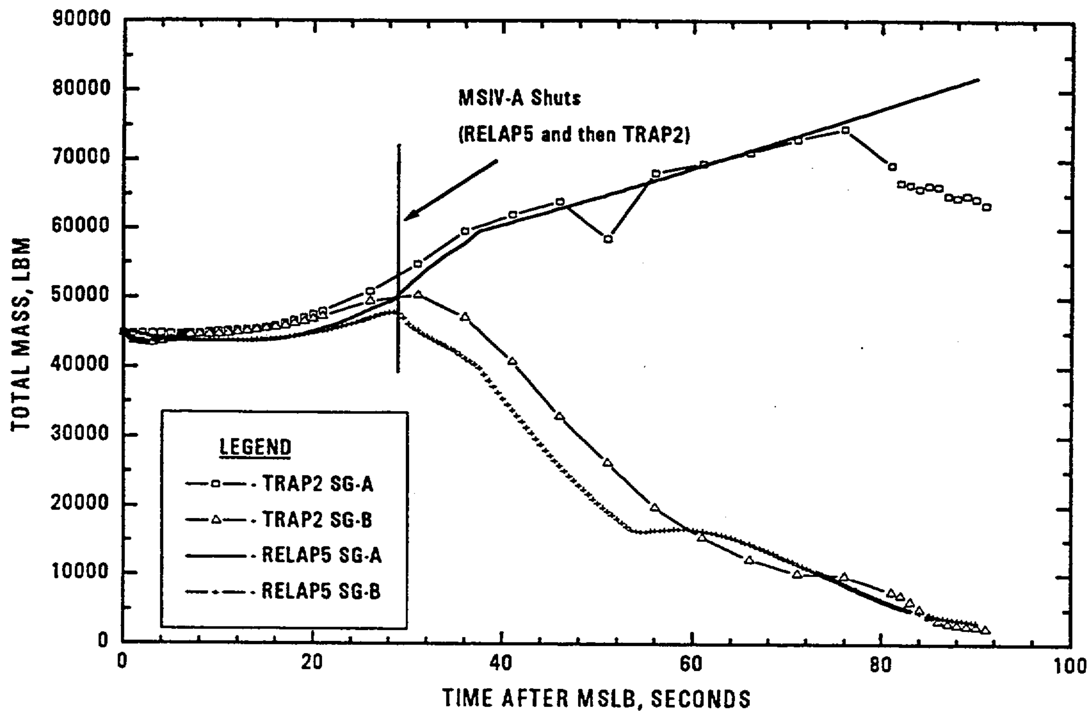


FIGURE 6-36. TOTAL REACTIVITY FEEDBACK DURING A SER MSLB.

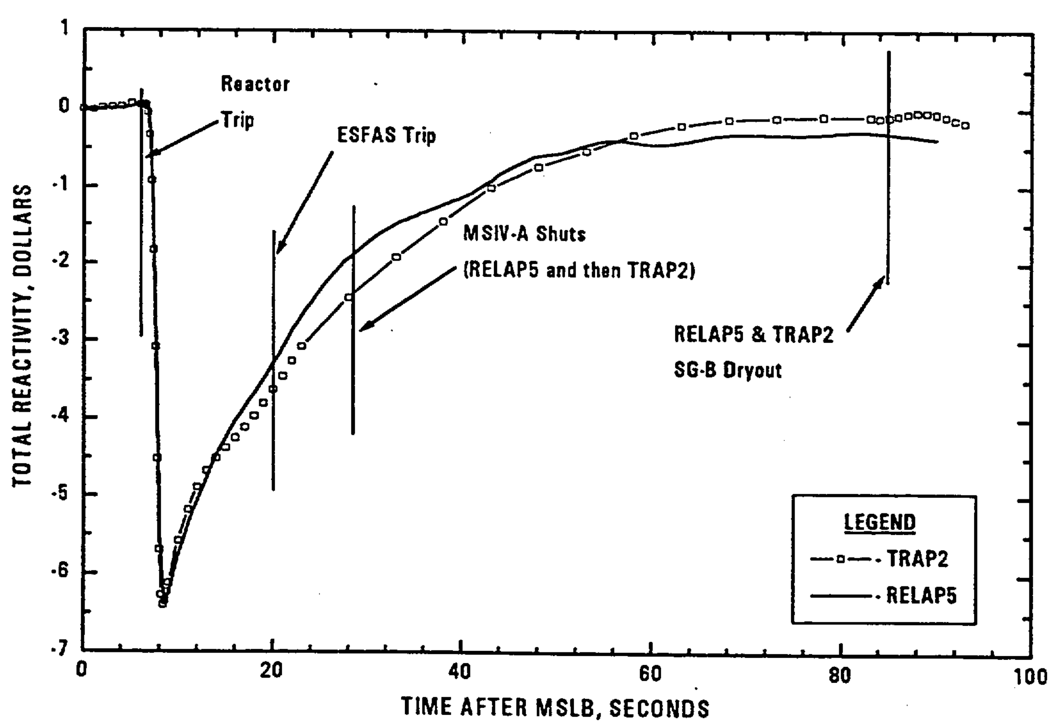


FIGURE 6-37. SURGELINE MASS FLOW RATE DURING A SER MSLB.

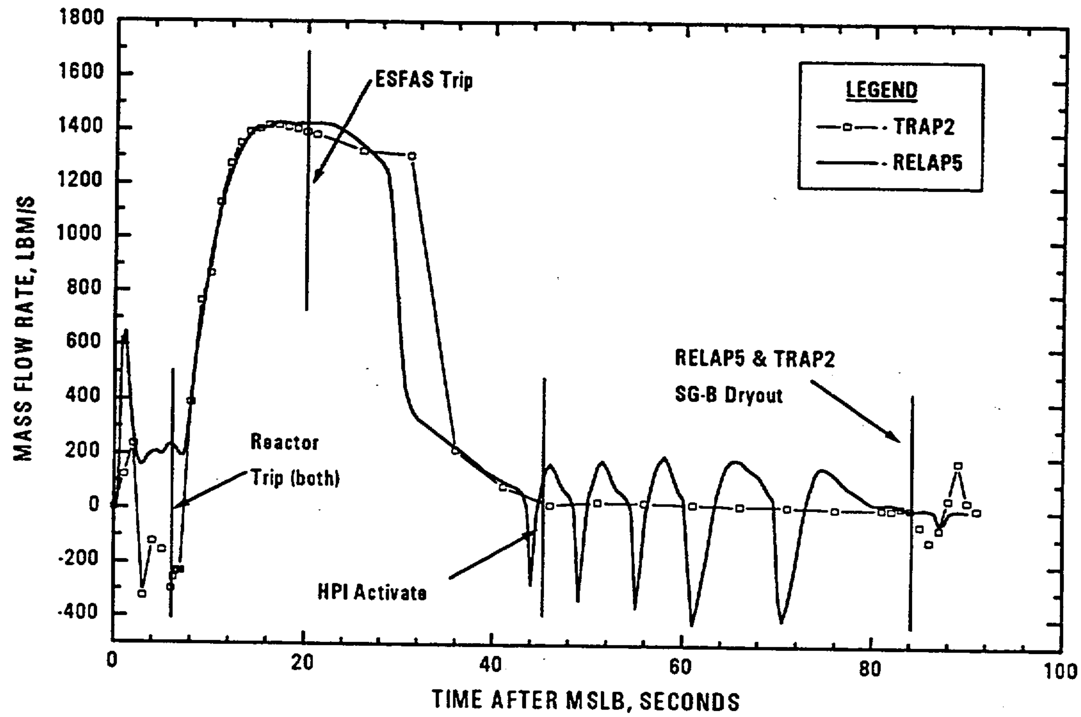
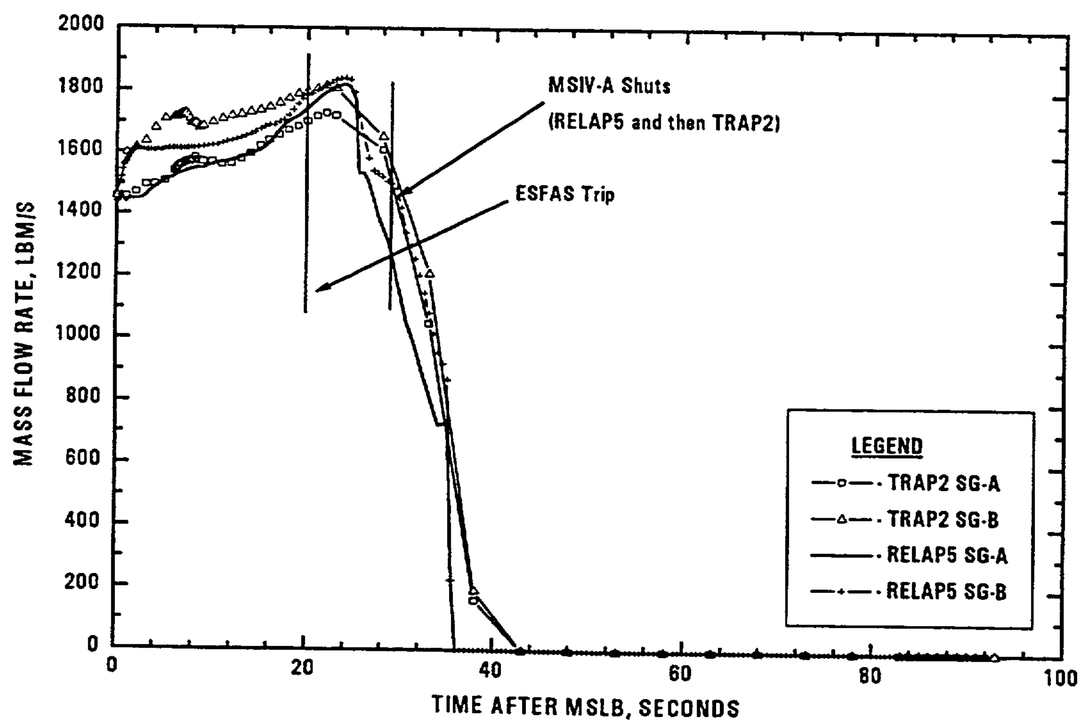


FIGURE 6-38. MAIN FEEDWATER MASS FLOW RATE DURING A SER MSLB.



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9. Babcock & Wilcox, *Thermal-Hydraulic Performance of a 19-Tube Nuclear Once-Through Steam Generator,* ARC Report 4643, June 1971.

APPENDIX

NON-LOCA ANALYSIS METHODOLOGY FOR B&W-DESIGNED PLANTS

APPENDIX

NON-LOCA ANALYSIS METHODOLOGY FOR B&W-DESIGNED PLANTS

During review of the original release of this report, the NRC requested a description of the methods that would be used to perform safety analyses with RELAP5/MOD2-B&W. Specifically, the NRC requested noding details for various accidents, specific options for constitutive models and correlations, and guidance on input assumptions if they differ from those used in the CADDS and TRAP2 analyses. The purpose of this appendix is to provide a brief compilation of this modeling information as it will be used with RELAP5/MOD2-B&W.

In general, the accident analysis methods are described in the Updated Final Safety Analysis Report for each operating plant. Framatome Technologies Group (formerly BWNT) intends to conform with the original analysis methods and licensing basis with some exceptions noted herein. The primary difference is that RELAP5/MOD2-B&W will be used rather than the original computer codes.

The plant modeling schemes that will be used for accident analyses are discussed in Section A.1. User input options for constitutive models and correlations are listed in Section A.2. Guidance on input assumptions is provided in Section A.3. Section A.4 provides guidance on modeling operator actions, and the effects of control systems are discussed in Section A.5.

It should be noted that, unless otherwise stated, the discussions herein pertain to the system analysis with RELAP5/MOD2-B&W. The detailed calculations of core response are performed with NRC-approved core neutron kinetics codes and thermal-hyrdraulics codes using NRC-approved departure from nucleate boiling correlations.

A.1 Plant Modeling in RELAP5/MOD2-B&W

Two basic plant models will be used to perform accident analyses on B&W-designed PWRs. A "large detail" model will be used for those accidents in which performance of the steam generators and/or secondary plant dominate the accident response. A "reduced detail" model can be used to analyze those transients that are dominated by core power or reactor coolant flow.

A.1.1 Large Detail Model

The large detail model is shown in Figure A-1. This model will be used for those transients in which the performance of the steam generator and/or secondary plant dominate the transient (Table A-1). The key characteristics of this model are the steam generators with eleven axial control volumes and the pressurizer composed of eleven control volumes. These representations were used in the plant and test facility benchmarks where they were shown to yield acceptable results.

With regard to the steam generator model, adequate results are obtained using the high elevation AFW model with a single region primary fluid model under forced circulation conditions. However, if reactor coolant pumps are tripped, the primary side of the tubes must be modeled with two radial regions to properly predict the heat transfer and coolant flow induced by the high elevation AFW model. The two radial regions consist of the outer annular region that represents ten percent of the tubes wetted by AFW and the inner region that represents the rest of the tubes (which are not wetted by AFW). The two-region steam generator model required for accurate AFW heat transfer predictions was used in the natural circulation test benchmark in Section 5.5.

The core is modeled with a minimum of three axial control volumes with associated core heat structures. This model was used in the plant benchmarks and provided acceptable results. Furthermore, this provides more detail than used in the CADDS and TRAP2 computer codes, which used the equivalent of a single core node. A region will be modeled to represent core bypass flow so that this coolant is not considered in calculations of moderator feedback.

Figure A-1 shows that the reactor vessel downcomer, core and core exit plenum are modeled using a single control volume for each region, consistent with the currently approved TRAP2 model. Framatome performed an investigation to verify that this model yields conservative core power predictions during steam line break. The investigation was performed using TRAP2 to determine the effects of limited thermal mixing at the core inlet on the predicted consequences of a steam line break accident on a B&W-designed PWR. It was shown that when no fluid mixing was allowed in the reactor vessel, the peak power due to subcritical multiplication was a factor of 1.07 times greater than the predicted peak power with perfect mixing. However, that same study showed that the peak core power predicted using multi-dimensional neutron kinetics and thermal-hydraulic models was forty (40) percent of the value predicted by the point kinetics core model. Consequently, it is concluded from that study that the point kinetics solution utilized by RELAP5/MOD2-B&W with perfect thermal mixing will provide a conservative prediction of core power during an asymmetric overcooling event.

Because the RELAP5/MOD2-B&W model in Figure A-1 will provide a conservatively high prediction of core power during an asymmetric overcooling event, it will provide conservative calculations of core power and of mass and energy releases from the steam generator for a steam line break accident. However, for some analyses of asymmetric overcooling accidents, core power might not be the parameter of interest. It can be postulated that for certain parameters of interest, perfect mixing in the reactor vessel would yield non-conservative results (e.g., minimum steam generator tube temperatures for calculating tube tensile loads). In these instances, a model will be used that conservatively models fluid mixing within the reactor vessel.

The limited-thermal-mixing-model is one that splits the downcomer, core, exit plenum and core bypass into equal halves (to model the line of symmetry). Fluid mixing at the core inlet and core outlet will be modeled based on the results of the Oconee plant tests.[A.1] No mixing will be modeled in the core.

The Oconee plant tests show that at least fifteen (15) percent of the reactor vessel flow is perfectly mixed in the lower plenum before it enters the core. Therefore, to ensure conservative inlet temperatures to the core, [e] of the flow will be mixed at the core inlet of the limited-thermal-mixing-model. The Oconee plant tests also show that the difference between A and B loop hot leg temperatures is only forty to fifty percent of the difference between the A and B loop cold leg temperatures. These tests were performed at beginning-of-cycle so that there was no core flux tilt associated with a negative, end-of-cycle moderator temperature coefficient. Consequently, these results indicate that at least twenty-five (25) percent of the reactor coolant is perfectly mixed before it exits the reactor vessel. Given [e] mixing at the core inlet and a small amount of mixing in the upper head region of the model, FTG has determined that [e] mixing at core exit will bound the Oconee plant test results (i.e., the hot leg temperature differential between the loops is fifty percent of the cold leg temperature differential between the loops).

The control volume arrangement depicted in Figure A-1 is the minimum level of detail for transients that require this model. FTG reserves the option to increase the number of control volumes and related heat structures in any particular region of the model. Similarly, although not depicted in Figure A-1, this model will often contain a detailed model of the feedwater system including feedwater pumps, control valves, isolation valves, condensate pumps, feedwater heaters, etc. The RELAP5/MOD2-B&W models of the feedwater and steam systems are superior to those used in TRAP2 because a relatively small limit on the number of control volumes in TRAP2 (<100) allowed only simple models of the secondary systems. The feedwater system and steam line arrangements are plant specific.

A.1.2 Reduced Detail Model

The reduced detail model is shown in Figure A-2. This model will be used for those transients in which the core response dominates the accident, predominantly reactivity events (Table A-1). The only difference between this model and that in Figure A-1 is in the steam generator. The computational speed of the model is increased by deleting the entire secondary system and by reducing the entire steam generator tube region to a single control volume and associated heat structure. Steam generator heat removal is set by a heat demand on the outside boundary of the tubing heat structure in each steam generator.

It should be noted that the control volume arrangement depicted in Figure A-2 is the minimum level of detail for transients that require this model. FTG reserves the option to increase the number of control volumes and related heat structures in any particular region of the model.

A.2 User Input Options For Code Models and Correlations

There are a number of code models and correlations that are required to predict properly the response of the B&W-designed PWR to non-LOCA accidents. Each model or correlation is listed herein with the specific inputs that will be used for accident analyses.

A.2.1 Interface Drag

All of the OTSG models in this report utilized the BWNT slug-drag model on the secondary side. The default inputs were used. These are listed in Reference A.2 as:

[c, e]

These values will be used in all accident analyses.

A.2.2 Heat Transfer Coefficients

A number of heat transfer coefficients are used to calculate properly the heat transfer from the secondary surface of the OTSG tubing to the secondary coolant. The following correlations and user inputs were used in the code benchmarks performed for this topical report. These same correlations and inputs will be used in accident analyses of B&W-designed PWRs.

Except as modified or clarified below, the default RELAP5/MOD2 heat transfer package will be used for pre-CHF, post-CHF and single-phase steam heat transfer.

A.2.2.1 *Nucleate Boiling*

The Chen nucleate boiling correlation is used to calculate heat transfer from the tube prior to dryout. Because the Chen nucleate boiling coefficient becomes unrealistically large as void fraction approaches 1.0, a multiplier is applied to the Chen correlation which reduces the correlation value **[c, e]** over the void fraction range of α_{gr} to 1.0. The user input value for α_{gr} is **[c,e]** for use with the eleven axial node model used in safety analyses. If more axial control volumes are used in the OTSG model, a **[e]** α_{gr} value might be required to properly predict the boiling length. Any change to α_{gr} to values other than **[c,e]** will be justified by comparison with the 19-tube model OTSG boiling data in Section 4.3

A.2.2.2 *Critical Heat Flux*

As described in Reference A.2 and demonstrated in Section 4.3, the Biasi-Zuber critical heat flux (CHF) correlation available in RELAP5/MOD2 underpredicts the dryout location in the OTSG below eighty percent power. It has been demonstrated that the **[c, e]** CHF correlation predicts the dryout location more accurately. Consequently, the **[c, e]** CHF correlation will be used for all OTSG applications.

A.2.2.3 *Post- CHF Heat Transfer*

Post-CHF heat transfer to superheat the secondary steam only accounts for about five percent of the heat transfer in an OTSG. The Bromley film boiling correlation and the Dittus-Boelter correlation for single-phase steam do an adequate job of raising the steam superheat above the minimum design value (35 F). For most applications the post-CHF heat transfer using these correlations is adequate. However, the appropriate calculation of mass and energy releases from an OTSG requires an accurate prediction of steam superheat (50 to 60 F in the plant). A comparison of the 19-tube model OTSG steady-state data with RELAP5/MOD2-B&W predictions indicated that the film boiling region in the code calculation was too large and the heat transfer to single-phase steam was too low. These two effects degrade the steam superheat prediction by **[c]** at full power.

Nineteen-tube OTSG test data shows that the film boiling region is only a **[c, d, e]** long at full power. This appears to be due in part to **[c, d, e]**

[c, e] directly above the dryout elevation. [c, e]

[c, e]

The underprediction of the heat transfer to single-phase steam is cause by two effects. First, the Dittus-Boelter single-phase heat transfer correlation was developed for flow inside tubes. It underpredicts the heat transfer coefficient for parallel flow in heated bundles. Second, the relatively large control volumes used in the OTSG model cause a rather large discretization error in the primary-to-secondary temperature difference used in the calculations.

The form of the Dittus-Boelter correlation in RELAP5/MOD2-B&W is:

$$h_{DB} = C\,Re^{0.8}\,Pr^{0.4}$$

where

C = 0.023
Re = Reynolds Number
Pr = Prandtl Number

A generally accepted adjustment to the correlation for parallel flow in a heated bundle with a triangular pitch is:[A.3]

$$C = 0.026(s/D) - 0.006$$

where

s = tube pitch
D = tube diameter

For an OTSG, which has a tube diameter of 5/8 inches and a pitch of 7/8 inches, the value of C is 0.0304. This is a factor of 1.32 greater than the default value of C in RELAP5/MOD2-B&W.

The relatively large control volumes used in the OTSG model cause an excessive discretization error in the primary-to-secondary temperature difference used in the calculations. In other words, because the heat transfer to or from an entire heat structure is applied to a control volume, the control volume fluid temperatures used in the heat transfer calculations are equivalent to node exit temperatures. Because there is such a large change in temperature across a control volume, this difference between node average and node exit temperature leads to a large difference in the primary-to-secondary heat transfer. As the heat transfer coefficient is increased to increase the steam exit superheat, the discretization problem is exacerbated by further decreasing the primary-to-secondary temperature differential. It is

[c, e]

This value will be used for all accident analyses for which steam exit temperature is a critical parameter.

If more than eleven axial nodes are used to represent the OTSG in the future, a smaller multiplier will be required because the discretization effect will be reduced. In this case, a change to the user-input multiplier for the Dittus-Boelter steam heat transfer coefficient will be justified by comparison of superheat predictions to 19-tube OTSG data and full scale plant data.

A.2.3 Pressurizer Surge Line

All calculations performed for this report that included a pressurizer modeled the surge line-to-hot leg connection as a cross-flow junction because flow from/to the surge line is orthogonal to the hot leg fluid flow. Consequently, this type of junction will be used in all accident analyses.

A.2.4 Critical Flow Models for Secondary Pipe Ruptures

The RELAP5/MOD2-B&W main steam line break benchmarks of TRAP2 (see Section 6.2) used the Moody critical flow model with a discharge coefficient of 1.0. The TRAP2 computer code also uses the Moody critical flow model. It is generally accepted that the Moody critical flow model provides conservative mass flow rates for single-phase steam and two-phase mixtures. Consequently, the Moody critical flow model with a discharge coefficient of 1.0 will be used for calculating secondary system pipe ruptures.

A.3 Guidance on Input Assumptions

There are a significant number of input assumptions for any safety analysis. These include initial conditions, boundary conditions, reactivity coefficients, effects of control systems, single failure assumptions and loss-of-offsite power. The assumptions used for each analysis and the methodology Babcock & Wilcox employed in the analysis is described in the Updated Final Safety Analysis Report (UFSAR) for each plant. Many of these issues are discussed herein. Unless otherwise stated in this section, the original licensing basis of the plant being modeled takes precedence.

A.3.1 Initial Conditions

Except as noted, the RELAP5/MOD2-B&W initial conditions for non-LOCA accidents correspond to those used in the original safety analyses documented in the UFSAR for each plant. The initial conditions for power, reactor coolant pressure, reactor coolant temperature, reactor coolant flow, steam generator mass inventory, and pressurizer level are discussed below.

A.3.1.1 Power

The initial power level for a particular safety analysis will be consistent with the UFSAR and the plant rated power level. A heat balance uncertainty of two percent will be included in all calculations except steam line break. For steam line break, nominal core power will be used to minimize heat input to the reactor coolant system, but steam generator mass inventory will be increased (maximize heat removal from the reactor coolant system) to account for the heat balance uncertainty. For example, the core power will be set to 100 percent and the steam generator mass inventory will be set to the value expected for 102 percent power. This will provide a conservative steam line break analysis as compared with a simulation from 102 percent core power with a mass inventory for 102 percent power.

A.3.1.2 Pressure

Consistent with the original FSAR safety analyses, the initial reactor coolant pressure in all RELAP5/MOD2-B&W non-LOCA analyses will be set to the nominal value. Pressure measurement uncertainties are applied to safety and relief valve lift setpoints, reactor protection system trip setpoints, and engineered safety feature actuation system setpoints (see Section A.3.2) to yield conservative predictions of system response. Calculations of hot channel departure from nucleate boiling ratio (DNBR) are performed with other NRC-approved subchannel computer codes. Those calculations include pressure measurement uncertainties to ensure a conservative result.

A.3.1.3 Temperature

Consistent with the original FSAR safety analyses, the initial reactor coolant average temperature in all RELAP5/MOD2-B&W analyses will be set to the nominal value for the given power level. Calculations of hot channel departure from nucleate boiling ratio (DNBR) are performed with other NRC-approved subchannel computer codes. Those calculations include temperature measurement uncertainties to ensure a conservative result.

A.3.1.4 Reactor Coolant System Flow

Initial reactor coolant system flow will be set to minimum thermal design flow. When hot channel minimum DNBR is evaluated using an NRC-approved subchannel computer code, the reactor coolant flow versus time will be normalized to a limiting value consistent with the approved methodology being employed. Flow measurement uncertainties will also be addressed in the DNBR calculations in a manner consistent with the NRC-approved methodology that is employed.

A.3.1.5 Steam Generator Mass Inventory

With the exception of steam line break, loss of feedwater and feedwater line break accidents, the calculated core power and system responses during postulated accidents are not sensitive to initial steam generator mass inventory. Consequently, nominal mass inventory will be used in all accident analyses with the exception of the events noted above. At full power, nominal inventory is obtained for a steam generator operate range level of 65 to 80 percent.

A conservative calculation of steam line break requires a conservatively large initial mass inventory. Consequently, the initial steam generator mass inventory for steam line break will be set to a conservatively high value that bounds plant operation. Many of the plant UFSARs list the equation for initial mass inventory for SLB as

$$I = 41{,}200\ P + 13{,}800\ \text{lbm}$$

where

I = mass inventory, lbm
P = power level as a fraction of rated

For 102 percent of rated power the maximum mass inventory using this equation is 55,824 lbm. This is equivalent to an operate range level of approximately **[c,e]** percent.

Loss of main feedwater (LOFW) calculations performed to determine auxiliary feedwater flow requirements are performed using a conservatively low inventory to yield bounding results. Therefore, an initial mass inventory will be used that corresponds to an operate range level that is less than the plant operating value.

Feedwater line break (FLB) is not part of the licensing basis of most B&W-designed PWRs. However, any FLB analysis performed to assess the reactor coolant system response will use a minimum initial steam generator mass inventory, just as is used for LOFW. This will provide the maximum reactor coolant temperature and pressure. Mass and energy releases to the containment building following a FLB are bounded by main steam line break (MSLB) because the discharge from the FLB is two-phase as compared with predominantly single-phase steam discharge from the MSLB. Analyses of FLB to provide blowdown mass and energy releases will use the same maximum inventory as used in the MSLB analysis. This will provide a conservative mass and energy release to containment or subcompartments.

A.3.1.6 Pressurizer Level

Consistent with the original accident analysis on the B&W-designed PWRs, the initial pressurizer level will be set to the nominal value because a plant control system and the plant operator maintain the level at the nominal value. The exception to this rule is that for those accidents that cause an increase in pressurizer liquid level and/or reactor coolant system pressure, the initial pressurizer level will be set to a value greater than or equal to the nominal value plus measurement uncertainty. The reason for adding measurement uncertainty for these types of calculations is that the pressurizer level has an effect (albeit small) on pressure that is not addressed by adding uncertainties to reactor protection system trip setpoints or pressurizer safety valve setpoints. Events for which the pressurizer level measurement uncertainty will be bounded by the initial condition include:

1. Turbine trip or loss of external electrical load.
2. Loss of feedwater.
3. Rod withdrawal from a subcritical condition (startup accident).
4. Rod withdrawal at power.
5. Control rod assembly ejection.
6. Locked reactor coolant pump rotor.

A.3.2 Boundary Conditions

Boundary conditions for any given analysis might include reactor protection system trip setpoints and delays, safety valve setpoints, engineered safety feature actuation system setpoints and delays, emergency core cooling system flow rates, emergency feedwater flow rates, secondary system isolation setpoints, and isolation valve closure times, to name several. It is unlikely that every possible system, setpoint, or performance characteristic can be named here. The FTG philosophy is consistent with nuclear industry practice. Setpoints, measurement uncertainties, actuation delays, flow capacities, etc. are set to provide a conservative accident analysis and bound plant operation for the particular component or parameter that is modeled.

A.3.2.1 Reactor Trip Setpoints

Reactor trip setpoints are set conservatively high or low, depending upon which setting will provide the most conservative result. The accident analysis setpoint is equal to the allowable value in plant Technical Specifications--or the Core Operating Limits Report (COLR)--plus measurement uncertainty, which is applied in a limiting direction. The setpoint might also contain design margin to allow modifications to the plant equipment without invalidating the accident analysis. A conservative delay time is modeled following calculation of a trip condition to the start of gravity insertion of the control rod assemblies.

For most accidents, minimum tripped control rod worth provides the limiting condition with respect to core DNBR or post-trip return to power. In that case, the control rod worth available for insertion upon reactor trip is equal to the worth required to maintain the Technical Specification--or COLR--shutdown margin requirement at hot, zero power conditions with the highest worth rod stuck out of the core.

A.3.2.2 Core Decay Heat

When the B&W-designed PWRs were originally licensed in the 1970's, there was some question regarding the uncertainties in the decay heat standards used in accident analyses. Consequently, many analyses used to size auxiliary feedwater system characteristics (e.g., actuation setpoint, pump head requirements, flow capacities) were performed using 1.2 times the ANS 1971 decay heat standard. Since the mid-1980's the industry has better quantified the uncertainties in core decay heat following shutdown with the implementation of the ANS 1979 decay heat standard.

The ANS 1979 decay heat standard provides a more accurate prediction of core decay heat following reactor trip, and provides a method to conservatively apply uncertainties. However, the calculated decay heat using this standard will vary with power history, initial U-235 enrichment of the feed assemblies, and assembly burnup. FTG desired a decay heat standard for non-LOCA accident analysis that would be conservative and would not have to be reverified for every reload core design.

To this end, FTG calculated a number of decay heat levels (including actinides) versus time-after-shutdown using the ANS 1979 decay heat standard with 2σ uncertainty. The feed assembly U-235 enrichments and core burnup were varied to provide multiple decay heat predictions. These decay heat predictions were compared with 1.0 times the ANS 1971 decay heat standard for fission ("Decay Energy Release Rates Following Shutdown of Uranium Fueled Thermal Reactors" approved by subcommittee ANS-5 in October 1971 for infinite operation) plus B&W heavy isotopes (actinides). The comparisons of these calculations are shown in Figures A.3 and A.4. It is demonstrated that 1.0 times ANS 1971 plus B&W heavy isotopes bounds ANS 1979 plus 2σ uncertainty for a wide variation in feed assembly enrichment and burnup. Consequently, with the exception of main steam line break (MSLB) accident analyses, FTG will use 1.0 times the ANS 1971 decay heat standard for fission plus B&W heavy isotopes calculation of actinides. For MSLB at end-of-cycle for which it is desired to minimize heat input to the reactor coolant system, 0.9 times the ANS 1971 decay heat standard (with no actinides) will be used.

A.3.2.3 Safety Valves

Pressurizer safety valves are modeled to lift at the nominal setpoint plus the maximum lift tolerance allowed by plant Technical Specifications. No accumulation will be modeled because all B&W-designed plants have safety valves with steam-to-seat internals (i.e., no liquid loop seal). The EPRI/C-E test data of Dresser and Crosby pressurizer safety valves with steam-to-seat internals[A.5] show that these valves lift quickly (approximately 20 to 30 msec) and pass steam in excess of the rated values (i.e. no accumulation to reach rated flow). Each valve will be modeled to pass design flow of saturated steam at design pressure. The increase in critical flow through the valve with increasing pressurizer pressure will also be modeled.

Main steam safety valves are modeled to lift seventy (70) percent open at the nominal setpoint plus maximum lift tolerance allowed by plant Technical Specifications. The valves are modeled to reach full open within three percent of the lift setpoint. When full open, each valve will be modeled to pass design flow of saturated steam at design pressure. The increase in critical flow through the valve with increasing steam pressure will also be modeled.

In some instances a conservative calculation of the parameter of interest is obtained by an early lift of the safety valve(s). An example of this condition is the calculation of radioactive steam release to the atmosphere. If this situation should arise, the safety valves will be modeled to lift full open at the nominal setpoint minus the maximum lift tolerance allowed by plant Technical Specifications. A bounding valve blowdown will also be modeled.

A.3.2.4 Engineered Safety Feature Actuation System Setpoints

In any instance in which the engineered safety feature actuation system (ESFAS) would initiate emergency core cooling, an actuation setpoint will be used that bounds plant operation. At a minimum, the setpoint will bound the allowable value in plant Technical Specifications and include a conservative application of the measurement uncertainty.

A.3.2.5 Emergency Core Cooling System

The emergency core cooling system (ECCS) will be modeled in a conservative manner for any accident analyses in which it should be called upon to operate. For those accidents in which the ECCS mitigates the effects of the accident, a maximum delay from ESFAS to ECCS operation will be assumed, and conservatively low ECCS flow rates will be modeled. For those accidents in which the ECCS aggravates the effects of the accident, minimum time delays and maximum flow rates will be used.

A.3.2.6 Auxiliary Feedwater

Auxiliary feedwater (AFW) is always modeled in a way that will bound actual plant operation. For overheating events, conservative actuation setpoints are used that incorporate maximum uncertainties and delay actuation of AFW. Similarly, maximum signal delays are modeled from the time the actuation setpoint is reached and flow is started to the OTSG. In addition, a single failure of the highest flow AFW pump is usually assumed, and a conservatively low flow rate is modeled for the intact pump that bounds pump operation and flow measurement uncertainties in the plant. For overcooling events in which AFW is not isolated by a safety-related system, a zero actuation delay and maximum AFW flow is modeled.

A.3.2.7 Secondary System Isolation

Secondary system isolation following postulated piping ruptures is plant specific. Regardless, isolation setpoints are modeled to bound the plant value, including measurement uncertanties, so as to delay conservatively the onset of isolation. Maximum signal delays are incorporated from the time an isolation signal is reached and the pumps and valves begin to perform the isolation function. Main steam and main feedwater isolation valves are modeled with stroke times that exceed the test values for those components in the plant.

A.3.2.8 Reactivity Coefficients

Reactivity coefficients are selected that provide a conservative power response for the accident. These coefficients are selected to bound the limiting time-in-core life for the accident in question. If the limiting time-in-life cannot be determined from the existing UFSAR analysis, a time-in-life study is performed to determine the limiting reactivity coefficients.

The suitability of the selected reactivity coefficients is verified during the design of each reload core.[A.4]

A.3.2.9 Single Failure Assumptions

Consistent with the Standard Review Plan, it is the FTG philosophy to model a single active failure of safety-related equipment or of equipment important to safety. However, no operating B&W-designed plants were licensed to the Standard Review Plan. Consequently, the existing plant licensing basis in the UFSAR takes precedence.

A.3.2.10 Loss of Offsite Power

Loss of offsite power (LOOP) assumptions will be consistent with the licensing basis for the plant in question. In general, if loss of offsite power could aggravate the accident consequences, then LOOP will be assumed to occur on reactor trip. However, the plant licensing basis always takes precedence with regard to the occurance of LOOP and the time or conditions under which the LOOP is assumed to occur.

A.4 Modeling Operator Actions

There are few operator actions credited for accident mitigation in the B&W-designed plant UFSARs. The B&W-designed plants are "hot shutdown" plants in that, except for calculation of offsite doses, the event is considered safely terminated when a stable temperature and pressure condition is reached with the control rods inserted in the core. Consequently, there are few operator actions required to reach this stable, hot shutdown state. Consistent with the licensing bases of the B&W-designed plants, operator actions that can be performed from the control room will not be credited prior to ten minutes following the indication of an abnormal condition. It is possible that accident analyses might require remote operator actions to mitigate or terminate the event. In that case, the licensee will provide a conservative estimate of the time required to perform the action. In addition, the licensee will verify that the mission dose to the operator to perform the operation is acceptable. Any operator actions assumed for accident mitigation must be specified in plant procedures.

A.5 Modeling Control Systems

Plant control systems and other non-safety-related equipment and systems can affect the plant response to accidents. The licensing basis for each individual plant dictates which equipment and systems are credited to mitigate the responses to a particular accident. It is the FTG philosophy that if a control system aggravates the response to a postulated accident, the effects of the control system are modeled. If a control system mitigates the response to a postulated accident, the system is assumed to do nothing. When evaluating the effects of the control system on a particular accident, all of the functions of the control system are considered. For example, following a dropped control rod assembly, power and average system temperature decrease. The integrated control system (ICS) would normally pull control rods (normally Bank 7) in response to the decline in system temperature. However, the ICS has a rod pull inhibit that would prohibit withdrawal of control rods following a dropped rod. Furthermore, the ICS in some plants would begin a power runback, which would insert the control bank(s) and reduce core power and power peaks. Consequently, for the dropped control rod assembly accident, the ICS is not modeled to perform any function.

A.5 References

A.1 Electric Power Research Institute, *Thermal Mixing in the Lower Plenum and Core of a PWR,* EPRI NP-3545, May 1984.

A.2. B&W Nuclear Technologies, *RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis,* BAW-10164P-A, Rev. 03, July 1996.

A.3 L. S. Tong and J. Weisman, Thermal Analysis of Pressurized Water Reactors, Second Edition, American Nuclear Society, LaGrange Park, Illinois, 1979.

A.4 Framatome Cogema Fuel Company, BAW-10179P, *"Safety Criteria and Methodoloy for Acceptable Cycle Reload Analyses*, BAW-10179P, Revison 2, Ocotober 1997.

A.5 Electric Power Research Institute, NP-2770-LD, Volumes 4 and 6, Research Project V102-2, *EPRI/C-E PWR Safety Valve Test Report,* Interim Report, March 1983.

Table A.1 RELAP5/MOD2-B&W NSSS Models For Safety Analysis

Accident	Large Detail Model (Figure A.1)	Small Detail Model (Figure A.2)
INCREASE IN FEEDWATER FLOW	✔	
DECREASE IN FEEDWATER TEMPERATURE	✔	
STEAM SAFETY OR RELIEF VALVE FAILURE	✔	
STEAM LINE RUPTURE	✔	
DECREASE IN FEEDWATER FLOW	✔	
INCREASE IN FEEDWATER TEMPERATURE	✔	
TURBINE TRIP/LOSS OF ELECTRICAL LOAD	✔	
FEEDWATER LINE RUTPURE	✔	
TOTAL LOSS OF COOLANT FLOW	✔	
PARTIAL LOSS OF COOLANT FLOW	✔	
LOCKED REACTOR COOLANT PUMP ROTOR	✔	
CONTROL ROD BANK WITHDRAWAL FROM ZERO POWER (STARTUP ACCIDENT)		✔
CONTROL ROD BANK WITHDRAWAL AT POWER		✔
DECREASE IN BORON CONCENTRATION		✔
CONTROL ROD DROP	✔	
CONTROL ROD EJECTION		✔
SPURIOUS ECCS OPERATION	✔	
STEAM GENERATOR TUBE FAILURE	✔	
ANTICIPATED TRANSIENTS WITHOUT SCRAM	✔	

FIGURE A.1 LARGE DETAIL RELAP5/MOD2 MODEL FOR SAFETY ANALYSIS OF B&W-DESIGNED PWRS (SHEET 1 OF 2)

c, e

FIGURE A.1 LARGE DETAIL RELAP5/MOD2 MODEL FOR SAFETY ANALYSIS OF B&W-DESIGNED PWRS (SHEET 2 OF 2)

c, e

FIGURE A.2 REDUCED DETAIL RELAP5/MOD2 MODEL FOR SAFETY ANALYSIS OF B&W-DESIGNED PWRS (SHEET 1 OF 2)

c, e

FIGURE A.2 REDUCED DETAIL RELAP5/MOD2 MODEL FOR SAFETY ANALYSIS OF B&W-DESIGNED PWRS (SHEET 2 OF 2)

c, e

FIGURE A.3 COMPARISON OF ANS5.1 1979 DECAY HEAT STANDARD PLUS TWO SIGMA UNCERTAINTY WITH 1.0 TIMES ANS5.1 1971 DECAY HEAT STANDARD PLUS B&W HEAVY ISOTOPES

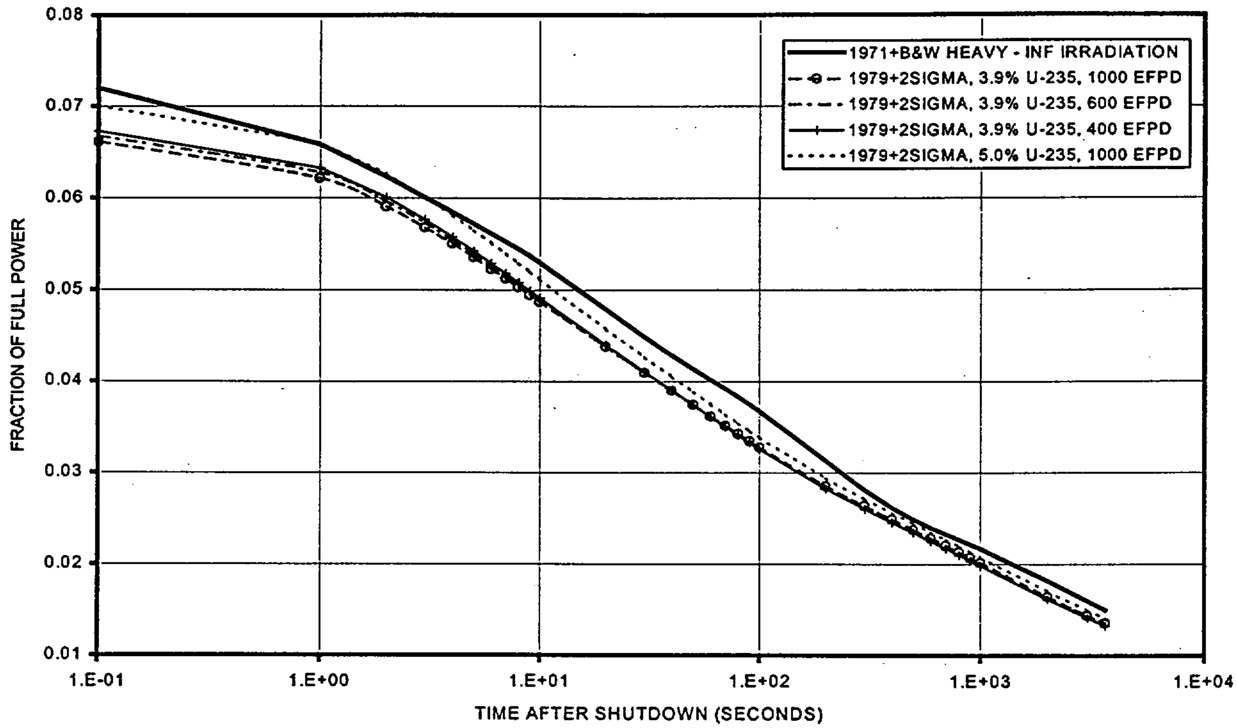

FIGURE A.4 COMPARISON OF INTEGRATED VALUES OF DECAY HEAT FOR ANS5.1 1979 PLUS TWO SIGMA UNCERTAINTY AND ANS5.1 1971 PLUS B&W HEAVY ISOTOPES