BAW-10169-A
Topical Report
October 1989

- RSG PLANT SAFETY ANALYSIS -

B&W Safety Analysis Methodology for Recirculating Steam Generator Plants

by

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

August 20, 1989

Mr. J. H. Taylor, Manager
Licensing Services
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P. O. Box 10935
Lynchburg, Virginia 24506-0935

Dear Mr. Taylor:

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT, BAW-10169P, "RSG PLANT SAFETY ANALYSIS"

We have completed our review of the subject topical report submitted by the Babcock & Wilcox Fuel Company (BWFC) by a letter dated October 22, 1987. We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation. This evaluation is enclosed and defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to ensure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that BWFC publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, BWFC and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

A C Thadani

Ashok C. Thadani, Assistant Director
for Systems
Division of Engieering & Systems Technology
Office of Nuclear Reactor Regulation

Enclosure:
Safety Evaluation

ENCLOSURE

SAFETY EVALUATION OF THE BABCOCK & WILCOX COMPANY
TOPICAL REPORT BAW-10169,
RSG PLANT SAFETY ANALYSIS

1.0 INTRODUCTION

In its effort to market fuel reloads for the pressurized water reactor (PWR) plants equipped with recirculating steam generators (RSG), the Babcock & Wilcox Fuel Company (BWFC) has developed reload safety analysis methodologies for loss-of-coolant accidents (LOCA) and non-LOCA transients and accidents. The LOCA evaluation model is described in Topical Report BAW-10168P, RSG LOCA (Ref. 1). The approach for safety analysis of non-LOCA transients is described in Topical Report BAW-10169P, "RSG Plant Safety Analysis," which is the subject of this safety evaluation.

The non-LOCA safety analysis methodology uses a reactor system transient computer code complemented with other core physics and thermal hydraulic codes to perform calculations of transients and accidents. A system transient analysis code, RELAP5/MOD2-B&W (Ref. 2), is used to model and calculate the system responses for each transient. Reactor core power during each transient is calculated by the point kinetics neutronic model in RELAP5/MOD2-B&W, with the physics parameters, such as reactivity coefficients and power peaking factors, obtained from independent core physics codes, such as PDQ07 and FLAME3 (Refs. 3 and 4). The resulting core thermal hydraulic conditions calculated by RELAP5/MOD2-B&W, such as power level, core flow rate, temperature and pressure as functions of time, are used as boundary conditions for another core thermal hydraulic code, such as LYNXT (Ref. 5) or a combination of LYNX1 and LYNX2 (Refs. 6 and 7), to determine the hot fuel rod temperature and departure from nucleate boiling ratio (DNBR). A critical heat flux (CHF) correlation, such as BWCMV (Ref. 8), is used in the subchannel code to calculate CHFs and DNBRs.

Rather than illustrating the complete safety analysis methodology, Topical Report BAW-10169 provides (1) a description of how the BWFC will use RELAP5/MOD2-B&W for modeling the reactor primary and secondary systems for analyses of various non-LOCA transients and accidents and (2) comparisons of the results of RELAP5/MOD2-B&W analyses with the results from safety analysis reports (SARs) for several plants with the Westinghouse four-loop design. The objective of Topical Report BAW-10169 is to show that RELAP5/MOD2-B&W, with proper reactor system noding model and inputs, is a viable tool for calculating transient reactor system response as part of a non-LOCA safety analysis. Justification for the plant-specific inputs used in safety analysis will be provided in plant-specific application reports. This objective is not without merit because all of the codes used for safety analysis except RELAP5/MOD2-B&W have previously been approved by the Nuclear Regulatory Commission (NRC) for licensing calculation.

RELAP5/MOD2-B&W, described in BAW-10164P, is currently under NRC review. The BWCMV CHF correlation, described in BAW-10162P, has been reviewed and approved for application to Mark BW fuel and Westinghouse optimized fuel assemblies.

Therefore, this safety evaluation addresses the acceptability of using RELAP5/MOD2-B&W with proper reactor system noding details for calculation of transient system response as part of the reload safety analysis of the non-LOCA accidents and transients, but does not address the individual codes and CHF correlations. The acceptability of the individual codes and correlations are addressed in their respective safety evaluation reports, and the limitations or restrictions associated with these codes or correlations would apply also to the safety analysis methodology. Specific inputs required for each safety analysis will be addressed during the review of the plant-specific application reports.

2.0 STAFF EVALUATION

The review includes the determination of the suitability of the RELAP5/MOD2-B&W code, the appropriateness of the reactor system modeling, the benchmark analysis, and the non-LOCA transients and accidents chosen for safety analysis.

2.1 RELAP5/MOD2-B&W

RELAP5/MOD2-B&W is a BWFC version of the RELAP5/MOD2 code. RELAP5/MOD2 was developed by the Idaho National Engineering Laboratory as a best-estimate code for simulation of a wide variety of PWR system transients of interest. The code, which is modularized according to components and functions, has been designed to model the behavior of all major components in the reactor system during accidents ranging from large-break LOCAs to anticipated operational transients involving the plant control and protection systems. The primary system, secondary system, feedwater train, system controls, and core neutronics can be simulated. Special component models include pumps, valves, heat structures, electric heaters, turbines, and separators and accumulators.

The fundamental equations, constitutive models and correlations, and method of solution of RELAP5/MOD2 are described in NUREG/CR-4312 (Ref. 9). A very detailed description of the models and correlations used in RELAP5/MOD2 was recently published in NUREG/CR-5194 (Ref. 10). RELAP5/MOD2-B&W preserves the original models of RELAP5/MOD2 and adds features and models required for licensing analysis for both LOCA and non-LOCA accidents and transients. A description of RELAP5/MOD2-B&W including the original RELAP5/MOD2 models and the BWFC modifications is provided in the Topical Report BAW-10169.

The RELAP5/MOD2-B&W hydrodynamic model is a one-dimensional, transient, two-fluid model for flow of a steam-water two-phase mixture. The two-fluid field equations consist of six equations: two phasic continuity equations, two phasic momentum equations, and two phasic energy equations. Therefore, RELAP5/MOD2-B&W is capable of treating nonhomogeneous, nonequilibrium flow. The hydrodynamics model also contains several options for invoking simpler hydrodynamics 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 violation of the material Courant limit and is suitable for steady-state calculations and for slowly varying, quasi-steady transient calculations.

RELAP5/MOD2-B&W has a point kinetics model with six delayed neutron groups to perform reactor physics calculation. It has provisions for fuel temperature, moderator temperature, and density reactivity feedback. Other reactivity feedback such as would result from boron concentration changes or tripped rod reactivity is provided, input tables for generalized reactivity versus time also are provided.

The constitutive models in RELAP5/MOD2-B&W include models for defining flow regimes and flow-regime-related models for wall friction, interfacial mass transfer, heat transfer, and drag force. A core structure heat transfer model and a fuel pin heat conduction model with a dynamic fuel cladding gap conductance model are included. The core heat transfer package is capable of calculating heat transfer coefficients for various heat transfer regimes ranging from single-phase convection nucleate boiling to post-CHF heat transfer regimes.

There are other special features very useful in thermal hydraulic analysis of PWRs, such as abrupt area change for single-phase and two-phase flows, centrifugal pump performance model with two-phase degradation effects, choked flow with treatment for horizontal stratification, nonhomogeneous two-phase flow, counter-current flow, cross-flow junction, decay heat, fine mesh renodalizing scheme for heat conduction, liquid entrainment, motor valve model, relief valve model, control system, and trip system.

The review of RELAP5/MOD2-B&W will be addressed separately in a safety evaluation report on BAW-10164P, "RELAP5/MOD2-B&W." In its review to date, the staff has concluded that the code contains appropriate phenomenological models suitable for calculating both LOCA and non-LOCA transients.

2.2 Plant System Modeling

In Figures 6.1 and 6.2 of BAW-10169P, BWFC developed two plant noding models to be used in conjunction with RELAP5/MOD2-B&W for safety analysis. Figure 6.2 is specifically for analysis of a steam line break at low power and is therefore called the low power model, whereas Figure 6.1 is for analyses of all other transients at full power and is termed the full-power model. BWFC developed these two models to provide noding arrangements with sufficient details to be able to describe important transient phenomena with sufficient accuracy of calculation, yet simple enough to minimize computer calculation time.

BWFC developed these models by using the reactor system noding arrangements in available literature. The full-power reactor vessel model (Figure 6.1) is similar to the LOFT test reactor vessel model described in NUREG/CR-3608 (Ref. 11), and the low-power steam-line-break vessel model (Figure 6.2) is similar to the model described in WCAP-7909 (Ref. 12). The rest of the plant model is consistent with the noding arrangement described in NUREG/CR-3977 (Ref. 13). The plant geometry and plant parameters are consistent with the 3411 MWt four-loop plants with a Westinghouse type-51 steam generator, such as the Trojan unit. For plants with different arrangement or configuration, the system noding should be modified to model the actual plant-specific geometry. For example, for a PWR with a different type of steam generator, such as a preheat D5 steam generator, the steam generator noding should be modified accordingly. The baffle gap is modeled as an upflow channel in Figures 6.1 and 6.2 and should be changed if a plant has a downflow configuration.

Each of the models consists of (1) a single loop and single recirculating steam generator (RSG) for the affected loop of such transients as a locked reactor coolant (RC) pump rotor and a steam line break and (2) a combined triple loop and triple RSGs for the intact loops. No heat structures are included except for heat transfer structures of fuel rods and RSG tubes. This is a conservative approach because it results in faster transient heatup and cooldown rates. Cold-plant dimensions are used for the model, which is conservative because its smaller minimum water volume also leads to higher heatup and cooldown rates.

The pressurizer model does not have a heater or spray model because, as the report indicates, they tend to minimize the heatup and cooldown rates during transients. The pressurizer spray and heater systems are part of pressurizer pressure control system. The staff agrees that no credit should be taken for control system operation if that operation mitigates the results of an accident or transient. If a control system operation results in more severe accident results, the control system should be considered in the safety analysis. For certain transients, such as a turbine trip, that result in a primary system pressurization. operation of pressurizer spray would mitigate pressurizer pressure. Neglect of the spray model would be conservative for overpressure consideration because it results in higher pressure. However, it could be nonconservative for the DNBR consideration because higher pressure would result in higher CHF and DNBR. Therefore, not modeling the pressurizer spray is not acceptable for those transients when it could result in nonconservative results. The modeling of operation of control systems in the safety analysis should be determined on the basis of specific accidents and safety parameters under consideration.

For the steam line break model, cross-flow paths are provided at the core inlet and outlet to allow for flow mixing between the faulted loop and the combined intact loop. Since the reactivity feedback is affected by the faulted loop coolant density, cross-flow mixing is an important parameter in determining the reactivity and return to power. In response to a staff question (Ref. 14) with regard to flow mixing modeling, BWFC indicated that it used an 80/20 mixing assumption, that is, the mixing of 80-percent-cold and 20-percent-hot fluid at the core inlet of the faulted loop section, based on an EPRI report (Ref. 15) that indicates this approach is conservative for a four-loop Westinghouse plant. In addition, the flow mixing is used in conjunction with a reactivity weighting scheme for the determination of reactivity feedback. The reactivity weighting method used a core water weighting factor to determine feedback of moderator temperature, density, and boron concentration used in the kinetic analysis. Since the point kinetics method is used in RELAP5/MOD2-B&W rather than a space-dependent kinetics model, a conservative result can be obtained with a proper use of a weighting factor. BWFC provided a sensitivity study to show the effects of mixing and feedback weighting on the steam line system parameters, especially the neutron power. The results show that the reactivity weighting is the more significant of the two modeling variables and that the equal weighting of the temperature reactivity contribution of the faulted and unfaulted core segment is a more conservative representation. Therefore, BWFC has determined that, for reload safety analysis of a steam line break, the 80/20 junction flow area mixing in the faulted loop and 50/50 faulted/unfaulted reactivity feedback weighting scheme will be used.

2.3 Transients Selection

To demonstrate the validity of the RELAP5/MOD2-B&W system code and the plant system modeling details of Figures 6.1 and 6.2 for plant safety analyses of non-LOCA transients, BWFC analyzed five bounding and representative transients and compared the results to those from existing safety analysis reports (SARs) of several representative plants with Westinghouse four-loop design. These five transients were chosen on the basis of the study of four major transient categories listed in Regulatory Guide 1.70. The four major categories are an increase and a decrease in heat removal by the secondary system, decrease in RC flow, and reactivity and power distribution anomalies. These categories are described below.

(1) For the category of "Increase in Heat Removal by the Secondary System," a steam line break at zero power is chosen as the representative case. This accident causes a reduction of RC temperature and, because of the negative moderator temperature coefficient, a return to power and possible fuel failure. A main steam pipe rupture is an American Nuclear Society (ANS) Condition IV limiting fault, and the amount of fuel failure must not result in a radiological release exceeding the guideline values of 10 CFR 100.

(2) For the category of "Decrease in Heat Removal by the Secondary System," a turbine trip is chosen as the representative case. A turbine trip, which may be caused by a loss of external load, is an ANS Condition II moderate frequency event. A safety analysis is made to verify that, with proper protection actions, this anticipated transient would not result in the specified acceptable fuel design limits (SAFDL) being exceeded.

(3) For the category of "Decrease in Reactor Coolant System Flow Rate," a RC four-pump coastdown and a locked RC pump rotor are chosen. A locked

rotor or shaft seizure results in a sudden reduction of RC flow through the affected loop, whereas a four-pump coastdown, which may be caused by a simultaneous loss of electrical supplies to all RC pumps, will result in complete loss of forced flow (LOFF). Though both events could result in exceeding the SAFDLs and consequent fuel failure, the acceptance criteria are different. A locked rotor accident is an ANS Condition IV limiting fault; therefore, fuel failure is allowed as long as the radiological consequences do not exceed the criteria of 10 CFR 100. A four-pump coastdown or complete LOFF is classified as a Condition II event in Section 15.3.3 of the Standard Review Plan (SRP) (Ref. 16). Though the topical report classified a complete LOFF as an infrequent ANS Condition III event, BWFC, in response to a staff question (Ref. 14), indicated that the acceptance criteria for a Condition II event, that is, the SAFDLs such as the 95/95 DNBR limit, will be applied.

4. For the category of "Reactivity and Power Distribution Anomaly," a 75-pcm/sec rod cluster control assembly (RCCA) bank uncontrolled withdrawal is chosen. An RCCA bank withdrawal causes an increase in the core heat flux and the resulting higher RC temperature, which may cause a DNB if the steam generators lag. This transient is an ANS Condition II moderate frequency event, and the SAFDLs must not be exceeded.

The selection of these five transients provides a bounding and representative set of a variety of transient categories to demonstrate acceptability of the RELAP5/MOD2-B&W and the plant modeling for analyzing such transient categories. However, the reload safety analysis should not be limited to the bounding cases in each transient category because various transients and accidents have different acceptance criteria depending on the frequencies of occurrence or the ANS classifications. In response to a staff question (Ref. 14), BWFC provided a detailed list of transients and a discussion for each transient in the list to show how it will be treated in the reload analysis. The staff finds this to be acceptable guidance for the selection of transients for the plant specific safety analysis application reports. Specific findings on the adequacy of the transients chosen will be made during the staff's review of these reports.

2.4 Comparative Analysis

Analyses were performed for comparison with the results from the representative plants. The four plants chosen for these comparative analyses are Catawba, McGuire, Trojan, and the Westinghouse RESAR. Westinghouse performed the safety analyses of these plants using a system code LOFTRAN (Ref. 17) to calculate transient system responses, a subchannel code THINC-IV (Ref. 18) to calculate core thermal hydraulic and DNBR, and/or the FACTRAN code (Ref. 19) to calculate thermal behavior of the hot rod. The BWFC calculations provided in Chapter 7 of the topical report were performed using RELAP5/MOD2-B&W only. Although it is indicated in the topical report and in response to staff questions that, for reload safety analyses, a detailed thermal hydraulic code such as LYNXT will be used in conjunction with RELAP5/MOD2-B&W for the hot channel DNBR and hot rod thermal calculations, no detailed core thermal hydraulic calculations were performed for the comparative analyses. This is because the purpose of these analyses was to demonstrate that RELAP5/MOD2-B&W, with proper plant system modeling and appropriate major assumptions and inputs, can provide results with a reasonable agreement with those of an approved code such as LOFTRAN for such major parameters as neutron power, thermal power, reactor system pressure, and temperature and flow. The DNBRs were calculated using a RELAP5/MOD2-B&W control variable algorithm that considered thermal power, fluid flow, fluid temperature, and system pressure at constant values of power peaking as opposed to detailed thermal hydraulic analysis done for safety analysis. Though these DNBRs are not a valid calculation, they can provide comparisons of trends and timing with THINC and FACTRAN results.

To be consistent with the analyses performed for the representative plant SARs, the comparative analyses used the same assumptions and the same initial and boundary conditions as those in the SARs and the results were directly compared with the SARs. For instance, the radial and axial power distributions were determined by the use of an enthalpy rise factor and total peaking factor from each plant's Technical Specifications. Although the fuel conduction model will have five radial fuel nodes, one gap and two cladding nodes in the reload analysis, a single radial fuel node was modeled to be consistent with that used in the LOFTRAN code in the SAR. The reactivity coefficients that were used for the comparative analyses used the same approach as the SARs by using the Doppler power coefficient even though the reload application will use Doppler coefficient as a function of fuel temperature. However, BWFC analyses were done using the initial input values of the Doppler and moderator coefficients rather than subject them to changes as function of moderator temperature and power as was done in the SARs.

There also are differences between LOFTRAN and RELAP5/MOD2-B&W that could cause differences in the calculated results. For example, in calculating the loop and core coolant flow, RELAP5/MOD2-B&W considers the effect of fluid temperature on the RC pump performance and flow variation for various locations in the system, whereas LOFTRAN, using a constant flow model where the flow changes are input as function of time, does not consider these effects. Another difference is in the pressurizer modeling where LOFTRAN pressurizer uses a two-node equilibrium model, while the BWFC model is an eight-node nonequilibrium model. This also could cause differences in pressurizer pressure response.

The results of analyses and comparison to those from the four selected plant SARs are given in the topical report. Comparisons are made for neutron power, thermal power, pressurizer pressure and water level, core average temperature, core inlet temperature, core flow, faulted loop flow (for locked rotor and steam line break), and DNBR. In general, these comparisons show reasonably good agreement. Even though the magnitude of DNBRs calculated by RELAP5/MOD2-B&W are not valid in that they are not calculated with a valid correlation for the hot channel, they do show good agreement in the trend of its behavior. Therefore, the staff concludes that RELAP5/MOD2-B&W with proper reactor system noding details and inputs can be used as a part of the safety analysis of transients and accidents. Justifications for plant-specific noding and inputs will be required as part of the plant-specific safety analysis.

2.5 Reload Application

Chapter 7 of the topical report also provides application analyses to demonstrate the approach that BWFC intends to use for reload safety analyses for certain transients. The application analyses use the same reactor system noding details and same assumptions and approaches as the comparative analyses, with a few exceptions. The most notable differences between the application and comparative analyses are that (1) rather than just one radial fuel node, the application model has five radial fuel nodes and (2) rather than use of the Doppler power coefficient, the reload application model uses Doppler fuel temperature coefficient consistent with the time in life of a transient. Since the application models were not performed for an actual plant reload, the staff review concentrated on the validity of the approach and assumptions made in performing each transient rather than the quantitative results. The staff review findings of the application analyses are given below.

(1) As was the case for the comparative analyses, the application analyses were performed with RELAP5/MOD2-B&W only. No detailed core thermal hydraulic calculations were made with an approved subchannel code or approved CHF correlation. Therefore, the conclusion is not valid that the minimum DNBR is never less than the limit value for the rod withdrawal, complete loss of flow and turbine trip transients, respectively.

For reload safety analysis, RELAP5/MOD2-B&W should be used in conjunction with an approved detailed subchannel thermal-hydraulic code, such as LYNXT, and CHF correlation for the DNBR calculation. The noding details and inputs should be justified on a plant-specific basis.

(2) In response to a staff question (Ref. 14) regarding treatment of a mixed core of different fuel designs having different hydrodynamic characteristics during a transitional reload period, BWFC indicated that the core thermal hydraulic calculation will be based on a homogeneous core model of BWFC fuel and, if necessary, adjustments will be made on

DNBR to reflect a mixed core penalty. BWFC indicated this mixed-core penalty will be determined on a plant-specific basis.

(3) The analysis of the uncontrolled RCCA withdrawal at power was done with a moderator coefficient of zero in conjunction with the Doppler reactivity at end of life. In response to a staff question regarding the reactivity feedback assumptions for the RCCA withdrawal analysis, BWFC stated that to assure that both maximum and minimum coefficients are considered, reload safety analysis of the maximum positive insertion rate of 75-pcm/sec rod withdrawal will be performed assuming the most positive moderator coefficient and least negative Doppler coefficient. Since the maximum positive reactivity insertion rate is greater than that for simultaneous withdrawal of the combination of the two control banks and the maximum combined worth at maximum speed, it is assumed to be the limiting case. However, this pre-supposed assumption that the limiting case is the maximum reactivity insertion rate and certain feedback coefficients is not appropriate. In fact, a fast reactivity insertion rate may result in an earlier reactor trip by high flux trip and result in higher DNBR relative to a slower reactivity insertion rate case where the reactor may be tripped later in time by the over-temperature delta T trip and result in lower DNBR. Therefore, as stated in Section 15.4.2 of the SRP, the analysis of uncontrolled RCCA withdrawal transient should consider various reactivity insertion rates from very low to maximum possible for the control system and the fuel and moderator feedback reactivity coefficients covering the range expected throughout the cycle, including allowance for uncertainties.

(4) For the RC pump locked rotor transient, the analysis conservatively assumed the fuel rod at the hot spot was in DNB at the beginning of the transient, with implication that the fuel failure criterion for locked rotor accident was the peak cladding temperature of 2700°F, which has not been accepted by the NRC. In response to a staff question, BWFC indicated that the 95/95 DNBR limit will be used as the fuel failure criterion.

Also, no discussion was made regarding the treatment of cladding oxidation, though the zircaloy-steam reaction and its exothermic effect becomes significant for cladding temperature above 1800°F. In fact, there is no cladding oxidation model in the core thermal hydraulic code LYNXT. In response to a staff question(Ref. 14), BWFC indicated that the metal-water reaction effects will not need consideration because it is not expected that the peak cladding temperature (PCT) will exceed 1800°F because the reload analysis will not assume DNB at the start of the transient. However, if the plant specific safety analysis results in the PCT exceeding 1800°F, the effect of metal-water reaction must be included in the analysis, and BWFC should propose a method of accommodating this effect as part of the plant-specific analysis.

(5) No application analysis model is provided for a steam line break. The safety analysis for a steam line break should follow the guidelines of SRP Section 15.1.5 with conservative assumptions including:

- loss of offsite power
- worst single active component failure
- maximum worth rod stuck in fully withdrawn position
- burnup at most limiting combination of moderator temperature coefficient (MTC), void coefficient, and Doppler coefficient
- only safety grade equipment will be assumed operative to mitigate the consequence of break

3.0 SUMMARY

The staff has reviewed Topical Report BAW-10169 and finds that RELAP5/MOD2-B&W and the reactor system modelings of Figures 6.1 and 6.2 are acceptable for calculating the reactor system responses in performing safety analysis of non-LOCA transients and accidents. This acceptance is subject to the following conditions and restrictions:

(1) For a complete safety analysis, an approved core thermal hydraulic code and CHF correlation should be used in conjunction with the RELAP5/MOD2-B&W code. The noding details and inputs should be justified on a plant-specific basis.

(2) The selection of transients and accidents for reload safety analysis should be done with guidelines provided in BWFC's response to NRC Question 3 (Ref. 14). For each transient or accident analyzed, the analysis should either follow the SRP guidelines or comply with the plant-specific licensing basis.

(3) During a transitional reload period having a mixed core configuration with different fuel designs having different hydraulic characteristics, a mixed core penalty should be used to account for the DNBR difference when a homogeneous core is assumed in the analysis. The mixed core penalty can be a bounding value or a value calculated on a plant-specific basis.

(4) Neglect of the operation of the control systems, such as pressurizer spray, should be determined on the bases of specific accidents and safety parameters under consideration. If the operation of a control system results in more severe results of the accident, its operation must not be neglected.

(5) If the plant-specific safety analysis for a transient or accident, such as RC pump shaft seizure, results in the PCT exceeding 1800°F, the effect of metal-water reaction must be considered and a method of accommodating this effect should be included as part of the plant-specific analysis.

(6) An uncontrolled RCCA withdrawal transient should be analyzed with a spectrum of reactivity insertion rates and with both maximum and minimum reactivity feedback to bound the DNBR response.

(7) Until other criteria are submitted and approved by NRC, the fuel failure criteria for the RC pump locked rotor accident is the 95/95 DNBR limit.

(8) A complete loss of flow is an ANS Condition II moderate frequency event, the acceptance criteria should be complied with accordingly.

4.0 REFERENCES

1. BAW-10168P, "RSG LOCA - B&W Loss-of-coolant Accident Evaluation Model for Recirculating Steam Generator Plants, Volume I - Large Break, Volume II - Small Break," July 1988.

2. BAW-10164P, "RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," December 1987.

3. BAW-10117-A, "Babcock & Wilcox Version of PDQ07 User's Manual," January 1977.

4. BAW-10124-A, "FLAME 3 - Three Dimensional Nodal Code for Calculating Core Reactivity and Power Distribution," August 1976.

5. BAW-10156-A, "LYNXT - Core Transient Thermal-Hydraulic Program," February 1986.

6. BAW-10129-A, " LYNX1: Reactor Fuel Assembly Thermal-Hydraulic Transient Code," July 1982.

7. BAW-10130-A, " LYNX2: Subchannel Thermal-Hydraulic Analysis Program," July 1982.

8. BAW-10159P, "BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies," May 1986.

9. NUREG/CR-4312, EGG-2396, "RELAP5/MOD2 Code Manual," Volumes 1 and 2, August 1985.

10. NUREG/CR-5194, EGG-2531, "RELAP5/MOD2 Models and Correlations," August 1988.

11. NUREG/CR-3608, SAND-83-2549, "RELAP5 Assessment : LOFT Large Break L2-5," January 1984.

12. WCAP-7909, "MARVEL - A Digital Computer Code for Transient Analysis of a Multiloop PWR System," October 1972.

13. NUREG/CR-3977, EGG-2341, "RELAP5 Thermal/Hydraulic Analyses of Pressurized Thermal Shock Sequences for the H. B. Robinson Unit 2 Pressurized Water Reactor," April 1985.

14. Letter from J. H. Taylor (BWFC) to J. A. Norberg (NRC), "RSG Plant Safety Analysis Topical Report BAW-10169," November 23, 1988.

15. EPRI NP-4498, "Reactor Analysis Support Package (RASP), Volume 3: PWR Event Analysis Guidelines," May 1986.

16. NUREG-0800, "Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants," July 1981.

17. WCAP-7907, "LOFTRAN Code Description," October 1972.

18. WCAP-7956, "THINC-IV - An Improved Program for Thermal/Hydraulic Analysis of Rod Bundle Cores," June 1973.

19. WCAP-7908, "FACTRAN - A FORTRAN IV Code for Thermal Transients in a UO_2 Fuel Rod," July 1972.

1. QUESTION:

In sections 4 and 7.2, a four-pump coastdown or complete loss of forced coolant flow is classified as an ANS Condition III event (Infrequent fault). This classification is inconsistent with the Standard Review Plan, Section 15.3.1, where a complete loss of forced coolant flow is classified as a moderate frequency event (Condition II) and one of the acceptance criteria is to maintain fuel cladding integrity by assuring the 95/95 DNBR limit being met. Are your safety analysis goals to comply with the acceptance criteria of SRP for a loss of flow event, or to comply with the acceptance criteria for the Condition III events (such as 10 percent of the 10 CFR 100 limit)?

RESPONSE:

The classification of the four-pump coastdown as an infrequent fault in Sections 4 and 7.2 of the topical report is consistent with the classification applied in the safety analysis reports upon which the comparative analyses were based. In safety analyses for reload fuel, B&W will apply the acceptance criteria of the SRP for the loss of flow event by confirming that the 95/95 DNBR limit is met.

2. QUESTION:

In Section 4, a small break LOCA is classified as an ANS Condition III event. What is the rationale for this classification and what are the acceptance criteria other than those of 10 CFR 50.46?

RESPONSE:

The classification of a small break LOCA as an ANS Condition III event in Section 4 of BAW-10169 was based on past industry practice as reflected in the FSAR's used as references for the topical report. For the purpose of fuel reload safety evaluations, the acceptance criteria applied to the small LOCA will be those of 10 CFR 50.46. The methods and analyses applicable to loss-of-coolant accidents are presented in other reports and documents.

3. QUESTION:

In Section 4.2, Fuel Reload Evaluation Transients, it is indicated that the five reference transients selected in BAW-10169 represent a bounding and representative set for the first four categories of Regulatory Guide 1.70. These reference transients and accidents may be the bounding cases for demonstrating the acceptability of the RELAP5 code and modeling for analyzing such transient categories as an increase or decrease in heat removal by the secondary system, decrease in reactor coolant flow, and the reactivity and power distribution anomalies. However, the reload safety analysis should not be limited to the bounding case in each category because various transients in each category have different acceptance criteria depending on frequency of occurrence (ANS classification). Please provide a list of transients and accidents intended to be analyzed using RELAP5 for reload safety analysis and the bases for the selection.

RESPONSE:

Table 1 shows the B&W evaluation of all Regulatory Guide 1.70 transients as presented and discussed in the September 14, 1988 meeting with NRC staff members. This tabulation shows the transients to be analyzed using RELAP5 for the reload safety analysis and indicates the evaluation of all other Regulatory Guide 1.70 non-LOCA transients applicable to the Catawba and McGuire Units. This same list is generally applicable to all four-loop PWR's of Westinghouse design, however, other reload evaluations and scopes of analyses would need to be established on plant-specific bases.

4. QUESTION:

In the recirculating steam generator modeling discussed on Pages 6-5 and 6-6, it is indicated that no RSG water level model was developed. How are the heat transfer regimes, the regime boundaries, and heat transfer coefficients determined?

RESPONSE:

The text on pages 6-5 and 6-6 is somewhat misleading. It is intended to state that the steam generator level instrumentation and corresponding setpoints were not modeled since those systems were not applicable to the comparative analyses. The text was not meant to imply that the physical modeling of the steam generators did not consider or represent the shell side inventories and levels.

The recirculating steam generator models depicted in Figure 6.1 of the topical report divide both the primary and secondary regions into control volumes. For each control volume the flow regime is determined by either the horizonal or vertical flow regime map built into RELAP5/MOD2. Section 2.1.3 (Constitutive Models) of BAW-10164P, "RELAP5/MOD2-B&W", describes the logic and calculations that are performed for each control volume. The heat transfer regimes depend on the CHF condition, mass flux, and void fraction within the volumes. The CHF condition is determined internally by the code as described in Section 2.2.2 of BAW-10164P. The heat transfer regime logic and calculations are described in detail within that section.

The control volumes are homogeneous within RELAP5/MOD2 thus the extensive set of control variables built into RELAP5/MOD2 could be used to calculate water level if required for control, reactor trip or other purposes.

5. QUESTION:

In the RSG steamline break (low power) model (Figure 6.2), crossflow paths (Junctions connecting nodes 315 and 316, 314 and 317, 327 and 352, and 326 and 353) are provided at the core inlet and outlet to allow for flow mixing between the broken loop and the single intact loop. Since the reactivity feedback is determined by the single (broken) loop coolant density, the crossflow mixing is an important parameter in determining the reactivity and return to power.

(a) Provide a description and the bases on how the crossflow mixing and the value of the crossflow junction areas are determined.

(b) Since your RELAP5 model is used to compare with the FSAR results of various plants, do the crossflow junction areas used in the RELAP5 model provide consistent crossflow mixing as these plants which use a crossflow mixing factor? Explain why.

(c) Are the same crossflow junction areas to be used in reload analyses?

RESPONSE:

c,d,e

c,d,e

c,d,e

c,d,e

6. QUESTION:

Tables 5.1.1 and 7.0.1 provide the values of input parameters and initial conditions for transient analyses of RESAR, McGuire, Catawba, and Trojan plants, and the corresponding values used by B&W to simulate these analyses using RELAP5. Explain why values different from the SAR values of some of the parameters, such as core coolant flow rate, inlet temperature, core average coolant temperature and steam pressure, are used in the RELAP5 comparison analyses. Also justify why the RELAP5 results using different input parameter values will provide valid comparison to the SAR results.

RESPONSE:

The use of a single RELAP5 plant model with a single set of input parameters and initial conditions for the comparison cases was adopted for two reasons: First, examination of the input parameters and initial conditions tabulated in the SAR's for the four reference plants (Trojan, RESAR, Catawba, and McGuire) showed that, with few exceptions, these values were the same for all four sets of SAR analyses. (The exceptions were, in fact, confined to those cited in the question.) Second, by replotting the SAR results curves for the four plants, with common starting points--normalizing to the same initial conditions--the results became differentiated mainly by variations in the major analysis assumptions and not by relatively minor differences in the initial conditions. This makes sense because close review of the designs represented in the four FSAR analyses confirms that they are, indeed, generic. The geometries, capacities, thermal ratings, and other defining system parameters are either the same or virtually so. This observation does allow the system analyst to consider the four designs as thermal-hydraulically similar, at least from the system analysis standpoint.

It is specifically this similarity that strengthens the comparative analyses in BAW-10169P, wherein the primary purpose is to demonstrate the consistency--in terms of the effects of variations in the major analysis assumptions that define limiting cases--between the results obtained using RELAP5/MOD2-B&W and those calculated with already-approved methodology. By selecting a group of SAR analyses, representing a generic design, and unifying the results by normalizing to a single set of initial conditions, the effects of the key transient-related parameters can be isolated. The comparative analyses are indeed directly comparative in methodology and are not complicated or rendered indirect through the use of inconsistent initial conditions.

The plant initial conditions affect the DNBR analysis because the specific initial value of the DNBR is determined by the initial conditions of power, flow, temperature, pressure and peaking factor. The relative change in DNBR during a transient, however, is not sensitive to small changes in the initial conditions. Changing the DNB correlation or location of the DNB calculation (thimble channel or average channel) has a much larger effect during a transient than do variations in the initial conditions. For plant-specific applications, the appropriate initial conditions will be used.

7. QUESTION:

With respect to the analysis of a RCCA withdrawal at power transient, it is indicated in Section 7.1.2 that for the B&W application model analysis, the Doppler reactivity feedback consistent with the rod withdrawal time in life was used, and a moderator coefficient of zero was used for all analyses. This approach is inconsistent with the FSAR methodology where both maximum and minimum reactivity feedback were analyzed. The analyses of both maximum and minimum reactivity coefficients are necessary because various values of reactivity coefficients will affect the time and type of reactor trips (such as high flux trip or overtemperature delta T trip), and therefore affect the results of analysis. Explain how your method of treating reactivity feedback is sufficient in obtaining the limiting case result.

RESPONSE:

The moderator and Doppler coefficients used in the comparison analysis were chosen to agree with the plant specific SAR values, which assumed a moderator coefficient of zero in all cases. In the applications case, the use of a zero moderator coefficient in conjunction with a Doppler coefficient consistent with rod withdrawal at end of life was intended to combine the conservative zero moderator feedback with a more representative Doppler coefficient. The zero moderator coefficient is conservative for EOL conditions.

To assure that both maximum and minimum reactivity coefficients are considered, reload safety analyses of the 75 pcm/sec rod withdrawal will be performed assuming the most positive moderator coefficient and least negative

Doppler coefficient, which is consistent with the comparison analyses.

The reactivity feedback coefficients can affect the point of transition, in terms of rod withdrawal rate, from a high flux trip to an overtemperature delta T trip. In no case, however, does this present a DNB problem because the conditions (combination of rod withdrawal reactivity and feedback coefficients) at which the overtemperature trips occur represent comparatively slow power increases for which the lag between thermal power and neutron power becomes vanishingly small as withdrawal rate decreases. In these cases the overtemperature trip occurs such that the DNBR will be greater than the limit because the overtemperature trip setpoints are reached--by design--well in advance of DNBR limit values.

8. QUESTION:

(a) In Figures 7.1.10, 7.1.17, 7.2.9, 7.2.14, 7.2.19, 7.3.11 and 7.3.18, comparisons are made on the RELAP5 calculated DNBRs with those of Trojan, RESAR, Catawba and McGuire. Since the RELAP5 calculations represent the thermal hydraulic conditions of the core average channel compared to the hot channel conditions for the referenced plants, and also different CHF correlations may have been used for the RELAP5 calculations, how meaningful are these comparisons?

(b) It is indicated in Section 7.2. regarding the application analysis results of the complete loss of flow event using RELAP5, that Figure 7.2-24 shows the DNBR to be always greater than the limit values and that no fuel failure is produced. Which plant is this analysis based on? Which subchannel thermal-hydraulic code and critical heat transfer correlation are used in the analysis? What is the DNBR limit of the CHF correlation used?

(c) In B&W's analysis of steamline break using RELAP5, which subchannel thermal-hydraulic code and critical heat transfer correlation will be used? Do the ranges of applicability of the CHF correlation used in the analysis cover the range of transient during a steamline break? For example, the primary system pressure decrease to less than 1000 psia is outside the range of the BWCMV correlation.

RESPONSE:

8a. The DNBR results from RELAP5 presented in BAW-10169 were calculated using an algorithm based upon the BWFC critical heat flux correlation. The DNBR's were included in the comparison plots to provide completeness, recognizing that actual licensing analyses would utilize more detailed, approved thermal-hydraulics computer codes such as LYNXT, with an appropriate CHF correlation. It is interesting to note the results using the algorithm, which did include the effects of hot channel peaking, show a reasonable comparison when normalized with the SAR data.

8b. The DNBR values of Figure 7.2.24 were calculated by RELAP5 using an algorithm, as described above. These results will be presented in any licensing analysis with DNB ratios calculated using approved methodology.

8c. The core thermal-hydraulic analysis calculations will be performed with the LYNXT code. Appropriate CHF correlations are currently being investigated and will be selected on the basis of applicability to the range of transient conditions during the steamline break. Justification will be provided in support of plant-specific applications.

9. QUESTION:

Page 7.1-6 states that the only differences between the comparison RELAP5 and the B&W application are the fuel radial noding and Doppler coefficients. Since no subchannel thermal/hydraulic code was used in conjunction with RELAP5 in the comparison analysis, does that mean the same approach will be used in future reload applications?

RESPONSE:

In order to obtain agreement between RELAP5 results and SAR (usually LOFTRAN) results it was necessary to use a single radial fuel node model in RELAP5--since that is the fuel modeling used in the SAR analyses-- even though a multinode model is standard practice at B&W and will be used for actual reload safety analyses. Also, various methods for treating Doppler feedback were used in the SAR analyses. The typical B&W practice bases Doppler feedback on fuel temperature, and this approach was adapted to be equivalent to the SAR assumptions. (This in no way intended to imply the SAR methods are incorrect; rather, these two modeling conventions are more consistent with previous B&W analyses.) As stated in response to Question 8, approved methods for calculation of subchannel thermal-hydraulics and determination of DNB performance will be used in reload licensing applications.

10. QUESTION:

In Section 7.2.3, it is indicated that for the complete loss of flow calculation, different methods may have been used for various SARs in the power Doppler coefficient and moderator coefficients. However, the RELAP5 analyses were all done with the initial feedback taken as a constant throughout the transient. What is the validity of these comparisons in benchmarking RELAP5 when different reactivity feedback is used?

RESPONSE:

In reviewing the assumptions used for Doppler feedback modeling in the SAR analyses, it was noted that, in certain cases--the loss of flow transient cited in the question is an example--the specifications were incomplete. The SAR's do present the bounding Doppler coefficients as functions of power, and one of these figures is reproduced as Figure 7.0.1 in BAW-10169. When "maximum" Doppler is specified for the SAR loss of flow cases, it is clear which of the two curves is the operative one, but whether constant or variable Doppler feedback is assumed is not explicitly stated.

As it turns out, precise replication of the Doppler feedback used in the SAR cases is not vital to achieving valid comparison cases. While the Doppler modeling does reflect one--among a number--of the core-related parameters that define the limiting case, it is the trip reactivity insertion that is the overriding consideration. This was established in preliminary sensitivity studies in which both the Doppler feedback and rod insertion rate effects were examined. It was concluded that the time assumed for 85 percent rod insertion had a major effect upon the post-trip power shape. Doppler feedback was found to have an effect, albeit a relatively minor one. The approach taken, then, was to select a conservatively large (absolute value), constant value for Doppler-only power feedback--as will be used for actual reload safety analyses--for the comparison cases, having established that the effect upon the results would not be sufficient to invalidate direct comparison to the SAR curves.

11. QUESTION:

Section 7.4 discusses the reactor coolant pump shaft seizure accident. It is indicated that no credit is taken for the pressure reducing effect of the pressurizer spray or controlled feedwater flow after plant trip.

(a) Do you take credit for the pressure reducing effect of pressurizer power operated relief valves and steam dump?

(b) No discussion is made regarding the effect of cladding oxidation. Since the zirconium-steam reaction becomes significant for cladding temperature above 1800^0F, do you consider the effect of metal-water reaction in the pump seizure accident analysis?

RESPONSE:

11a. No credit was taken for the pressure reducing effect of pressurizer power operated relief valves. The analysis shown in Section 7.4 was done with an assumed steam dump, however the steam dump was modeled to occur after reactor trip and had no effect on the peak parameter results that occur at 4 seconds. Even though the steam dump does not affect the peak parameter results, the steam dump will not be used in any reload licensing analysis wherein the system pressure is increasing.

11b. For the pump seizure (locked rotor) accident analysis, the LYNXT code will be used to predict the minimum DNBR and the transient fuel rod temperatures. LYNXT does not consider the effect of metal-water reaction in these calculations, but also does not require the conservative assumption of DNB at the start of the transient. It is expected that, with the ability of LYNXT to predict the time and axial location of the inception of DNB, the peak cladding temperature will be less than 1800 F and therefore the metal-water reaction effects will not need consideration.

12. QUESTION:

In Section 7.4, it is indicated that the peak cladding surface temperature of the pump seizure accident is expected to be considerably less than 2700^{0}F is an acceptance criterion for determining fuel failure for the pump seizure accident? Please note that the staff has only accepted the 95/95 DNBR limit as the fuel failure criterion.

RESPONSE:

The 95/95 DNBR limit will be used as the fuel failure criterion.

13. QUESTION:

In Section . .5.3, you indicate that the steamline break analysis is heavily plant specific in auxiliary feedwater flow and boron injection assumptions. However, only a single RELAP5 analysis is made to compare with the SAR results of Trojan, RESAR, Catawba and McGuire.

(a) Please identify which plant your RELAP5 analysis is based on.

(b) Why is it valid to compare your RELAP5 results to other plants which you do not model?

(c) What is your basis for concluding the RELAP5 model has shown agreement with SAR steamline break results of several different plants?

RESPONSE:

13a. The generic RELAP5 model used in the analysis most directly corresponds to the design described and analyzed in the Trojan FSAR. That is to say, the RELAP5 model represents a Westinghouse generic four-loop PWR design rated at 3411 Mwt with Type 51 steam generators and conventional ECCS. The performance parameters specified for the model were taken from Trojan SAR data.

13b. In arriving at the main steam line break comparison case, the four reference SAR's were reviewed in detail. Several points distinguished this set of transient results from those of the other comparison cases. First, the scaling used for presentation of the plotted results in the SAR's made it most difficult--not possible, in fact--to digitize and replot the SAR curves on common axes. On the other hand, the timing and relative magnitudes of the key events or phenomena could be ascertained from the complete sets of results: the plots, tables, and descriptive text. What these indicate is that, for the four reference safety analyses, the steam line break results are differentiated mainly by the boron addition characteristics. This is exemplified by contrasting the earlier Trojan SAR plots (Figure 7.5.3 in the report) to those taken from the July 1985 update for Trojan (Figure 7.5.4) The later results, for 2000 ppm boron addition line up well with the Catawba and McGuire results which are based on the same boron concentration. The remaining differences are relatively subtle and are associated mainly with inputs derived from plant data at a level of detail beyond that provided in the SAR's.

It was concluded that, from the standpoint of the major system parameters, the SAR cases at the same boron injection concentration were indeed generic and that a single comparison case would provide valid demonstration of the sensitivity of the RELAP5 model to the major inputs. Further, the depth of information used to model the plant- or analysis-specific inputs is available in various degrees among the four reference SAR's. The decision was to use the same plant model that had been used for the other comparison studies in conjunction with the most complete available data base for inputs, assumptions, and comparable results.

As acknowledged in response to 13a, the RELAP5 model used in these studies is most closely tied in initial conditions to those of the Trojan FSAR. For the steam line break case, however, the differences among the four SAR analyses for hot zero power initial conditions vary from barely distinguishable (RCS initial statepoints) to marginal (steam generator inventories). The RCS total volumes and distributions are virtually the same, differing mainly in the reactor vessel internals, which do not play a large part in the steam line break event. So, from this standpoint, it is not unreasonable to use a set of event-specific inputs that has the best commonality with all four reference SAR's to arrive at results that are valid for comparison on the points compared in BAW-10169P. To be sure, the results are compared more extensively to those drawn from the Catawba and McGuire FSAR's. That is simply because these references provided the most extensive set of presented results, and more information was available to support inputs applicable to those plant-specific cases.

13c. The response to this question has been included in the response provided for 13b since the two are closely linked.

14. QUESTION:

The comparison analyses presented in BAW-10169P demonstrate qualitatively that RELAP5 can predict the trend of the transients analyzed. Since the acceptability of the B&W safety analysis methodology using RELAP5 is demonstrated by comparing the results of the B&W calculations to the referenced SAR results, please provide more detailed comparisons to show that your calculations are either in agreement with or more conservative than those calculated with approved codes. To establish the validity of the comparisons the following should be considered:

(a) Comparison should be made between the RELAP5 output and the SAR results which reflect the outputs of the approved system code LOFTRAN, and the subchannel thermal hydraulic code such as LYNXT to the output of the subchannel code THINC for the hot channel conditions. For the DNBR result comparisons to be meaningful, the same CHF correlations as used in the SARs should be used.

(b) Detailed comparison should be made to the important parameters including the reactor coolant flow rate, power level, pressure and core inlet temperature as a function of time, the local enthalpy, mass flux, temperature, coolant density, void fraction, heat flux and DNBR distributions along the axial channel for each time step.

(c) For the comparison to be meaningful, the calculations should use initial conditions and other parameters such as reactivity feedback coefficients as those used in the referenced plants.

RESPONSE:

14a. With the exception of the DNBR results--these are discussed in response to Question 8--the remainder of the comparisons presented in BAW-10169P are direct comparisons of system analysis results calculated by RELAP5/MOD2-B&W to system analysis (primarily LOFTRAN) results calculated by Westinghouse. Thus, there is consistency in the overall set of comparison studies. The overriding purpose in these analyses was to demonstrate that, for the same <u>or equivalent</u> major assumptions and inputs, the B&W application of RELAP5 to SRP transients in Westinghouse-designed reactors would produce system results consistent with those produced by reviewed and approved methods.

14b. In each of the comparison cases presented in BAW-10169P, the results shown comprised the full set of plotted system parameters as presented in the respective SAR's. Obviously, the actual SAR analyses produced a wider scope of output variables than was actually presented, but it is not unreasonable to conclude that the ones set forth in the SAR plots constitute a sufficient number of system parameters to define the results. The RELAP5 results for the comparison cases do include such system parameters as reactor coolant flow rate, core power level (both neutron power and thermal power), and core inlet temperature--among others--for those cases in which these parameters make up the definitive set of system results.

14c. As stated earlier, the differences in initial conditions do not affect the trend of the transient and the exact values in terms of system parameters (power fraction, flow fraction, pressure change and temperature change) have been adjusted for by normalization. Exact initial conditions for each plant would produce different numerical initial values of DNBR but the change in DNBR during the transient will be controlled by the DNBR correlation and transient system parameter which have been shown to agree with previous SAR results.

15. QUESTION:

During the transition period when the B&W reload fuel and the existing Westinghouse fuel will coexist in a mixed core configuration, how do you intend to model the mixed core and different fuel designs? If a homogeneous core of single fuel design will be modeled, how do you intend to correct for the effect of other fuel design having different hydraulic characteristics? If you intend to apply a mixed core penalty factor, how do you obtain the factor and how would you apply it?

RESPONSE:

During the transition from an existing fuel design to a BWFC fuel design the core thermal-hydraulic evaluations will be based on a homogeneous core model of the BWFC fuel, with adjustments to reflect a mixed core penalty if necessary. The mixed core penalty will be determined on a plant-specific basis by modelling a limiting mixed core combination and comparing the DNBR predictions for that model to those for the homogeneous model. The process to be followed will be:

1. A homogeneous core model will be established for the BWFC reload fuel design. This will be a single-pass LYNXT model, developed in the manner described in BAW-10156-A. This model will be used for DNBR safety evaluation and design calculations, including the establishment of core safety limits, analysis of DNB-limited transients, etc. The criterion used in conjunction with this model will be that the minimum DNBR will be no less than the Thermal Design Limit (TDL) as described in BAW-10170P. The TDL provides some margin to the 95/95 safety limit, called the

Statistical Design Limit (SDL) also defined in BAW-10170P. This margin is intended to be used to offset the effects of penalties such as that due to the transition, or mixed core.

2. A mixed core model will be developed in which the potential hydraulic mismatch will be conservatively represented. This model will use the LYNX1 and LYNX2 codes and will represent a limiting core configuration for the transition cycles. The specifics of the model will depend on the two fuel designs being represented, but in general will be selected to permit the determination of a mixed core penalty such that a homogeneous core model, with the use of a penalty factor, can be used to conservatively predict minimum DNBR for the transition cores. The magnitude of the penalty factor will be established for a specific combination of fuel assemblies by performing several DNBR calculations at limiting statepoints and comparing the results with the homogeneous core model.

Table 1

Safety Applications Topical Report

Overview Presentation

- o Event Descriptions

  Brief summary of transient and consequences as presented in FSAR. Evaluation of consequences and listing of significant core-related parameters.

- o Applications TR Treatment

  Outline of evaluation/analysis and method to be presented in safety applications TR.

Increases in Heat Demand by the Secondary System

		Event Description/GDC's/Core Parameters	FSAR Consequences/Comments
15.1.1	Decrease in Feedwater Temperature	Reduction in FW temperature causes increase in core power by reducing RCS temperature. GDC 10 Fuel Design Limits GDC 15 RCS Pressure Boundary GDC 26 Reactivity Control	Bounded by 10 percent step load increase (15.1.3)
15.1.2	Increase in Feedwater Flow	Increase in FW flow causes rise in core power by decreasing RCS temperature. GDC 10, 15, 26. Doppler power coefficient — Min Doppler temperature coeff. — Max Moderator density coeff. — Min Moderator density coeff. — Max	MDNBR > 2.0 and not limiting. RCS pressure bounded by turbine trip event (15.2.3). Core power limited by reactor trip following turbine trip on high SG level.
15.1.3	Step Load Increase	Increase in steam flow (10 percent step) causes rise in core power by overcooling RCS. GDC 10, 15, 26. Same core-related parameters as 15.1.2.	MDNBR > 2.0 and not limiting. RCS pressure decreases during event. Core power to equilibrium without RPS trips.
15.1.4	Spurious Opening of SG Relief or Safety Valve	Failure of single steam dump valve overcools RCS, causing increase in core power for negative MTC. Event is at hot shutdown conditions for minimal stored energy in RCS. GDC 10, 15, 26. Shutdown margin — Min Moderator feedback — Max Power feedback — Min Peaking factor — Max	Not DNBR event. Pressure decreases in RCS. Demonstrates GDC 26.
15.1.5	Steam Piping Failure-MSLB	Steam line failure produces limiting overcooling of RCS resulting in positive reactivity insertion. Event is at hot shutdown conditions for minimal stored energy. Same core-related parameters as 15.1.4.	Analyzed to show no DNB per GDC 10. RCS pressure remains at or below nominal. Demonstrates GDC 27, 28 for limits to fuel damage and continued core cooling. Demonstrates GDC 35 for emergency cooling and reactivity control.

Increases in Heat Removal by the Secondary System

15.1.1	Decrease in Feedwater Temperature	BOUND: Less severe than 15.1.3.
15.1.2	Increase in Feedwater Flow	BOUND: MSLB in 15.1 series, other(s) for DNBR; SG level trip, no OT trip.
15.1.3	Increase in Steam Flow	BOUND: MSLB in 15.1 series, other(s) for DNBR; no OT trip.
15.1.4	SG Valve Opening	Not affected by reload: CHECK core/cycle parameters.
15.1.5	Steam Line Break (Small)	BOUND by MSLB to Condition II GDC's.
15.1.5	Main Steam Line Break	CHECK Condition II for no DNB - ANALYZE. RELAP5 transient analysis for offsite power available and offsite power unavailable conditions. PDQ/FLAM3/NOODLE calculations for power, peaking, and reactivity. LYNXT thermal hydraulic analyses for DNBR under both conditions.

Decreases in Heat Demand by the Secondary System

	Event Description/GDC's/Core Parameters	FSAR Consequences/Comments
15.2.2 Loss of External Load	Loss of steam load causes closure of turbine control valves. Reduced steam flow increases SG pressure and temperature, causing RCS heatup.	Bounded by more rapid reduction in steam flow for the turbine trip event (15.2.3). Environmental consequences determined by steam dump capacity with assumed defective fuel and primary-to-secondary leak rate.
15.2.3 Turbine Trip	Rapid closure of turbine stop valves on trip signal initiates SG and RCS heatups. Analysis considers loss of main FW and no reactor trip coincident with initiating event. No credit for AFW or pressure controls via steam dump or pressurizer. Demonstrates core protection and overpressure protection margins. GDC 10, 15, 26. Doppler power coefficient Max Moderator density coeff. Min Moderator density coeff. Max	Reactor trip on high pressurizer pressure plus pressurizer safety valves limit overpressure transient. DNBR increases throughout event. Core conditions remain within static OP and OT limit lines, so turbine trip is not a DNBR event.
15.2.4 MSIV Closure	Causes turbine trip.	Turbine trip.
15.2.5 Loss of Condenser Vacuum	Causes turbine trip with loss of steam dump to condenser.	Bounded by the analysis conditions addressed for the turbine trip.
15.2.6 Loss of AC Power	Loss of non-emergency AC power causes loss of main feedwater with loss of RCP power at reactor and turbine trips. GDC 10, 15, 26. Trip reactivity Min Delayed neutron fraction Max Fuel stored energy Max Decay heat Max	RCS flow coastdown from initiation is covered by the total loss of flow event. System statepoints remain within static overtemperature limits, so transient is not DNBR limiting. Demonstrates AFW capability for decay heat removal.

15.2.7	Loss of Normal Feedwater	Termination of feedwater reduces SG heat demand, leading to RCS heatup. Reactor trips on low-low SG level at about 1 min. Primary and secondary pressures controlled to safety valve setpoints. AFW initiated on trip signal with 1-min. startup time. Same core-related parameters as loss of AC power event.	RCS conditions remain within static overtemperature limits, and transient is not DNBR limiting. As analyzed, parallels loss of AC power event.
15.2.8	Feedwater System Pipe Break	Loss of MFW to all generators, followed by reverse blowdown of affected SG. Reactor trip on low SG level, and steam isolation on low steam line pressure. Excess heat removal post-trip until MSIV's close. Heatup following MSIV closure turned around by auxiliary FW removal of core heat. GDC 27: Fuel design limits GDC 28: Continued core cooling GDC 31: RCS brittle fracture GDC 35: RCS systems for core cooling Moderator density coeff. Max Doppler-only power coeff. Max Delayed neutron fraction Max Decay Max	Primary side temperatures and delta-T's remain within static limits, and pressure is limited at the safety valve setpoint. DNBR limits are not approached, nor are overpressure design limits for RCS and secondary. Primary issue is continued heat removal via safety injection and relief flows combined with emergency feedwater.

Decreases in Heat Removal by the Secondary System

15.2.2	Loss of Electrical Load	Bounded by turbine trip: CHECK core/cycle parameters.
15.2.3	Turbine Trip	Describe system response unaffected by reload; CHECK core/cycle/safety parameters.
15.2.4	MSIV Closure	Causes turbine trip.
15.2.5	Loss of Condenser Vacuum	Causes turbine trip.
15.2.6	Loss of Nonemergency AC Power	Not affected by reload; CHECK core/cycle parameters.
15.2.7	Loss of Normal Feedwater	Not affected by reload; CHECK core/cycle parameters.
15.2.8	Feedwater Line Break	CHECK core/cycle parameters.

Decreases in Reactor Coolant System Flow

		Event Description/GDC's/Core Parameters	FSAR Consequences/Comments
15.3.1	Partial Loss of RCS Flow	Coastdown of one RCP causes reduction in core flow at power. Reactor trip on low flow in one loop precedes minimum DNBR by 3-4 sec. GDC 10, 15, 26. Doppler power coeff. — Max Moderator density coeff. — Min Initial fuel temperature — Max Trip reactivity insertion — Min Delayed neutron fraction — Max	Partial loss of flow is bounded by total loss of flow (15.3.2) for latter analyzed to Condition II GDC's.
15.3.2	Total Loss of RCS Flow	Coastdown of all RCP's at power causes reduced core DNBR. Transient terminated by low voltage/underfrequency trip. GDC 10, 15, 26. Same core-related parameters as partial loss of flow event (15.3.1)	Limiting DNBR for loss of flow transients.
15.3.3	RCP Locked Rotor	RCP shaft seizure reduces flow in affected loop and through core at power. Reactor trip is on reduced flow. Analyzed for FSAR assuming DNB at start of transient and no pressurizer or steam dump pressure control. GDC 27, 28, 31. Doppler power coeff. — Max Moderator density coeff. — Min Initial fuel temperature — Max Trip reactivity insertion — Min Delayed neutron fraction — Max	Demonstrates no clad melting and overpressure within faulted condition limits.
15.3.4	RCP Shaft Failure	Same transient as locked rotor event except free spinning rotor permits greater reverse flow in affected loop.	

Decreases in Reactor Coolant System Flow

15.3.1	Partial Loss of Flow	Show core thermal response within GDC 10, 15; ANALYZE.
15.3.2	Complete Loss of Flow	Confirm DNBR for Condition II GDC's - ANALYZE.
15.3.3	Locked Rotor	Define criteria for Mk-BW and check - ANALYZE.
15.3.4	RCP Shaft Break	Refer to Locked Rotor.

Reactivity and Power Distribution Transients

		Event Description/GDC's/Core Parameters	FSAR Consequences/Comments
15.4.1	RCCA Withdrawal from Subcritical or Low Power Condition	Rod withdrawal causes positive reactivity insertion and local power excursion. FSAR aalysis considers insertion rate G.E. two sequential banks at max combined worth. Negative Doppler coefficient limits excursion through feedbackprior to termination by high flux trip. GDC 10 GDC 20 RPS automatic function GDC 25 RPS protection of fuel design for single failure in reactivity control Doppler temperature coeff. — Min Moderator temperature coeff. — Max+ Delayed neutron fraction — Max Prompt neutron lifetime — Max Reactivity insertion rate — Max	Doppler and reactor trip limit core power to well within steady state DNBR envelope. No RCS statepoint excursion. Need to confirm local power and peaking for fuel and clad limits.
15.4.2	RCCA Withdrawal at Power	Rod withdrawal at power causes power and heat flux increase from the nominal statepoint. GDC 10, 20, 25 Core thermal limits Doppler power coefficient — Min/Max Doppler temperature coeff. — Min/Max Moderator density coeff. — Min/Max Delayed neutron fraction — Max Initial fuel temperature — Min Reactivity insertion rate — Max Trip insertion reactivity — Min	Both high and low insertion rates are considered. Rapid rod withdrawal trips on high flux, with relatively higher post-trip overshoot. Thermal lag makes these statepoints less limiting. At low withdrawal rates, the trip statepoints "ride" the overtemperature limits. These produce the lowest DNBR's, but statepoints are always within static thermal limits. Transient peaking must be determined at the bounding levels of the core-related parameters.

15.4.3 Control Rod Misoperation

1. Dropped RCCA (one or more, same group) 2. Dropped RCCA Bank	Dropped RCCA causes negative reactivity insertion. For no negative flux rate trip, power limited by high flux trip or by reactivity feedback within trip limits. GDC 10, 20, 25. Radial peaking — Max Dropped RCCA worth — Max Doppler power coeff. — Min Moderator density coeff. — Min Control bank diff. worth — Max	Transient can be bounded by considering control bank differential worth equivalent to maximum dropped RCCA worth. Nominal condition is limiting system state point for DNB evaluation. Transient radial peaking is most significant consequential parameter.
3. Misaligned RCCA	Static misalignment produces no system transient. Steady state power distributions could produce peaking that challenges design limits. Same GDC's Same core-related parameters.	Most severe condition is for one RCCA fully inserted or for bank D fully inserted with one RCCA fully withdrawn. Analyzed at full power to confirm insertion limits and DNBR.
4. Single RCCA Withdrawal	Continuous withdrawal of single RCCA in manual control mode increases core power and coolant temperature. Increase in hot channel factor could challege DNBR limits. Similar to bank withdrawal, but local peaking more severe. Same GDC's Same core-related parameters with addition of fuel power census.	Analyzed for worst bank D rod withdrawn with bank D at insertion limit. Fuel power census used to determine number of rods in DNB at the overtemperature trip point.

15.4.4	Inactive RCP Startup	Improper startup of RCP at power adds colder coolant to core, causing positive reactivity insertion and power increase. Transient terminates with flux/flow trip at P-8 setpoint. GDC 10, 15, 26 Doppler power coefficient — Min Moderator density coeff. — Max Shutdown margin — Min	Tech Specs do not allow operation at N-1 loops. Reactivity insertion below that considered for rod withdrawal with less limiting system statepoints. Since overtemperature limits not reached, thermal limits are not challenged.
15.4.6	Boron Dilution	Dilution of boron during refueling, startup, and at power can cause positive reactivity insertion and power transient. Same GDC's. Initial boron concentration — Max Boron concentration for return to criticality — Max	Time-to-critical concentration calculated to show sufficient time for operator action during refueling and startup events. Reactivity for event at power is bounded by rod withdrawal events.
15.4.7	Improper Fuel Loading	Errors in core loading or fuel enrichment can cause core peaking beyond values calculated for correct loading.	

15.4.8 RCCA Ejection

Rapid ejection of RCCA causes fast reactivity insertion with adverse power distribution. High local power could produce fuel damage. Event terminated by either high flux or high flux rate trips. GDC 28, RG 1.77.

Ejected rod worth	Max
Initial local fuel temp.	Max
Moderator temperature coeff.	Min(abs)
Doppler power coeff.	Min(abs)
Hot channel factor	Max
Delayed neutron fraction	Min
Shutdown margin	Min

FSAR bounds with BOL and EOL cases at both zero and full power. Full power analyses produce highest pellet temperatures. Zero power results show highest peaking and clad temperatures. Melting confined to full power rod ejection, less than 10 percent of the pellet. Fewer than 10 percent of fuel rods in DNB.

Reactivity and Power Distribution Transients

15.4.1	RCCA Withdrawal from Subcritical or Low Power	Show core overpower and DNBR protection for GDC 10, 15; ANALYZE. Determine bounding core neutron power from CADD parametrics, consistent with bounding FSAR coefficients. Calculate peaking using FLAM3/NOODLE w/ no feedback other than minimum Doppler. LYNXT DNBR analysis with I.C. to be selected.
15.4.2	RCCA Withdrawal at Power	Demonstrate core protection for 75 pcm/s transient; evaluate core parameters; ANALYZE. RELAP5 transient analysis for 75 pcm/s. LYNXT thermal-hydraulic analysis from SCD I.C.s.
15.4.3	RCCA Misoperation	EVALUATE core/cycle parameters versus FSAR.
	1. Dropped RCCA	CHECK maximum worth, peaking, power for MDNBR at nominal P,T statepoint. Static analysis with FLAM3/PDQ for rod worths, peaking, OCD tilt, and power deficit at nominal P,T. Reactivity balance calculated to fix peak power, limited by OP or flux setpoints. Static LYNXT calculation at nominal P,T.
	2. Dropped RCCA Bank	Same (No tilt).
	3. Misaligned RCCA	Same. FLAM3 cycle-by-cycle calculation of static peaking at maximum misalignment.
	4. Single RCCA Withdrawal	CHECK DNB from P,T points at OTDT limit. PDQ/NOODLE/FLAM3 calculation of peaking. Fuel rod census for number of pins below DNB limit.

15.4.4	Inactive RCP Startup	CHECK core parameters.
15.4.6	Boron Dilution	CALCULATE and CHECK times to critical and insertion rate at power.
15.4.7	Fuel Loading	Not affected by reload.
15.4.8	RCCA Ejection	CHECK maximum worth and peaking vs FSAR; compare fuel parameters and core loadings. PDQ/NOODLE/FLAM3 calculations for maximum worth and peaking.

Increases in Reactor Coolant Inventory

		Event Description/GDC's/Core Parameters	FSAR Consequences/Comments
15.5.1	Inadvertent ECCS Operation at Power	ECCS operation causes boron injection, resulting in load/power mismatch. Pressure and temperature of RCS decrease to low pressure trip setpoint. GDC 10, 15, 26. Moderator density coeff. Min Doppler power coeff. Max	Delta-T limits never approached as DNBR increases throughout transient.
15.5.2	System Malfunction Causing RCS Inventory Increase	Boron dilution event.	Covered by boron dilution event.

Increases in Reactor Coolant Inventory

15.5.1	ECCS Actuation at Power	CHECK core/cycle parameters; not DNB limiting.
15.5.2	RCS Inventory Increase	Refer to Boron Dilution event.

Decreases in Reactor Coolant Inventory

		Event Description/GDC's/Core Parameters	FSAR Consequences/Comments
15.6.1	Inadvertent Opening of a Pressurizer Relief or Safety Valve	RCS pressure decreases at approximately constant temperature, leading to reduced DNBR. Trip on low pressure or overtemperature delta-T fixes point of minimum DNBR. GDC 10, 15, 26. Moderator density coeff. Min Doppler power coeff. Max(abs)	Core-related parameters have little effect upon limiting results. Key output of analysis is minimum DNBR at the trip point.
15.6.2	Steam Generator Tube Failure	Postulated rupture of single SG tube causes loss of RCS inventory, leading to reactor/turbine trips on low pressure or OTDT. Low pressure signal starts SI to maintain primary inventory. Steam flow from safety valves terminated for affected generator by operator action within 30 minutes. GDC 55 line isolation requirements 10 CFR 100 radiological consequences	Consequences bounded assuming 1 percent defective fuel and SG leakage prior to event. Not reload related.

Decreases in Reactor Coolant Inventory

15.6.1	PZR Valve Opening	CALCULATE and CHECK DNBR at setpoint for nominal flow and temperature. Static LYNXT DNBR at low pressure trip setpoint for nominal power, flow, temperature.
15.6.2	SG Tube Failure	Not reload related.
15.6.4	Small LOCA	EVALUATE
15.6.4	Large LOCA	ANALYZE

FIGURE 1. STEAM LINE BREAK - NEUTRON POWER FRACTION.

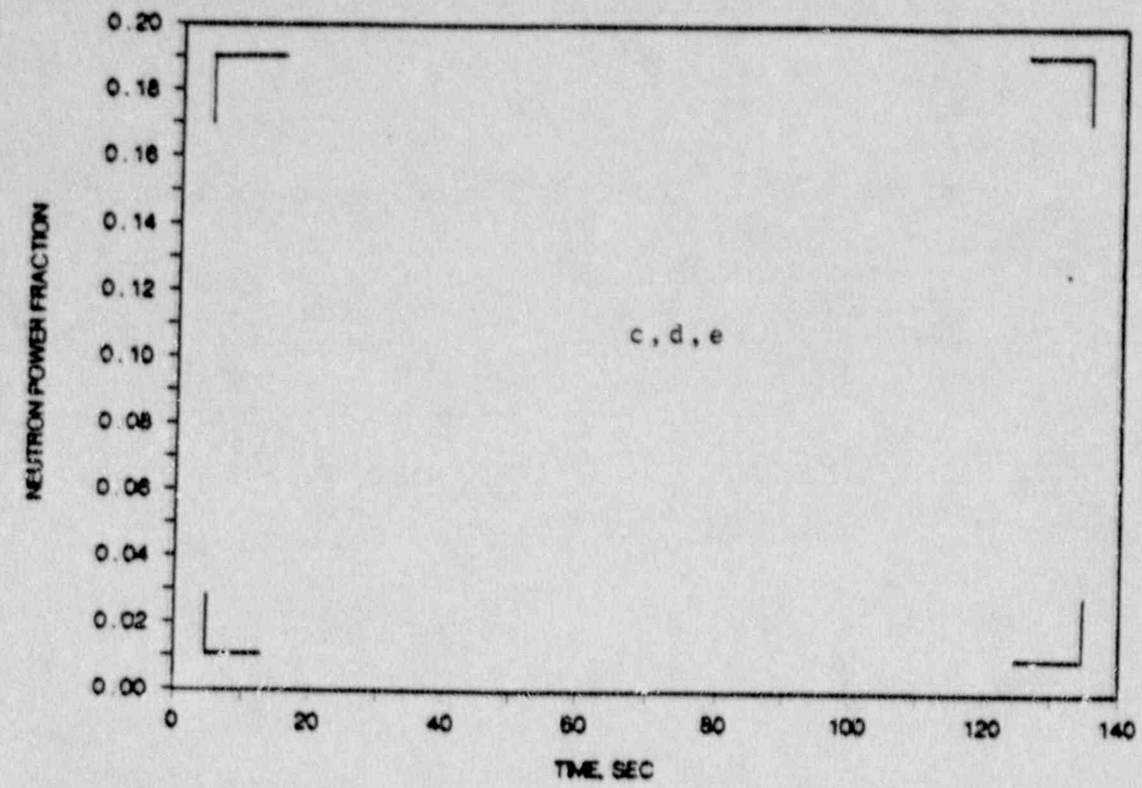


FIGURE 2. STEAM LINE BREAK - CORE AVERAGE TEMPERATURE.

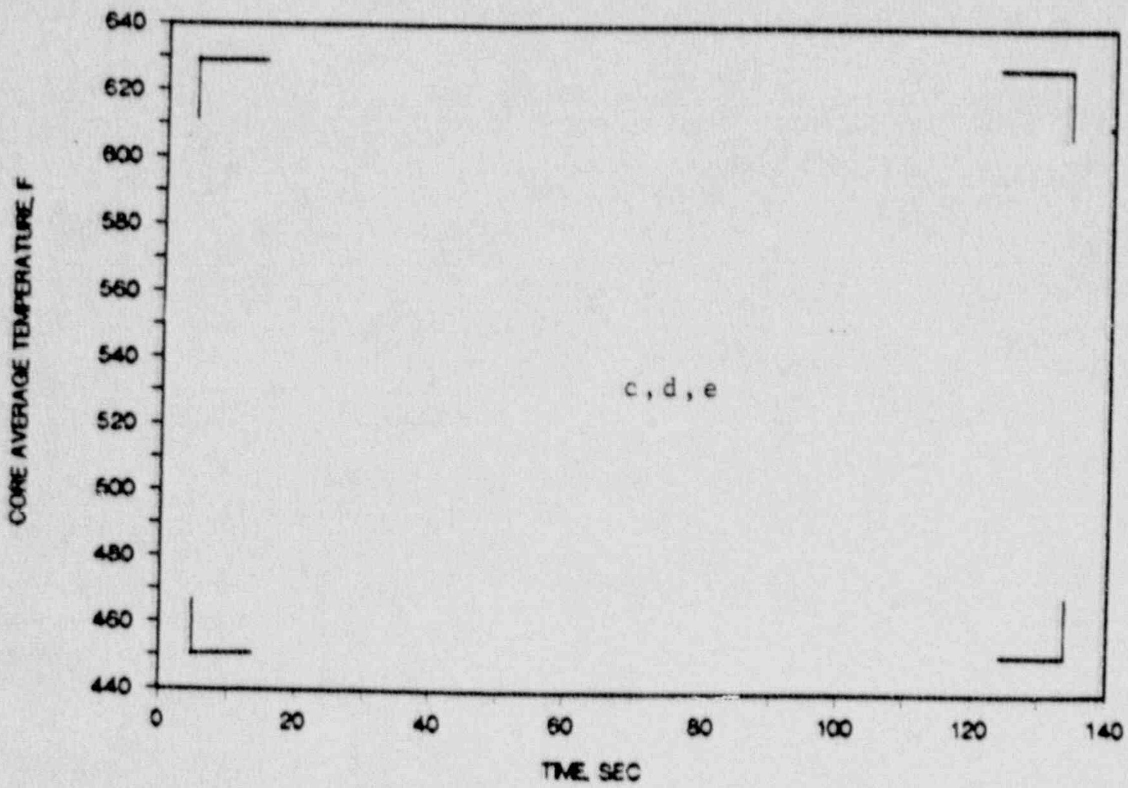


FIGURE 3. STEAM LINE BREAK - TRIPLE LOOP RV INLET TEMPERATURE.

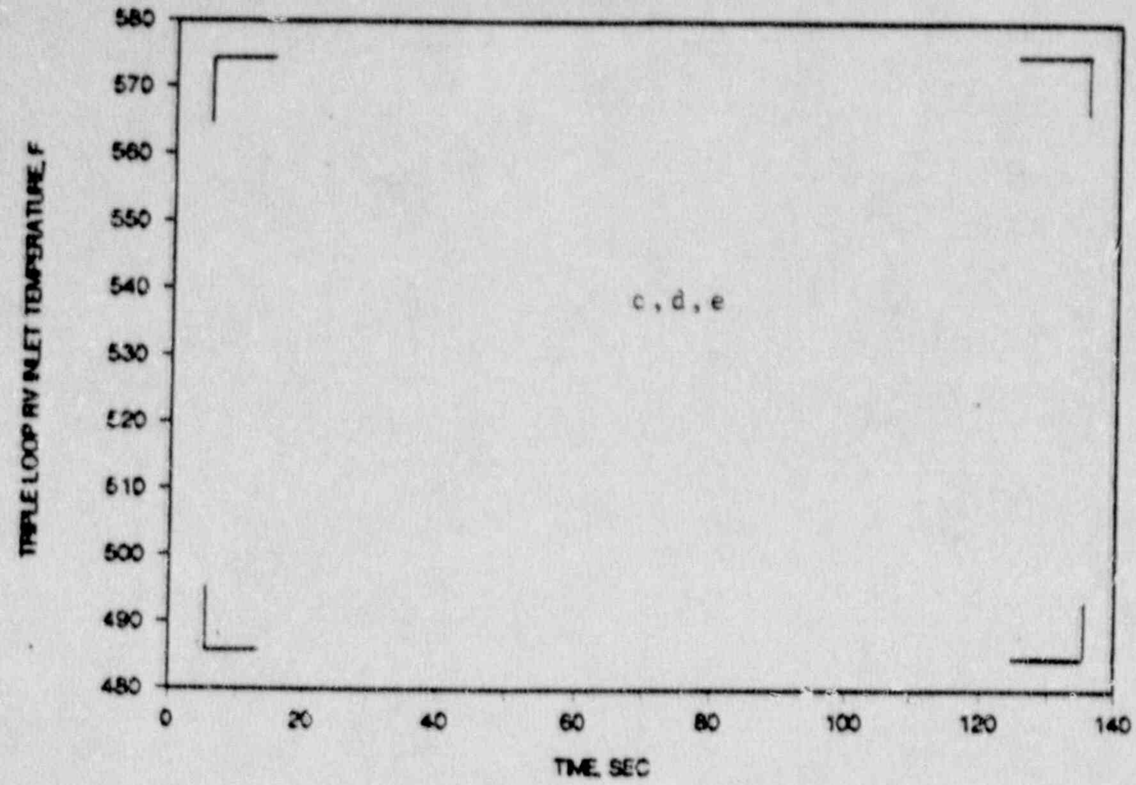


FIGURE 4. STEAM LINE BREAK - SINGLE LOOP RV INLET TEMPERATURE.

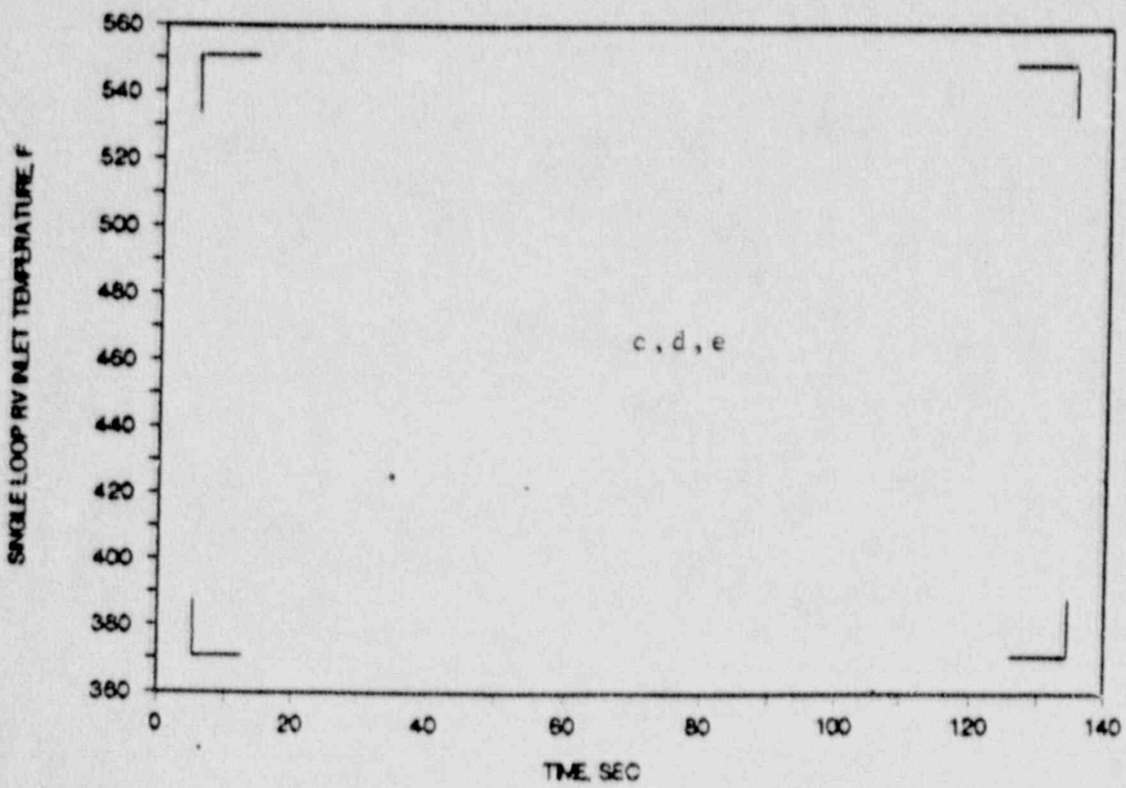


FIGURE 5. STEAM LINE BREAK - REACTIVITY.

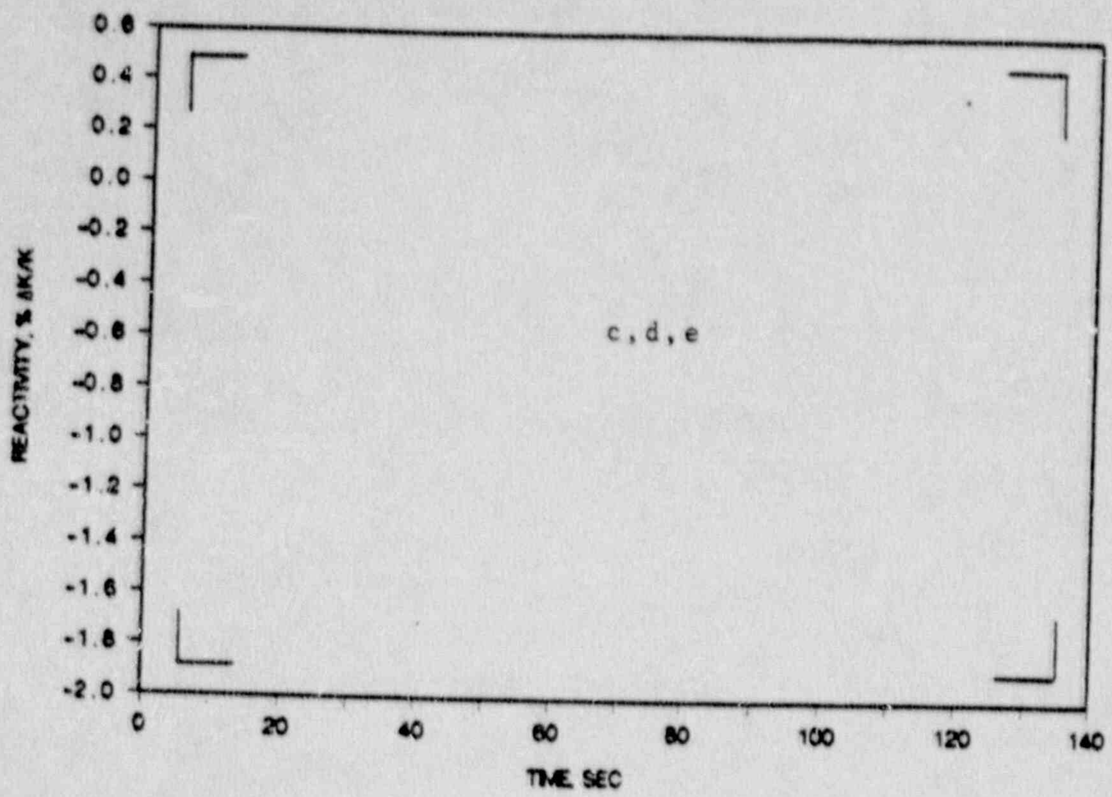


FIGURE 6. STEAM LINE BREAK - NEUTRON POWER FRACTION.

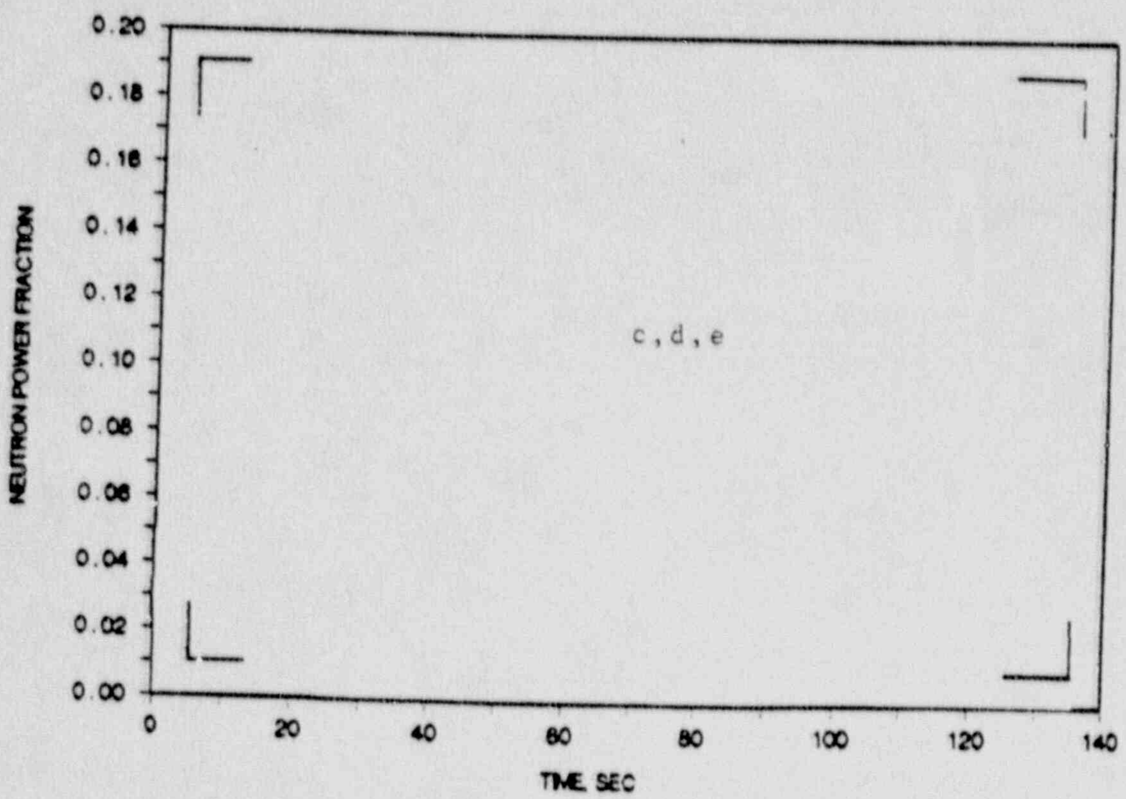


FIGURE 7. STEAM LINE BREAK - CORE AVERAGE TEMPERATURE.

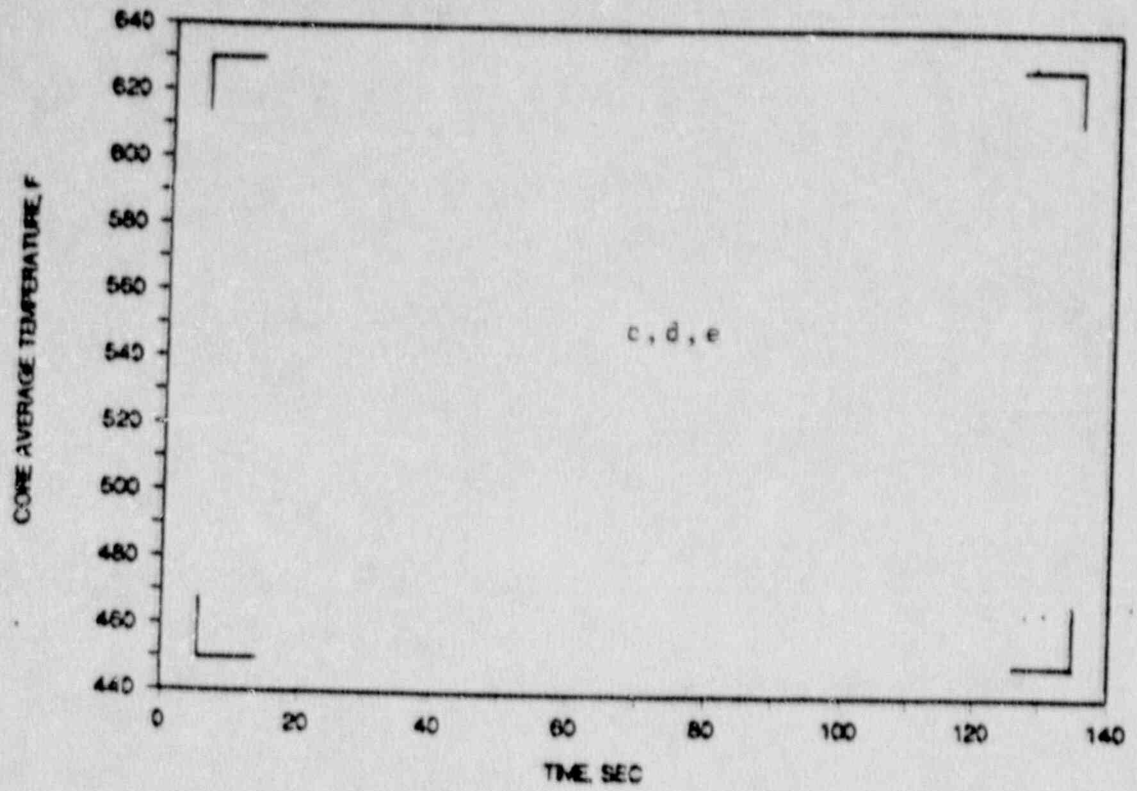


FIGURE 8. STEAM LINE BREAK - SINGLE LOOP RV INLET TEMPERATURE.

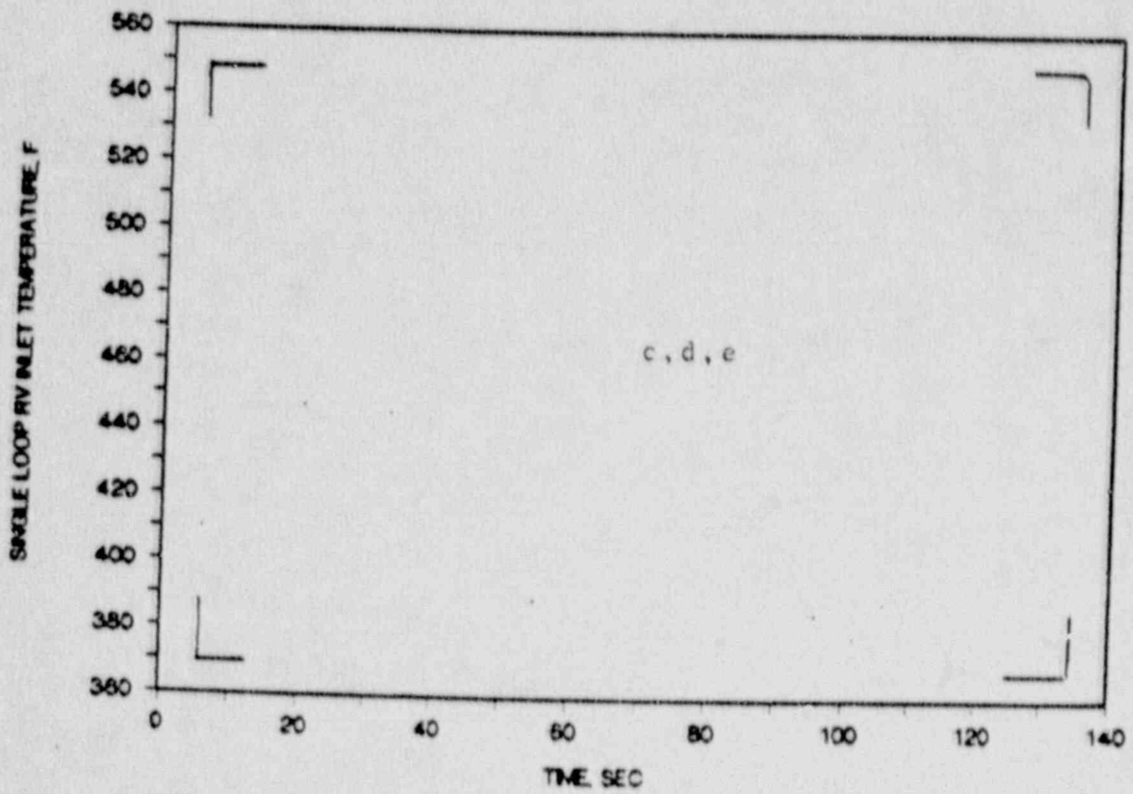


FIGURE 9. STEAM LINE BREAK - TRIPLE LOOP RV INLET TEMPERATURE.

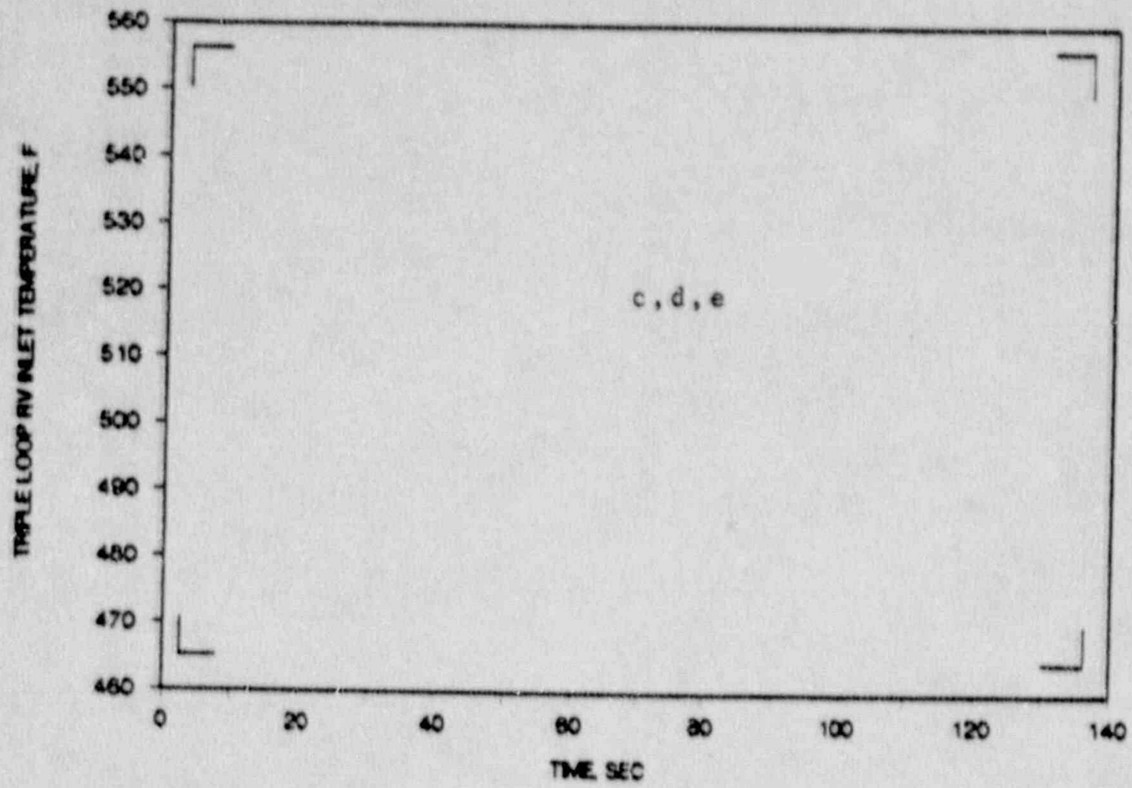
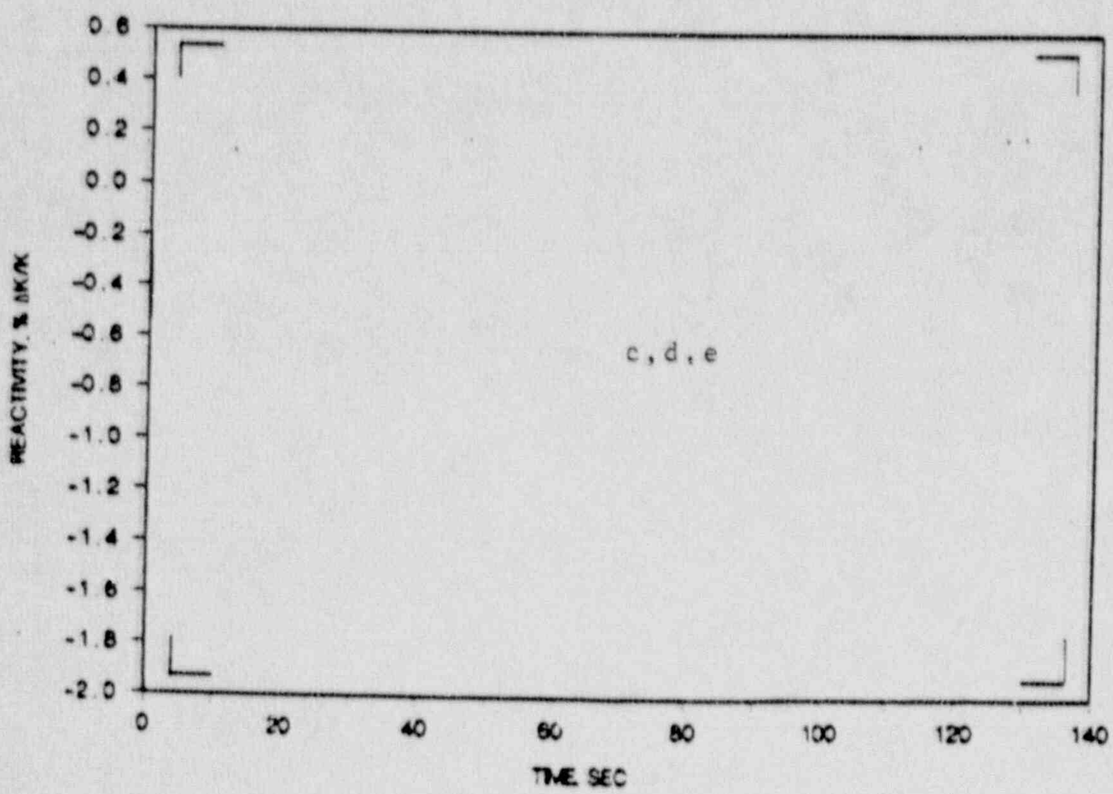


FIGURE 10. STEAM LINE BREAK - REACTIVITY.



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- RSG Plant Safety Analysis -

B&W Safety Analysis Methodology for
Recirculating Steam Generator Plants

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Key Words: Safety Analysis, Recirculating Steam Generator Plant, Transient Analysis Comparisons

ABSTRACT

The overall objective of this report is to demonstrate B&W's approach to the complete safety analysis for any recirculating steam generator plant. This topical report provides transient analysis comparisons to Safety Analysis Report results for several Westinghouse four loop plants operating at a power level of 3411 MWt. These comparisons show the RELAP5/MOD2 computer code with appropriate noding and plant parameters is a valid application model for recirculating steam generator plants.

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1. INTRODUCTION

Babcock & Wilcox has developed the capability and method to perform safety analysis calculations for recirculating steam generator (RSG) plants. This report presents the approach to plant modeling that will be used with the RELAP5 advanced system code to perform system analyses for non-LOCA transients, primarily to support fuel reload licensing for competitor-designed RSG plants. For reload safety applications, system analyses using RELAP5 will be complemented by core physics and thermal-hydraulics calculations by codes and methods reviewed and approved for those purposes on the B&W-designed plants. The specific transients analyzed will be based upon a review of all of the FSAR transients to determine which events could be affected by the fuel reload. The affected transients will be evaluated to identify those changes in the key safety parameters indicating the need for reanalysis. The purpose of this report is to show that this capability is appropriate and acceptable by presenting the results of a representative group of system analyses performed with RELAP5/MOD2-B&W that compare favorably with SAR results for the same transients.

The RSG methodology builds upon two decades of experience in safety analyses for the B&W-designed once-through steam generator pressurized water reactors. The safety analysis plant model uses a B&W version of the RELAP5 advanced system code, designated RELAP5/MOD2-B&W, to calculate reactor system response for non-LOCA transients. This report describes the plant model, which represents that to be used for plant-specific analyses, and the comparative analyses used to validate B&W's application of RELAP5 to RSG plant safety analysis. The comparative analyses benchmark the B&W model to four sets of safety analysis report results for plants of the four-loop Westinghouse design. The benchmarks demonstrate that, for the same or equivalent major assumptions and inputs, the B&W model produces results consistent with those of methods that have been reviewed and accepted. The success of the comparative analyses shows that B&W can model and analyze RSG plant system response for the full spectrum of safety analysis transients.

The major sections of this report, Sections 4 through 7, follow a brief outline of the approach to the comparative analyses presented in Section 2 and the summary and conclusions presented in Section 3. A discussion of the full scope of safety analysis transients is included in Section 4, leading to the selection of the transients in each category selected for the comparative analyses. Plant initial conditions and assumptions for the safety analysis reports and as used for the B&W model are described in Section 5. The plant model, based upon the Westinghouse four-loop 3411 MWt design, is presented in detail in Section 6. Finally, the five subsections of Section 7 discuss the safety analysis report data and the results of the RELAP5 comparative analyses for each of the transients. In addition, analyses are presented to show the performance of the plant model as it will typically be applied for reload safety analyses. Actual applications of the B&W safety analysis methodology for recirculating steam generator plants will be reported on a plant specific basis.

2. APPROACH

The approach chosen to demonstrate that the B&W application of RELAP5 to plant safety analyses is valid is to compare results obtained using the RELAP5 code and B&W-developed model to analyses presented in safety analysis reports for plants with recirculating steam generators. The various industry and NRC programs aimed at code assessment through comparative analyses have underscored the difficulties that can be encountered in bringing analyses performed by a diversity of analysts using different codes and models to a common point of comparison. In developing the comparative study presented in this report, the plant designs and features, the significant phenomena, and the key parameters and assumptions have been investigated in detail. The resulting comparative analyses show that the safety analysis plant model and RELAP5 code to be used by B&W can produce virtually the same results for a representative set of transients, given the same or equivalent analysis assumptions and modeling methods, as models and methods already reviewed and approved for the same applications.

The safety analysis report (SAR) data used as the bases for the comparative analyses were taken from four sources: the Catawba, McGuire, and Trojan FSARs and the Westinghouse RESAR. The purpose was to show a spectrum of results and comparisons for the Westinghouse four-loop, 3411 MWt design. This allowed the B&W analyses to show appropriate sensitivity to the differences in design or inputs. Section 5 presents a comparison of plant parameters for the four plants listed above and the B&W model. All of these plants are four loop plants operating at a power level of 3411 MWt, however the Catawba/McGuire plants have a preheat Model D recirculating steam generator and the Trojan plant has a Type 51 recirculating steam generator.

The run matrix for the comparative analyses considered five limiting or representative safety analysis transients: (1) 75 pcm/sec rod withdrawal, (2) four-pump coastdown, (3) turbine trip, (4) locked rotor event, and (5) main steam line break. The first four are full power transients. The steam line break is analyzed, for the most conservative result, from low power conditions. The bases for the selection of these transients is presented in Section 4. In order to provide direct comparison of the SAR results to each other and to the B&W calculations, the SAR curves were digitized and replotted to common starting values -- normalized -- for each of the key parameters. For completeness, the SAR curves are presented without normalizing for each transient as it is discussed.

3. SUMMARY AND CONCLUSIONS

The comparative analyses presented in this report demonstrate that the method to be used by B&W to apply RELAP5 in safety analyses of recirculating steam generator (RSG) plants is acceptable. A representative plant model, based upon the Westinghouse four-loop design, was used to perform RELAP5 analyses for a group of limiting or representative safety analysis transients as presented in plant safety analysis reports. The RELAP5 analyses incorporate the same or equivalent assumptions and boundary conditions as those performed for the SAR's, and the results are directly comparable to the SAR results. The validity of the B&W model and application is demonstrated by the agreement of the results obtained using essentially identical inputs and assumptions with those produced by methods that have been reviewed and accepted for similar applications.

The safety analysis report data used in these comparisons were taken from four sources: the Catawba, McGuire, and Trojan FSAR's (References 1-4) and the Westinghouse RESAR (Reference 5). The comparison to this set of plant safety analyses demonstrates the ability of the B&W method to reflect the impacts of changes in the significant analysis inputs, assumptions, and boundary conditions. In addition, the comparison shows consistency in the major results with methods that have been reviewed and accepted for the same applications. The diversity of the transients and conditions presented shows that the B&W approach for using RELAP5 is broadly applicable to licensing cases.

The five transients chosen for the comparison are cases limiting or representative based on review of all the transients of

Regulatory Guide 1.70 and the American Nuclear Society (ANS) classification of plant conditions. The scope of the transients chosen and analyzed indicates an ability to evaluate any of the non-LOCA safety analysis transients of Regulatory Guide 1.70.

All of the analyses were done using nominal plant initial conditions for system parameters except that the core power was taken as 102 percent. For the DNB calculations, conservative steady state errors were used. The reactor protection system setpoints used for this study included the maximum steady state errors and the maximum trip delay times. The comparison of the results to those for plants with a range of initial conditions showed that the safety analysis results are not significantly sensitive to minor plant parameter variations.

The RELAP5/MOD2 code version used for these analyses is a version developed at B&W for both LOCA and non-LOCA applications, based upon the EG&G Cycle 36.02 of RELAP5/MOD2. The input plant model, representing the Westinghouse four loop, 3411 MWt design, included detailed noding of the primary coolant system and the recirculating steam generators. Two different reactor vessel models were used; one of the vessel models included split core noding for the steam line break transient. The approach to the plant modeling was derived from B&W experience in similar analyses for B&W-designed plants and for integral system tests, from models of the four-loop design reported in the open literature, and from safety analysis report information.

Analyses were performed for the selected transients using the B&W-developed plant model. For each transient, comparison cases were run using inputs and assumptions derived from the four separate safety analysis reports. The results of these cases were compared to those presented in the SAR's and found to be in good agreement with the SAR results. Moreover, the RELAP5 analyses exhibited the same sensitivity in results to major input assumptions as shown in the four reference analyses. This demonstrates that the proposed application of RELAP5 to safety analyses of reactor systems with recirculating steam generators is consistent with methods and results that have been reviewed and approved for those plants.

The B&W application of RELAP5 to system safety analyses as presented in this report, combined with the appropriate core thermal-hydraulics and physics methods, forms a suitable and acceptable basis for evaluating non-LOCA transients as required in fuel reload safety reviews. The comparison cases and application analyses show the ability of the proposed method to represent the effects of significant changes to core and system conditions. The RELAP5 modeling and inputs developed for this study indicate B&W's familiarity with the design and performance of recirculating steam generator plants and with the sensitivity of the plant safety analyses to a wide scope of postulated transients and assumptions. B&W has demonstrated the capability to review plant safety analysis reports with respect to the significant safety parameters and evaluate or reanalyze accordingly those non-LOCA transients that may be affected by reload fuel.

4. TRANSIENT SELECTION

This section presents the bases for selecting five transients as representative and bounding transients for all of the Regulatory Guide 1.70 Representative Initiating Events to be considered in safety analyses for fuel reload applications. This selection is based upon review of reference safety analysis reports to identify the representative transient or transients in four categories. The five transients are:

1. Rod Withdrawal (full power)
2. Four Pump Coastdown (full power)
3. Turbine Trip (full power)
4. Locked Rotor (full power)
5. Steam Line Break (low power).

In the following paragraphs, the regulatory requirements related to the initiating events and types of transients to be considered in safety analyses are described. The relationship between the selected transients and the regulatory requirements and transient categories is discussed.

4.1. Regulatory Requirements

Regulatory Guide 1.70, Revision 3, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition provides a listing of representative initiating events to be analyzed in Section 15 of the Safety Analysis Report (SAR). Table 15-1 from Regulatory Guide 1.70 is shown as Table 4.1.1 in this report.

It is important to note the list of initiating events is broken down into eight major categories as follows:

1. Increase in Heat Removal by the Secondary System.
2. Decrease in Heat Removal by the Secondary System.
3. Decrease in Reactor Coolant System Flow Rate.
4. Reactivity and Power Distribution Anomalies.
5. Increase in Reactor Coolant Inventory.
6. Decrease in Reactor Coolant Inventory.
7. Radioactive Release from a Subsystem or Component.
8. Anticipated Transients Without Scram.

Of these major categories, 5 and 6 are more related to LOCA than the non-LOCA safety analysis. Category 7 is a radiological release category that is not transient related. Category 8 is a specialized category that is usually addressed as a separate topical report and not as a part of the SAR Chapter 15 transient analysis.

Since 1970, the American Nuclear Society (ANS) classification of plant conditions has been used which divides plant conditions into four categories in accordance with anticipated frequency of occurrence and potential radiological consequences to the public. The four categories are as follows:

1. Condition I: Normal Operation and Operational Transients
2. Condition II: Faults of Moderate Frequency
3. Condition III: Infrequent Faults
4. Condition IV: Limiting Faults

The basic principle applied in relating design requirements to each of the conditions is that most probable occurrences should yield the least radiological risk to the public and those extreme situations having the potential for the greatest risk to the public shall be those least likely to occur. Where applicable, reactor trip system and engineered safeguards functioning is assumed, to the extent allowed by considerations such as the single failure criterion, in fulfilling this principle.

Condition I - Normal Operation and Operational Transients

Condition I occurrences are those which are expected frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Inasmuch as Condition I occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III, and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to adverse conditions which can occur during Condition I operation.

A typical list of Condition I events is listed below:

1. Steady state and shutdown operations

   a. Power operation (> 5 to 100 percent of rated thermal power).

   b. Startup ($K_{eff} \geq 0.99$ to ≤ 5 percent of rated thermal power).

   c. Hot standby (subcritical, Residual Heat Removal System isolated).

d. Hot shutdown (subcritical, Residual Heat Removal System in operation).

e. Cold shutdown (subcritical, Residual Heat Removal System in operation).

f. Refueling.

2. Operation with permissible deviations

Various deviations which may occur during continued operation as permitted by the plant Technical Specifications must be considered in conjunction with other operational modes. These include:

a. Operation with components or systems out of service (such as power operation with a reactor coolant pump out of service).

b. Radioactivity in the reactor coolant, due to leakage from fuel with clad defects

   1. Fission products
   2. Corrosion products
   3. Tritium

c. Operation with steam generator leaks up to the maximum allowed by the Technical Specifications.

d. Testing as allowed by technical specifications.

3. Operational transients

a. Plant heatup and cooldown (up to 100°F/hour for the Reactor Coolant System; 200°F/hour for the pressurizer during cooldown and 100°F/hour for the pressurizer during heatup).

b. Step load changes (up to ±10 percent).

c. Ramp load changes (up to ±5 percent/minute).

d. Load rejection up to and including design full load rejection transient.

Condition II - Faults of Moderate Frequency

These faults, at worst, result in the reactor trip with the plant being capable of returning to operation. By definition, these faults (or events) do not propagate to cause a more serious fault, i.e., Condition III or IV events. In addition, Condition II events are not expected to result in fuel rod failures or reactor coolant system or secondary system overpressurization.

For the purposes of this report, the following faults are included in this category:

1. Feedwater system malfunctions that result in an increase in feedwater temperature.

2. Feedwater system malfunctions that result in an increase in feedwater flow.

3. Excessive increase in secondary steam flow.

4. Inadvertent opening of a steam generator relief or safety valve.

5. Loss of external electrical load.

6. Turbine trip.

7. Inadvertent closure of main steam isolation valves.

8. Loss of condenser vacuum and other events resulting in turbine trip.

9. Loss of nonemergency AC power to the station auxiliaries.

10. Loss of normal feedwater flow.

11. Partial loss of forced reactor coolant flow.

12. Uncontrolled rod cluster control assembly bank withdrawal from a subcritical or low power startup condition.

13. Uncontrolled rod cluster control assembly bank withdrawal at power.

14. Rod cluster control assembly misalignment (dropped full length assembly, dropped full length assembly bank, or statically misaligned full length assembly).

15. Startup of an inactive reactor coolant pump at an incorrect temperature.

16. Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant.

17. Inadvertent operation of the emergency core cooling system during power operation.

18. Chemical and volume control system malfunction that increases reactor coolant inventory.

19. Inadvertent opening of a pressurizer safety or relief valve.

Condition III - Infrequent Faults

By definition Condition III occurrences are faults which may occur very infrequently during the life of the plant. They will be accommodated with the failure of only a small fraction of the fuel rods although sufficient fuel damage might occur to preclude resumption of the operation for a considerable outage time. The release of radioactivity will not be sufficient to interrupt or restrict public use of those areas beyond the exclusion radius. A Condition III fault will not, by itself, generate a Condition IV fault or result in a consequential loss of function of the reactor coolant system or containment barriers. For the purposes of this report the following faults are included in this category:

1. Steam system piping failure (minor).

2. Complete loss of forced reactor coolant flow.

3. Rod cluster control assembly misoperation (single rod cluster control assembly withdrawal at full power).

4. Inadvertent loading and operation of a fuel assembly in an improper position.

5. Loss of coolant accidents resulting from a spectrum of postulated piping breaks with the reactor coolant pressure boundary (small break).

6. Radioactive gas waste system leak or failure.

7. Radioactive liquid waste system leak or failure.

8. Postulated radioactive releases due to liquid tank failures.

9. Spent fuel cask drop accidents.

Condition IV - Limiting Faults

Condition IV occurrences are faults which are not expected to take place, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. They are the most drastic which must be designed against and represent limiting design cases. Condition IV faults are not to cause a fission product release to the environment resulting in an undue risk to public health and safety in excess of guideline values of 10 CFR 100. A single Condition IV fault is not to cause a consequential loss of required functions of systems needed to cope with the fault including those of the emergency core cooling system and the containment. For the purposes of this report, the following faults have been classified in this category:

1. Steam system piping failure (major).

2. Feedwater system pipe break.

3. Reactor coolant pump shaft seizure (locked rotor).

4. Reactor coolant pump shaft break.

5. Spectrum of rod cluster control assembly ejection accidents.

6. Steam generator tube failure.

7. Loss of coolant accidents resulting from the spectrum of postulated piping breaks with the reactor coolant pressure boundary (large break).

8. Design basis fuel handling accidents.

4.2. Fuel Reload Evaluation Transients

The scope of analyses needed to support reload licensing will generally be narrower than that required for the original plant licensing. That is, the complete safety analysis will not need to be repeated. The fuel reload safety analyses need only consider those transients affected by changes to plant performance that are fuel reload-related. The limiting transient or transients within each category typically remain the limiting cases. With these points in mind, safety analysis reports were reviewed for four separate plants (Trojan, the Westinghouse RESAR, Catawba, and McGuire) to identify the limiting or representative transients in four categories. These transients were chosen for comparative analysis using the B&W-developed plant model and the RELAP5 system code. The objective in selecting the transients was twofold: Agreement between the results produced by the B&W model and those documented in the four separate safety analysis reports would show consistency with established methods for diverse cases. Second, successful comparison to five different transients would indicate that the B&W method is consistent with the original licensing methods in identification of the limiting transients.

The five transients selected by B&W represent a bounding and representative set for the Regulatory Guide 1.70 (Reference 12) list. The five are distributed in the first four major categories as follows:

1. Category 1 - Increase in Heat Removal by the Secondary System

   a. Steam line break transient

2. Category 2 - Decrease in Heat Removal by the Secondary System

   a. Turbine trip transient

3. Category 3 - Decrease in Reactor Coolant System Flow Rate

   a. Four pump coastdown transient
   b. Locked rotor transient

4. Category 4 - Reactivity and Power Distribution Anomalies

   a. Rod withdrawal transient

The five transients selected are distributed in the four ANS frequency of occurrence categories as follows:

1. Condition I - Normal Operation and Operational Transients

   a. None

2. Condition II - Faults of Moderate Frequency

   a. Turbine trip
   b. Rod withdrawal

3. Condition III - Infrequent Faults

   a. Four pump coastdown

4. Condition IV - Limiting Faults

   a. Steam line break
   b. Locked rotor

The analysis of the five transients selected by B&W shows any of the Regulatory Guide 1.70 transients could have been analyzed using appropriate noding, analysis assumptions, and plant parameters.

Table 4.1.1. Representative Initiating Events to be Analyzed in Sections 15.X.X of the SAR

1. Increase in Heat Removal by the Secondary System

1.1 Feedwater system malfunctions that result in a decrease in feedwater temperature.

1.2 Feedwater system malfunctions that result in an increase in feedwater flow.

1.3. Steam pressure regulator malfunction or failure that results in increasing steam flow.

1.4 Inadvertent opening of a steam generator relief or safety valve.

1.5 Spectrum of steam system piping failures inside and outside of containment in a PWR.

2. Decrease in Heat Removal by the Secondary System

2.1 Steam pressure regulator malfunction or failure that results in decreasing steam flow.

2.2 Loss of external electric load.

2.3 Turbine Trip (stop valve closure).

2.4 Inadvertent closure of main steam isolation valves.

2.5 Loss of condenser vacuum.

2.6 Coincident loss of on-site and external (off-site) a.c. power to the station.

Table 4.1.1. (Cont'd.)

2.7 Loss of normal feedwater flow.

2.8 Feedwater piping break.

3. Decrease in Reactor Coolant System Flow Rate

3.1 Single and multiple reactor coolant pump trips.

N.A. 3.2 BWR recirculation loop controller malfunctions that result in decreasing flow rate.

3.3 Reactor coolant pump shaft seizure.

3.4 Reactor coolant pump shaft break.

4. Reactivity and Power Distribution Anomalies

4.1 Uncontrolled control rod assembly withdrawal from a subcritical or low power startup condition (assuming the most unfavorable reactivity conditions of the core and reactor coolant system), including control rod or temporary control device removal error during refueling.

4.2 Uncontrolled control rod assembly withdrawal at the particular power level (assuming the most unfavorable reactivity conditions of the core and reactor coolant system) that yields the most severe results (low power to full power).

4.3 Control rod maloperation (system malfunction or operator error), including maloperation of part length control rods.

Table 4.1.1. (Cont'd.)

4.4 Startup of an inactive reactor coolant loop or recirculating loop at an incorrect temperature.

N.A. 4.5 A malfunction or failure of the flow controller in a BWR loop that results in an increased reactor coolant flow rate.

4.6 Chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant of a PWR.

4.7 Inadvertent loading and operation of a fuel assembly in an improper position.

4.8 Spectrum of rod ejection accidents in a PWR.

N.A. 4.9 Spectrum of rod drop accidents in a BWR.

5. Increase in Reactor Coolant Inventory

5.1 Inadvertent operation of ECCS during power operation.

5.2 Chemical and volume control system malfunction (or operator error) that increases reactor coolant inventory.

N.A. 5.3 A number of BWR transients, including items 2.1 through 2.6 and item 1.2.

6. Decrease in Reactor Coolant Inventory

6.1 Inadvertent opening of a pressurizer safety or relief valve in a PWR or a safety or relief valve in a BWR.

Table 4.1.1. (Cont'd.)

6.2 Break in instrument line or other lines from reactor coolant pressure boundary that penetrate containment.

6.3 Steam generator tube failure.

N.A. 6.4 Spectrum of BWR steam system piping failures outside of containment.

6.5 Loss-of-coolant accidents resulting from the spectrum of postulated piping breaks within the reactor coolant pressure boundary, including steam line breaks inside of containment in a BWR.

N.A. 6.6 A number of BWR transients, including items 2.7, 2.8, and 1.3.

7. Radioactive Release from a Subsystem or Component

7.1 Radioactive gas waste system leak or failure.

7.2 Radioactive liquid waste system leak or failure.

7.3 Postulated radioactive releases due to liquid tank failures.

7.4 Design basis fuel handling accidents in the containment and spent fuel storage buildings.

7.5 Spent fuel cask drop accidents.

8. Anticipated Transients Without Scram

8.1 Inadvertent control rod withdrawal.

Table 4.1.1. (Cont'd.)

8.2 Loss of feedwater.

8.3 Loss of a.c. power.

8.4 Loss of electrical load.

8.5 Loss of condenser vacuum.

8.6 Turbine trip.

8.7 Closure of main steam line isolation valves.

N.A. - Not applicable to a PWR.

5. PLANT INITIAL CONDITIONS

For all of the accident analyses in this report, nominal values of initial conditions were used for system parameters except the core power was 102% and for the DNB calculations, conservative steady state errors were used. The reactor protection system setpoints used in this report included the maximum steady state errors and the maximum trip delay times. A single set of initial conditions was used for all transient analyses to simplify case-to-case comparison and for convenience. Where specific conservative initial conditions can have an adverse effect on time to trip, valve opening, and so forth, the effect magnitude can be accounted for in the trip value, valve opening setpoint, etc. This method also minimizes double accounting for the same error.

5.1. Core and Plant Parameters

Table 5.1.1 lists the values of the pertinent parameters for the B&W model and the four recirculating steam generator plants used for these comparisons. This listing of input parameters and initial conditions for transients and accidents is consistent with the requirements of Regulatory Guide 1.70 Table 15-2.

When the DNBR was calculated the following conservative steady state errors were assumed in the DNBR analysis:

1. Core power: + 2 percent allowance for calorimetric error.

2. Average reactor coolant temperature: +4 F allowance for controller system temperature dead band and measurement error.

3. Pressurizer pressure: +30 pounds per square inch (psi) allowance for steady state fluctuations and measurement error.

5.2. Reactor Protection System Setpoints

Table 5.2.1 lists the limiting reactor protection setpoints and total trip delay time used in this transient analysis for the B&W model and the four recirculating steam generator plants used for these comparisons. The trip delay is defined as the total time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. The difference between the limiting trip point assumed for the analysis and the nominal trip point represents an allowance for instrumentation channel error and setpoint error. The nominal trip setpoints are specified in the Plant Technical Specifications.

Not all reactor protection system setpoints and trip delays are listed in Table 5.2.1, however, all setpoints and trip delays used in this analysis are listed.

TABLE 5.1.1.

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS

PARAMETER		FULL POW EVAL MOD	LOW POW EVAL MOD	CATAWBA	MCGUIRE	RESAR	TROJAN
NEUTRON POWER (10+6 WATTS)		3479 (102%)	17 (0.5%)	3411 (100%)	3411 (100%)	3411 (100%)	3411 (100%)
MODERATOR TEMP COEFFICIENT (10-5 ΔK/K/F)	BOL	0 TO 7.0	--	0.0	0 TO 5.0	0.0	0.0
	EOL	--	--	--	--	--	--
MODERATOR DENSITY COEFFICIENT	BOL	--	--	--	--	--	--
	EOL	--	(1)	--	--	--	--
DOPPLER COEFF (10-5 ΔK/K/%)	BOL(MIN)	-6.5	--	-6.0	-6.0	-6.0	-6.5
	EOL(MAX)	-12.5	(2)	-12.5	-12.5	-12.5	-12.5
EFFECTIVE NEUTRON LIFETIME (10-6 SEC)	BOL	19.40	19.40	19.40	19.40	19.40	19.40
	EOL	18.10	18.10	18.10	18.10	18.10	18.10
DELAYED NEUTRON FRACTION	BOL	0.0075	0.0075	0.0075	0.0075	0.0075	0.0075
	EOL	0.0044	0.0044	0.0044	0.0044	0.0044	0.0044
AVERAGE HEAT FLUX (10+3 BTU/HR-FT2)		198.3	0.972	197.2	197.2		189.8
MAXIMUM HEAT FLUX (10+3 BTU/HR-FT2)		440.9	1.567				
MINIMUM DNBR		2.38(3)	--				
AXIAL POWER DISTRIBUTION		1.68	1.68				
RADIAL POWER DISTRIBUTION		1.38	1.38				
CORE COOLANT FLOW RATE (10+6 LB/HR)		132.5	132.8	138.3	146.1		132.7
CORE COOLANT INLET TEMP (F)		553.28	555.78	560.6	559.2		552.5
CORE AVERAGE COOLANT TEMP (F)		586.05	555.91	590.8	590.1	592.0	584.9

(TABLE 5.1.1. Continued)

CORE COOLANT EXIT TEMP (F)	618.8	556.03				617.2
HOT ASSEMBLY COOLANT EXIT TEMPERATURE (F)	639.53	--				
AVERAGE FUEL TEMP (F)	1162.39	558.2				
MAXIMUM FUEL CENTERLINE TEMP (F)	3230.7	561.48				
REACTOR COOLANT SYSTEM INVENTORY (10+6 LB)	0.5040	0.4987				
COOLANT VOLUME IN PRESSURIZER (FT3)	1080	500	1080	1175	1080	1080
REACTOR COOLANT PRESSURE (PSIA)	2250	2250	2250	2250	2250	2250
STEAM FLOW RATE (10+6 LB/HR)	15.29	0.059	15.15	15.14		15.07
STEAM PRESSURE (PSIA)	874.76	1080	1000	1000		910
STEAM QUALITY	1.0	1.0				
FEEDWATER FLOW RATE (10+6 LB/HR)	15.36	0.075				
FEEDWATER TEMPERATURE (F)	440	79.8	440	440	440	440
TOTAL CONTROL ROD WORTH (%ΔK/K)	4	--	4	4	4	4

(1) FUNCTION OF MODERATOR DENSITY

(2) FUNCTION OF POWER LEVEL

(3) BASED ON RELAP5 CONTROL VARIABLE ALGORITHM

TABLE 5.2.1.

TRIP POINTS AND TIME DELAYS TO TRIP

TRIP FUNCTION		EVAL MOD	CATAWBA	MCGUIRE	RESAR	TROJAN
HIGH NEUTRON	SETPOINT	118	118	118	118	118
FLUX TRIP (*)	DELAY	0.5 SEC	0.5 SEC	0.5 SEC	0.5 SEC	0.5 SEC
OVERTEMPERATURE	K41	1.129				
ΔT TRIP (F)	K42	0.01702/F				
	TAU41	21.0 SEC				
	TAU42	3.0 SEC				
	TAVGO	584.8 F				
	K43	86 E-5 PSI				
	PO	2250 PSIA				
	DELAY	6.0 SEC				
HIGH PRESSURIZER	SETPOINT	2425	2425	2425	2425	2425
PRESS TRIP (PSIA)	DELAY	2 SEC	2 SEC	2 SEC	2 SEC	2 SEC
LOW REACTOR	SETPOINT	87	87	87	87	87
COOLANT FLOW	DELAY	1 SEC	1 SEC	1 SEC	1 SEC	1 SEC
TRIP (*)						
UNDERVOLTAGE	SETPOINT	68	68	68	68	68
TRIP(*)	DELAY	1.5 SEC	1.5 SEC	1.5 SEC	1.5 SEC	1.2 SEC

6. RELAP5 PLANT MODEL DESCRIPTION

The RELAP5/MOD2 code version used for this analysis is described in BAW-10164P. This version represents an upgrade to EG&G Cycle 36.02 of RELAP5/MOD2. The RELAP5 code is well documented; thus, the purpose of this section is to describe and discuss the noding arrangement and features of the B&W safety analysis plant model. Two distinct models are described:

1. RSG Plant (Full Power) Application Model (Figure 6.1 and Table 6.1).

2. RSG Plant (Low Power) Application Model (Figure 6.2 and Table 6.2).

The full power model was used for the analysis of:

1. Rod withdrawal transient
2. Four pump coastdown transient
3. Turbine trip transient
4. Locked rotor transient.

The low power model was used for the analysis of the steam line break transient.

The first step in establishing a noding arrangement was a thorough review of available literature to determine what has been used successfully. The two models presented most closely resemble the following:

1. The full power reactor vessel model is similar to the LOFT test reactor vessel model as described in "RELAP5 Assessment: LOFT Large Break L2-S," NUREG/CR-3608, SAND 83-2549, Sandia National Laboratories, January 1984.

2. The low power steam line break reactor vessel model is similar to the model described in WCAP-7909, "MARVEL A Digital Computer Code for Transient Analysis of a Multiloop PWR System," October 1972 and letter, J. R. Miller to J. M. Griffin, CESEC Code Verification, March 20, 1984.

3. The rest of the plant model (reactor coolant loops, recirculating steam generators, etc.) is consistent with the noding arrangement described in NUREG/CR-3977, EGG-2341, "RELAP5 Thermal-Hydraulic Analyses of Pressurized Thermal Shock Sequences for the H. B. Robinson Unit 2 Pressurized Water Reactor," April 1985.

The plant geometry and plant parameters are consistent with the 3411 MWt four loop plant as described in the Trojan Safety Analysis Report.

6.1. RSG Plant (Full Power) Model

Figure 6.1 shows the noding arrangement for the full power model and Table 6.1 describes each of the nodes. The following is a description of the features of this model.

Overall Plant Model

The RSG plant (full power) application model represents a four-loop plant of Westinghouse design operating at 3479 MWt (102% of

3411 MWt). As shown by Figure 6.1, a single loop and single recirculating steam generator are modeled as the 200- and 600-series nodes. The 100-series and 700-series nodes represent a combined triple loop and triple recirculating steam generator. The 400-series nodes are the pressurizer, and the 300-series nodes are the reactor vessel.

No heat structures were included except for heat transfer heat structures (RSG tubes and fuel clad). Excluding other heat structures is conservative because these tend to mitigate both transient heatup and cooldown rates. Cold plant dimensions were used for the model. Cold plant dimensions are conservative since the transient heatup and cooldown rates are greatest for the minimum water volume.

The RELAP5/MOD2 nonequilibrium option was used throughout, thus the model can calculate two-phase conditions anywhere in the system. Detailed pressure drops and flow distributions used in the model were based on calculations done using the B&W-developed SAVER (Reference 11) computer code.

<u>Fuel Pin and Nucleonics</u>

The fuel pin model consists of 5 radial nodes, 1 gap node, and 2 clad nodes. In addition, there are 6 axial nodes. The specific heat and thermal conductivity for all materials are input as parameter versus temperature tables.

RELAP5/MOD2 has a point kinetics model that has six delayed neutron groups. The built-in delayed neutron constants were used because the values are input as delayed neutron precursor yield "ratios" which do not vary significantly from beginning-of-life to end-of-life. RELAP5/MOD2 has provision for water temperature, fuel temperature, and water density reactivity feedback. Other reactivity feedbacks such as boron can be modeled by the use of control variables and general tables. Tripped rod reactivity versus time was input as a generalized reactivity versus time table. These tables were used for rod withdrawal reactivity addition and subcritical margin input.

The RELAP5/MOD2 built-in 1973 ANS Standard Fission Product Data and a fission product yield factor of 1.2 was used. Also, a 0.4 U-239 yield factor was used.

Reactor Core

The reactor core for the RSG plant (full power) model consists of an average channel (nodes 316-326), a hot channel (nodes 328-338), a core bypass flow (nodes 340, 342, and 344), and a baffle gap flow (nodes 346, 348, and 350). The baffle gap is modeled as an upflow channel in this instance but could be easily changed to a downflow configuration.

Reactor Vessel

The reactor vessel model for the RSG plant (full power) model consists of a triple downcomer (nodes 302-308) and a single downcomer (nodes 368-374). The reactor vessel lower head is modeled as node 310 and the core inlet plenum is modeled as two nodes, one for mixing and the other for core flow distribution. The core outlet plenum is modeled as a single node (352). The remaining nodes are all associated with the reactor vessel upper head.

Reactor Coolant Loops

The hot leg is separated into 4 nodes, the RSG inlet plenum (nodes 120 and 220) and RSG outlet plenum (nodes 130 and 230) are single nodes, the RSG tubes (nodes 125 and 225) are separated into eight segments, the cold leg RC pump suction consists of 5 nodes, the RC pump (nodes 160 and 260) is a single node, and the cold leg RC pump discharge consists of 4 nodes.

Reactor Coolant Pumps

For this model, the RELAP5/MOD2 built-in Westinghouse pump data was used. The model is adequate for the reactor coolant flow variations due to transient pressure and temperature changes, however, pump speed versus time and reactor coolant flow versus time were used for the four pump coastdown and locked rotor analysis of this report. If the RELAP5/MOD2 pump model is used for flow coastdown calculations in the future, plant specific or conservative homologous tables will be input.

Pressurizer

The pressurizer model consists of three parts, the surge line (node 400), the eight section pressurizer (node 410), and a valve model (junction 415). The model does not have a heater nor spray model because they tend to minimize the heatup and cooldown rates during transients.

Initial condition water volume in the pressurizer is specified by setting the void fraction in the nodes to liquid, vapor, or a mixture. The time dependent junction of the valve model (junction 415) can be used to input a mass flow versus pressure that is representative of power operated relief valves, safety valves, or combinations of valves.

Recirculating Steam Generator

The single RSG (600 series nodes) and triple RSG (700 series nodes) are the same except for the size they represent, thus the following discussion will be limited to the single RSG nodes. The feedwater mixing node (620) combines the subcooled feedwater with saturated fluid from the separator (node 650) to produce a subcooled fluid that enters the four-section downcomer (node 625). The four-section tube riser (node 630) accepts the heat transferred from the RSG tubes (node 225). Nodes 625, 620, 645, 655, and 665 are annulus nodes. The tube riser nodes accept heat from both the hot inlet and cold outlet parts of the RSG tubes (node 225). Two liquid volumes are modeled below the separator (node 650) and two steam volumes are modeled above the separator. The separator component in RELAP5/MOD2 acts as a steam separator and dryer since two-phase fluid enters from the bottom (junction 650003), steam goes out the top (junction 65001), and saturated fluid goes back to the downcomer (junction 65002). Node 655 is a separator bypass node that is recommended in RELAP5/MOD2 to provide pressure equalization across the separator. The separator bypass model has been shown to produce more realistic transient results. Nodes 645 and 655 are used to support the separator bypass modeling. Out of the steam dome (node 670) the 1.4 ft^2 flow restrictor (junction 671) is modeled as an integral part of the steam generator.

The model represents a Westinghouse Type 51 steam generator model (51,500 ft^2 tube heat transfer area) with a 6% reduction for tube plugging. The recirculation ratio of this model is 3.8.

No recirculating steam generator water level model was developed because none of the transients considered for this report requires water level for control or for reactor trip. A RSG water level could easily be implemented using the RELAP5/MOD2 control variables.

Feedwater System

The feedwater system was modeled as a time-dependent volume (node 600) and a time-dependent junction (junction 601). The time-dependent volume can also be used to adjust feedwater temperature versus time, and the time-dependent junction can be used to adjust mass flowrate versus time. This type of modeling is adequate for the transients of this report where the feedwater flow is terminated early in the transient and can also be used for quite complex feedwater flow situations.

Auxiliary feedwater was not modeled as a separate system since it usually is not used in the short duration full power transients. Auxiliary feedwater can be represented as continued flow through the feedwater time dependent junction.

Steam Line

The steam lines (node 675) run from the steam generator to the turbine which is modeled as a time dependent volume (node 690) to maintain the proper backpressure. Junction 677 is a motor operated valve that can be used to represent either a main steam stop valve or a main steam isolation valve. Time dependent junction 676 and time dependent volume 680 provide modeling for steam line safety valves, power operated relief valves, and combinations of valves. Time dependent junction 678 and time dependent volume 550 provide modeling for a steam dump system that consists of a three second valve opening to a preset mass flowrate.

6.2. RSG Plant Steam Line Break (Low Power) Model

Figure 6.2 shows the noding arrangement for the low power model and Table 6.2 describes each of the nodes. The following is a description of the changes from the full power model. The only significant changes from the full power model are the reactor core and the addition of safety injection.

Overall Plant Model

The RSG plant steam line break (low power) model is a four loop plant operating at 17 MWt (0.5% of 3411 MWt). The use of this very low power level allows the RELAP5/MOD2 model to be run to steady state prior to the start of the transient calculations.

The low power steady-state was found by allowing the full power RELAP5/MOD2 model to achieve steady-state with a new set of temperatures, lower core power level and lower feedwater flow conditions. The RSG model represents a Westinghouse Type 51 steam generator model with a 51,500 ft^2 tube heat transfer area.

Reactor Core

The reactor core for the RSG plant steam line break (low power) model consists of an average channel triple loop core (nodes 316-326), average channel single loop core (nodes 317-327), triple loop core bypass (nodes 340, 342, 344) and single loop core bypass (nodes 341, 343, 345). The splitting of the core into a triple and single loop segment allows the steam line break affected loop to be isolated from the triple loop. Crossflow paths are provided at the core inlet and outlet to represent various flow mixing situations. Both the inlet and outlet plenum are broken into triple loop and single loop segments.

Reactivity feedback due to the temperature decrease of the steam line break is based on the single loop core density.

Safety Injection System

Time dependent volume 930 and time dependent junction 931 were used to model safety injection of boron into the single loop reactor coolant system during the steam line break. A mass flowrate versus pressure table was used to represent the safety injection system that pumps 2000 ppm boron into the reactor coolant system that has no boron initially. The boron injection was activated on low pressurizer pressure and a 12 second time delay was used to represent valve openings and pumps coming up to full speed.

Only the safety injection was modeled because that is all that is used in the steam line break analysis since the reactor coolant pressure does not go low enough for the accumulator tanks to operate. RELAP5/MOD2 has an accumulator tank model and the residual heat removal system could easily be modeled.

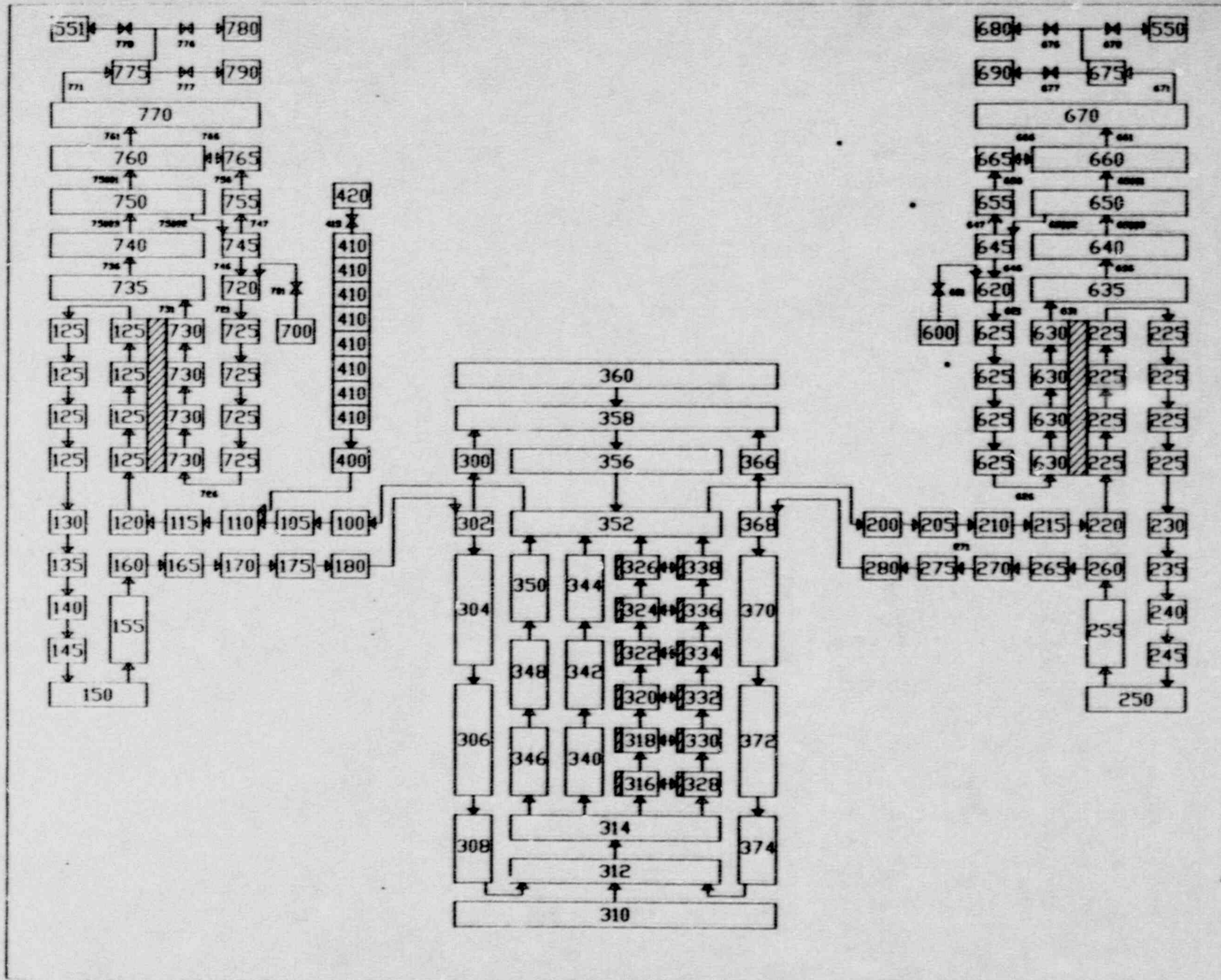


Figure 6.1. Babcock & Wilcox RSG Plant (Full Power) Model

TABLE 6.1. RELAP5 RSG PLANT (FULL POWER) MODEL
NODE & JUNCTION DESCRIPTION

REACTOR VESSEL AND CORE COMPONENTS

#	TYPE	ORIENTATION	DESCRIPTION
300	BRANCH	+90	TRIPLE UPPER INLET PLENUM
302	BRANCH	-90	TRIPLE LOWER INLET PLENUM
304	BRANCH	-90	TRIPLE LOWER DOWNCOMER 1
306	BRANCH	-90	TRIPLE LOWER DOWNCOMER 2
308	BRANCH	-90	TRIPLE LOWER DOWNCOMER 3
310	BRANCH	+90	LOWER HEAD
312	BRANCH	+90	LOWER PLENUM
314	BRANCH	+90	CORE INLET PLENUM
316	BRANCH	+90	AVERAGE CORE SEGMENT 1 OF 6
318	BRANCH	+90	AVERAGE CORE SEGMENT 2 OF 6
320	BRANCH	+90	AVERAGE CORE SEGMENT 3 OF 6
322	BRANCH	+90	AVERAGE CORE SEGMENT 4 OF 6
324	BRANCH	+90	AVERAGE CORE SEGMENT 5 OF 6
326	BRANCH	+90	AVERAGE CORE SEGMENT 6 OF 6
328	BRANCH	+90	HOT CHANNEL SEGMENT 1 OF 6
330	BRANCH	+90	HOT CHANNEL SEGMENT 2 OF 6
332	BRANCH	+90	HOT CHANNEL SEGMENT 3 OF 6
334	BRANCH	+90	HOT CHANNEL SEGMENT 4 OF 6
336	BRANCH	+90	HOT CHANNEL SEGMENT 5 OF 6
338	BRANCH	+90	HOT CHANNEL SEGMENT 6 OF 6
340	BRANCH	+90	CORE BYPASS SEGMENT 1 OF 3
342	BRANCH	+90	CORE BYPASS SEGMENT 2 OF 3
344	BRANCH	+90	CORE BYPASS SEGMENT 3 OF 3

TABLE 6.1. RELAP5 RSG PLANT (FULL POWER) MODEL NODE & JUNCTION DESCRIPTION (CONT'D.)

REACTOR VESSEL AND CORE COMPONENTS

#	TYPE	ORIENTATION	DESCRIPTION
346	BRANCH	+90	BAFFLE REGION SEGMENT 1 OF 3
348	BRANCH	+90	BAFFLE REGION SEGMENT 2 OF 3
350	BRANCH	+90	BAFFLE REGION SEGMENT 3 OF 3
352	BRANCH	+90	CORE OUTLET PLENUM
356	BRANCH	-90	UPPER PLENUM SEGMENT 1 OF 2
358	BRANCH	-90	UPPER PLENUM SEGMENT 2 OF 2
360	BRANCH	-90	UPPER HEAD
366	BRANCH	+90	SINGLE UPPER INLET PLENUM
368	BRANCH	-90	SINGLE LOWER INLET PLENUM
370	BRANCH	-90	SINGLE LOWER DOWNCOMER 1
372	BRANCH	-90	SINGLE LOWER DOWNCOMER 2
374	BRANCH	-90	SINGLE LOWER DOWNCOMER 3

SINGLE RECIRCULATING STEAM GENERATOR

#	TYPE	ORIENTATION	DESCRIPTION
600	TMDPVOL		SINGLE RSG FEEDWATER TANK
601	TMDPJUN		SINGLE RSG FEEDWATER VALVE
620	ANNULUS	-90	SINGLE RSG FEEDWATER INLET
621	SNGJUN		SINGLE RSG FDWT INLET TO LOWER DWNCMR
625	ANNULUS	-90	SINGLE RSG LOWER DOWNCOMER (4 SEGMENTS)
626	SNGJUN		SINGLE RSG DWNCMR TO TUBE SECONDARY
630	PIPE	+90	SINGLE RSG TUBE SECONDARY (4 SEGMENTS)

TABLE 6.1. RELAP5 RSG PLANT (FULL POWER) MODEL NODE & JUNCTION DESCRIPTION (CONT'D.)

SINGLE RECIRCULATING STEAM GENERATOR

#	TYPE	ORIENTATION	DESCRIPTION
631	SNGJUN		SINGLE RSG TUBE SECONDARY TO LOWER LIQUID
635	BRANCH	+90	SINGLE RSG LOWER LIQUID
636	SNGJUN		SINGLE RSG LOWER TO UPPER LIQUID
640	BRANCH	+90	SINGLE RSG UPPER LIQUID
645	ANNULUS	-90	SINGLE RSG UPPER DOWNCOMER
646	SNGJUN		SINGLE RSG UPPER DWNCMR TO FDWT INLET
647	SNGJUN		SINGLE RSG UPPER DWNCMR TO LOWER SEP BYPASS
650	SEPATR	+90	SINGLE RSG SEPARATOR
655	ANNULUS	+90	SINGLE RSG LOWER SEPARATOR BYPASS
656	SNGJUN		SINGLE RSG LOWER TO UPPER SEPARATOR BYPASS
660	BRANCH	+90	SINGLE RSG LOWER STEAM DOME
661	SNGJUN		SINGLE RSG LOWER TO UPPER STEAM DOME
665	ANNULUS	+90	SINGLE RSG UPPER SEPARATOR BYPASS
666	SNGJUN		SINGLE RSG UPPER SEP BYPASS TO LOWER ST DOME
670	BRANCH	+90	SINGLE RSG UPPER STEAM DOME
671	SNGJUN		SINGLE RSG STEAM LINE FLOW RESTRICTOR
675	SNGLVOL		SINGLE RSG STEAM LINE
676	TMDPJUN		SINGLE RSG STEAM SAFETY VALVE

TABLE 6.1. RELAP5 RSG PLANT (FULL POWER) MODEL NODE & JUNCTION DESCRIPTION (CONT'D.)

SINGLE RECIRCULATING STEAM GENERATOR

#	TYPE	ORIENTATION	DESCRIPTION
677	MTRVLV		SINGLE RSG STEAM VALVE
680	TMDPVOL		SINGLE RSG STEAM SAFETY VALVE TANK
690	TMDPVOL		SINGLE RSG STEAM LINE TANK

TRIPLE RECIRCULATING STEAM GENERATOR

#	TYPE	ORIENTATION	DESCRIPTION
700	TMDPVOL		TRIPLE RSG FEEDWATER TANK
701	TMDPJUN		TRIPLE RSG FEEDWATER VALVE
720	ANNULUS	-90	TRIPLE RSG FEEDWATER INLET
721	SNGJUN		TRIPLE RSG FDWT INLET TO LOWER DWNCMR
725	ANNULUS	-90	TRIPLE RSG LOWER DOWNCOMER (4 SEGMENTS)
726	SNGJUN		TRIPLE RSG DWNCMR TO TUBE SECONDARY
730	PIPE	+90	TRIPLE RSG TUBE SECONDARY (4 SEGMENTS)
731	SNGJUN		TRIPLE RSG TUBE SECONDARY TO LOWER LIQUID
735	BRANCH	+90	TRIPLE RSG LOWER LIQUID
736	SNGJUN		TRIPLE RSG LOWER TO UPPER LIQUID
740	BRANCH	+90	TRIPLE RSG UPPER LIQUID
745	ANNULUS	-90	TRIPLE RSG UPPER DOWNCOMER
746	SNGJUN		TRIPLE RSG UPPER DWNCMR TO FDWT INLET

TABLE 6.1. RELAP5 RSG PLANT (FULL POWER) MODEL NODE JUNCTION DESCRIPTION (CONT'D.)

TRIPLE RECIRCULATING STEAM GENERATOR

#	TYPE	ORIENTATION	DESCRIPTION
747	SNGJUN		TRIPLE RSG UPPER DWNCMR TO LOWER SEP BYPASS
750	SEPATR	+90	TRIPLE RSG SEPARATOR
755	ANNULUS	+90	TRIPLE RSG LOWER SEPARATOR BYPASS
756	SNGJUN		TRIPLE RSG LOWER TO UPPER SEPARATOR BYPASS
760	BRANCH	+90	TRIPLE RSG LOWER STEAM DOME
761	SNGJUN		TRIPLE RSG LOWER TO UPPER STEAM DOME
765	ANNULUS	+90	TRIPLE RSG UPPER SEPARATOR BYPASS
766	SNGJUN		TRIPLE RSG UPPER SEP BYPASS TO LOWER ST DOME
770	BRANCH	+90	TRIPLE RSG UPPER STEAM DOME
771	SNGJUN		TRIPLE RSG STEAM LINE FLOW RESTRICTOR
775	SNGLVOL		TRIPLE RSG STEAM LINE
776	TMDPJUN		TRIPLE RSG STEAM SAFETY VALVE
777	MTRVLV		TRIPLE RSG STEAM VALVE
780	TMDPVOL		TRIPLE RSG STEAM SAFETY VALVE TANK
790	TMDPVOL		TRIPLE RSG STEAM LINE TANK

TABLE 6.1. RELAP5 RSG PLANT (FULL POWER) MODEL
NODE & JUNCTION DESCRIPTION (CONT'D.)

SINGLE LOOP HOT AND COLD LEGS

#	TYPE	ORIENTATION	DESCRIPTION
200	BRANCH	HORIZ	SINGLE LOOP HOT LEG SEGMENT 1 OF 4
205	BRANCH	HORIZ	SINGLE LOOP HOT LEG SEGMENT 2 OF 4
210	BRANCH	HORIZ	SINGLE LOOP HOT LEG SEGMENT 3 OF 4
215	BRANCH	ANGLE	SINGLE LOOP HOT LEG SEGMENT 4 OF 4
220	BRANCH	+90	SINGLE RSG INLET PLENUM
225	PIPE	+90/-90	SINGLE RSG PRIMARY TUBES (8 SEGMENTS)
230	BRANCH	-90	SINGLE RSG OUTLET PLENUM
235	BRANCH	ANGLE	SINGLE RSG COLD LEG SEGMENT 1 OF 5
240	BRANCH	-90	SINGLE RSG COLD LEG SEGMENT 2 OF 5
245	BRANCH	ANGLE	SINGLE RSG COLD LEG SEGMENT 3 OF 5
250	BRANCH	HORIZ	SINGLE RSG COLD LEG SEGMENT 4 OF 5
255	BRANCH	ANGLE	SINGLE RSG COLD LEG SEGMENT 5 OF 5
260	PUMP	ANGLE	SINGLE LOOP RC PUMP
265	BRANCH	HORIZ	SINGLE PUMP COLD LEG SEGMENT 1 OF 4
270	BRANCH	HORIZ	SINGLE PUMP COLD LEG SEGMENT 2 OF 4
271	MTRVLV		SINGLE LOOP LOCA VALVE
275	BRANCH	HORIZ	SINGLE PUMP COLD LEG SEGMENT 3 OF 4

TABLE 6.1. RELAP5 RSG PLANT (FULL POWER) MODEL
NODE & JUNCTION DESCRIPTION (CONT'D.)

SINGLE LOOP HOT AND COLD LEGS

#	TYPE	ORIENTATION	DESCRIPTION
280	BRANCH	HORIZ	SINGLE PUMP COLD LEG SEGMENT 4 OF 4

TRIPLE LOOP HOT AND COLD LEGS

#	TYPE	ORIENTATION	DESCRIPTION
100	BRANCH	HORIZ	TRIPLE LOOP HOT LEG SEGMENT 1 OF 4
105	BRANCH	HORIZ	TRIPLE LOOP HOT LEG SEGMENT 2 OF 4
110	BRANCH	HORIZ	TRIPLE LOOP HOT LEG SEGMENT 3 OF 4
115	BRANCH	ANGLE	TRIPLE LOOP HOT LEG SEGMENT 4 OF 4
120	BRANCH	+90	TRIPLE RSG INLET PLENUM
125	PIPE	+90/-90	TRIPLE RSG PRIMARY TUBES (8 SEGMENTS)
130	BRANCH	-90	TRIPLE RSG OUTLET PLENUM
135	BRANCH	ANGLE	TRIPLE RSG COLD LEG SEGMENT 1 OF 5
140	BRANCH	-90	TRIPLE RSG COLD LEG SEGMENT 2 OF 5
145	BRANCH	ANGLE	TRIPLE RSG COLD LEG SEGMENT 3 OF 5
150	BRANCH	HORIZ	TRIPLE RSG COLD LEG SEGMENT 4 OF 5
155	BRANCH	ANGLE	TRIPLE RSG COLD LEG SEGMENT 5 OF 5
160	PUMP	ANGLE	TRIPLE LOOP RC PUMP
165	BRANCH	HORIZ	TRIPLE PUMP COLD LEG SEGMENT 1 OF 4

TABLE 6.1. RELAP5 RSG PLANT (FULL POWER) MODEL NODE & JUNCTION DESCRIPTION (CONT'D.)

TRIPLE LOOP HOT AND COLD LEGS

#	TYPE	ORIENTATION	DESCRIPTION
170	BRANCH	HORIZ	TRIPLE PUMP COLD LEG SEGMENT 2 OF 4
175	BRANCH	HORIZ	TRIPLE PUMP COLD LEG SEGMENT 3 OF 4
180	BRANCH	HORIZ	TRIPLE PUMP COLD LEG SEGMENT 4 OF 4

PRESSURIZER

#	TYPE	ORIENTATION	DESCRIPTION
400	BRANCH	-90	PRESSURIZER SURGE LINE
410	PIPE	-90	PRESSURIZER (8 SEGMENTS)
415	TMDPJUN		PRESSURIZER SAFETY VALVE
420	TMDPVOL		PRESSURIZER STEAM SINK

STEAM DUMP SYSTEM

#	TYPE	ORIENTATION	DESCRIPTION
678	TMDPJUN		SINGLE RSG STEAM DUMP CONTROL VALVE
550	TMDPVOL		SINGLE RSG STEAM DUMP TANK
778	TMDPJUN		TRIPLE RSG STEAM DUMP CONTROL VALVE
551	TMDPVOL		TRIPLE RSG STEAM DUMP TANK

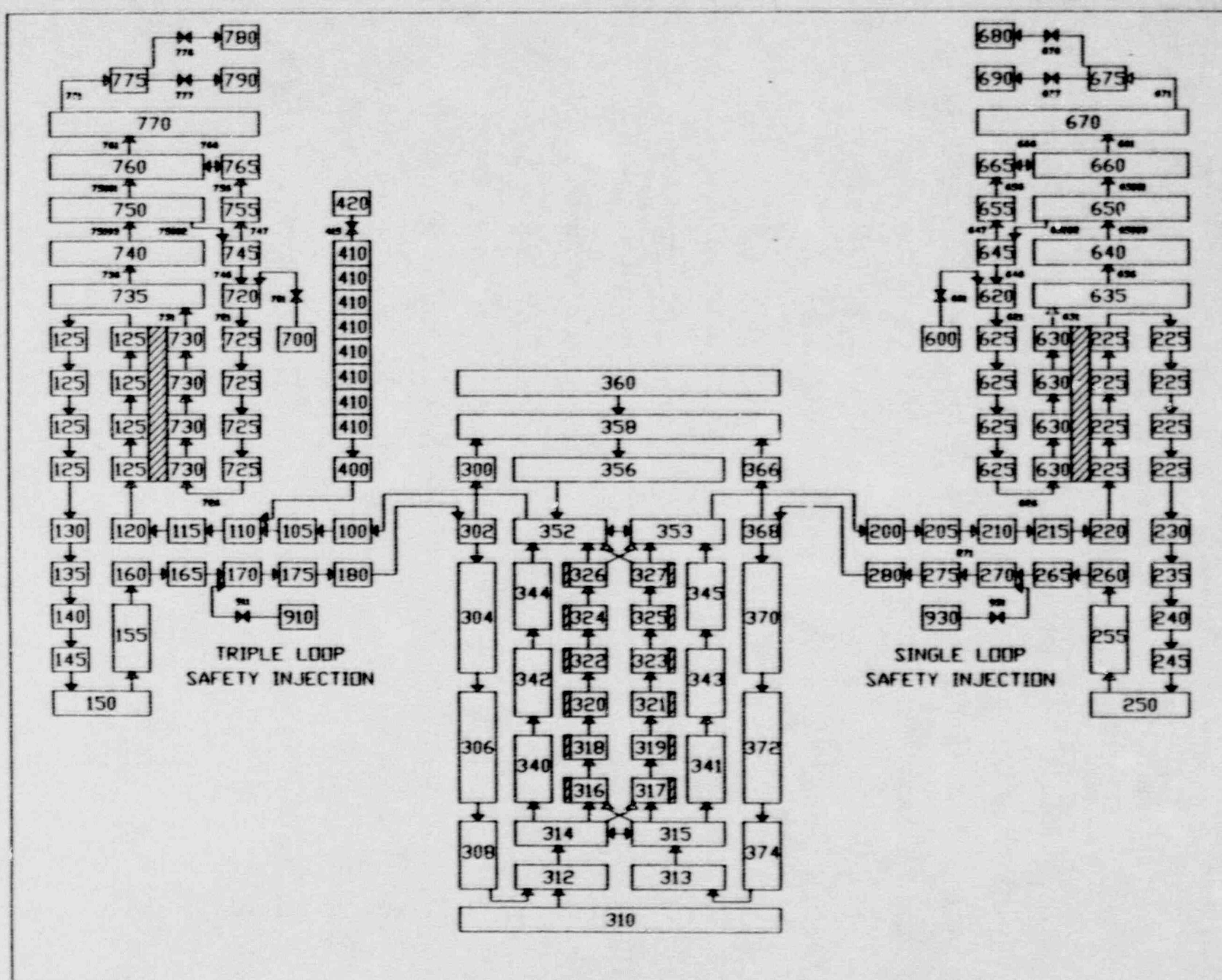


Figure 6.2. Babcock & Wilcox RSG Plant Steam Line Break (Low Power) Model

TABLE 6.2. RELAP5 RSG PLANT STEAM LINE BREAK (LOW POWER) MODEL NODE & JUNCTION DESCRIPTION

REACTOR VESSEL AND CORE COMPONENTS

#	TYPE	ORIENTATION	DESCRIPTION
300	BRANCH	+90	TRIPLE UPPER INLET PLENUM
302	BRANCH	-90	TRIPLE LOWER INLET PLENUM
304	BRANCH	-90	TRIPLE LOWER DOWNCOMER 1
306	BRANCH	-90	TRIPLE LOWER DOWNCOMER 2
308	BRANCH	-90	TRIPLE LOWER DOWNCOMER 3
310	BRANCH	+90	LOWER HEAD
312	BRANCH	+90	TRIPLE LOWER PLENUM
313	BRANCH	+90	SINGLE LOWER PLENUM
314	BRANCH	+90	TRIPLE CORE INLET PLENUM
315	BRANCH	+90	SINGLE CORE INLET PLENUM
316	BRANCH	+90	TRIPLE CORE SEGMENT 1 OF 6
318	BRANCH	+90	TRIPLE CORE SEGMENT 2 OF 6
320	BRANCH	+90	TRIPLE CORE SEGMENT 3 OF 6
322	BRANCH	+90	TRIPLE CORE SEGMENT 4 OF 6
324	BRANCH	+90	TRIPLE CORE SEGMENT 5 OF 6
326	BRANCH	+90	TRIPLE CORE SEGMENT 6 OF 6
317	BRANCH	+90	SINGLE CORE SEGMENT 1 OF 6
319	BRANCH	+90	SINGLE CORE SEGMENT 2 OF 6
321	BRANCH	+90	SINGLE CORE SEGMENT 3 OF 6
323	BRANCH	+90	SINGLE CORE SEGMENT 4 OF 6
325	BRANCH	+90	SINGLE CORE SEGMENT 5 OF 6
327	BRANCH	+90	SINGLE CORE SEGMENT 6 OF 6
340	BRANCH	+90	TRIPLE CORE BYPASS SEGMENT 1 OF 3

TABLE 6.2. RELAP5 RSG PLANT STEAM LINE BREAK (LOW POWER) MODEL NODE & JUNCTION DESCRIPTION (CONT'D)

REACTOR VESSEL AND CORE COMPONENTS

#	TYPE	ORIENTATION	DESCRIPTION
342	BRANCH	+90	TRIPLE CORE BYPASS SEGMENT 2 OF 3
344	BRANCH	+90	TRIPLE CORE BYPASS SEGMENT 3 OF 3
341	BRANCH	+90	SINGLE CORE BYPASS SEGMENT 1 OF 3
343	BRANCH	+90	SINGLE CORE BYPASS SEGMENT 2 OF 3
345	BRANCH	+90	SINGLE CORE BYPASS SEGMENT 3 OF 3
352	BRANCH	+90	TRIPLE CORE OUTLET PLENUM
353	BRANCH	+90	SINGLE CORE OUTLET PLENUM
356	BRANCH	-90	UPPER PLENUM SEGMENT 1 OF 2
358	BRANCH	-90	UPPER PLENUM SEGMENT 2 OF 2
360	BRANCH	-90	UPPER HEAD
366	BRANCH	+90	SINGLE UPPER INLET PLENUM
368	BRANCH	-90	SINGLE LOWER INLET PLENUM
370	BRANCH	-90	SINGLE LOWER DOWNCOMER 1
372	BRANCH	-90	SINGLE LOWER DOWNCOMER 2
374	BRANCH	-90	SINGLE LOWER DOWNCOMER 3

SINGLE RECIRCULATING STEAM GENERATOR

#	TYPE	ORIENTATION	DESCRIPTION
600	TMDPVOL		SINGLE RSG FEEDWATER TANK
601	TMDPJUN		SINGLE RSG FEEDWATER VALVE
620	ANNULUS	-90	SINGLE RSG FEEDWATER INLET

TABLE 6.2. RELAP5 RSG PLANT STEAM LINE BREAK (LOW POWER) MODEL NODE & JUNCTION DESCRIPTION (CONT'D)

SINGLE RECIRCULATING STEAM GENERATOR

#	TYPE	ORIENTATION	DESCRIPTION
621	SNGJUN		SINGLE RSG FDWT INLET TO LOWER DWNCMR
625	ANNULUS	-90	SINGLE RSG LOWER DOWNCOMER (4 SEGMENTS)
626	SNGJUN		SINGLE RSG DWNCMR TO TUBE SECONDARY
630	PIPE	+90	SINGLE RSG TUBE SECONDARY (4 SEGMENTS)
631	SNGJUN		SINGLE RSG TUBE SECONDARY TO LOWER LIQUID
635	BRANCH	+90	SINGLE RSG LOWER LIQUID
636	SNGJUN		SINGLE RSG LOWER TO UPPER LIQUID
640	BRANCH	+90	SINGLE RSG UPPER LIQUID
645	ANNULUS	-90	SINGLE RSG UPPER DOWNCOMER
646	SNGJUN		SINGLE RSG UPPER DWNCMR TO FDWT INLET
647	SNGJUN		SINGLE RSG UPPER DWNCMR TO LOWER SEP BYPASS
650	SEPATR	+90	SINGLE RSG SEPARATOR
655	ANNULUS	+90	SINGLE RSG LOWER SEPARATOR BYPASS
656	SNGJUN		SINGLE RSG LOWER TO UPPER SEPARATOR BYPASS
660	BRANCH	+90	SINGLE RSG LOWER STEAM DOME
661	SNGJUN		SINGLE RSG LOWER TO UPPER STEAM DOME
665	ANNULUS	+90	SINGLE RSG UPPER SEPARATOR BYPASS

TABLE 6.2. RELAP5 RSG PLANT STEAM LINE BREAK (LOW POWER) MODEL NODE & JUNCTION DESCRIPTION (CONT'D)

SINGLE RECIRCULATING STEAM GENERATOR

#	TYPE	ORIENTATION	DESCRIPTION
666	SNGJUN		SINGLE RSG UPPER SEP BYPASS TO LOWER ST DOME
670	BRANCH	+90	SINGLE RSG UPPER STEAM DOME
671	SNGJUN		SINGLE RSG STEAM LINE FLOW RESTRICTOR
675	SNGLVOL		SINGLE RSG STEAM LINE
676	TMDPJUN		SINGLE RSG STEAM SAFETY VALVE
677	MTRVLV		SINGLE RSG STEAM VALVE
680	TMDPVOL		SINGLE RSG STEAM SAFETY VALVE TANK
690	TMDPVOL		SINGLE RSG STEAM LINE TANK

TRIPLE RECIRCULATING STEAM GENERATOR

#	TYPE	ORIENTATION	DESCRIPTION
700	TMDPVOL		TRIPLE RSG FEEDWATER TANK
701	TMDPJUN		TRIPLE RSG FEEDWATER VALVE
720	ANNULUS	-90	TRIPLE RSG FEEDWATER INLET
721	SNGJUN		TRIPLE RSG FDWT INLET TO LOWER DWNCMR
725	ANNULUS	-90	TRIPLE RSG LOWER DOWNCOMER (4 SEGMENTS)
726	SNGJUN		TRIPLE RSG DWNCMR TO TUBE SECONDARY
730	PIPE	+90	TRIPLE RSG TUBE SECONDARY (4 SEGMENTS)
731	SNGJUN		TRIPLE RSG TUBE SECONDARY TO LOWER LIQUID

TABLE 6.2. RELAP5 RSG PLANT STEAM LINE BREAK (LOW POWER) MODEL NODE & JUNCTION DESCRIPTION (CONT'D)

TRIPLE RECIRCULATING STEAM GENERATOR

#	TYPE	ORIENTATION	DESCRIPTION
735	BRANCH	+90	TRIPLE RSG LOWER LIQUID
736	SNGJUN		TRIPLE RSG LOWER TO UPPER LIQUID
740	BRANCH	+90	TRIPLE RSG UPPER LIQUID
745	ANNULUS	-90	TRIPLE RSG UPPER DOWNCOMER
746	SNGJUN		TRIPLE RSG UPPER DWNCMR TO FDWT INLET
747	SNGJUN		TRIPLE RSG UPPER DWNCMR TO LOWER SEP BYPASS
750	SEPATR	+90	TRIPLE RSG SEPARATOR
755	ANNULUS	+90	TRIPLE RSG LOWER SEPARATOR BYPASS
756	SNGJUN		TRIPLE RSG LOWER TO UPPER SEPARATOR BYPASS
760	BRANCH	+90	TRIPLE RSG LOWER STEAM DOME
761	SNGJUN		TRIPLE RSG LOWER TO UPPER STEAM DOME
765	ANNULUS	+90	TRIPLE RSG UPPER SEPARATOR BYPASS
766	SNGJUN		TRIPLE RSG UPPER SEP BYPASS TO LOWER ST DOME
770	BRANCH	+90	TRIPLE RSG UPPER STEAM DOME
771	SNGJUN		TRIPLE RSG STEAM LINE FLOW RESTRICTOR
775	SNGLVOL		TRIPLE RSG STEAM LINE
776	TMDPJUN		TRIPLE RSG STEAM SAFETY VALVE
777	MTRVLV		TRIPLE RSG STEAM VALVE

TABLE 6.2. RELAP5 RSG PLANT STEAM LINE BREAK (LOW POWER) MODEL NODE & JUNCTION DESCRIPTION (CONT'D)

TRIPLE RECIRCULATING STEAM GENERATOR

#	TYPE	ORIENTATION	DESCRIPTION
780	TMDPVOL		TRIPLE RSG STEAM SAFETY VALVE TANK
790	TMDPVOL		TRIPLE RSG STEAM LINE TANK

SINGLE LOOP HOT AND COLD LEGS

#	TYPE	ORIENTATION	DESCRIPTION
200	BRANCH	HORIZ	SINGLE LOOP HOT LEG SEGMENT 1 OF 4
205	BRANCH	HORIZ	SINGLE LOOP HOT LEG SEGMENT 2 OF 4
210	BRANCH	HORIZ	SINGLE LOOP HOT LEG SEGMENT 3 OF 4
215	BRANCH	ANGLE	SINGLE LOOP HOT LEG SEGMENT 4 OF 4
220	BRANCH	+90	SINGLE RSG INLET PLENUM
225	PIPE	+90/-90	SINGLE RSG PRIMARY TUBES (8 SEGMENTS)
230	BRANCH	-90	SINGLE RSG OUTLET PLENUM
235	BRANCH	ANGLE	SINGLE RSG COLD LEG SEGMENT 1 OF 5
240	BRANCH	-90	SINGLE RSG COLD LEG SEGMENT 2 OF 5
245	BRANCH	ANGLE	SINGLE RSG COLD LEG SEGMENT 3 OF 5
250	BRANCH	HORIZ	SINGLE RSG COLD LEG SEGMENT 4 OF 5
255	BRANCH	ANGLE	SINGLE RSG COLD LEG SEGMENT 5 OF 5
260	PUMP	ANGLE	SINGLE LOOP RC PUMP

TABLE 6.2. RELAP5 RSG PLANT STEAM LINE BREAK (LOW POWER) MODEL NODE & JUNCTION DESCRIPTION (CONT'D)

SINGLE LOOP HOT AND COLD LEGS

#	TYPE	ORIENTATION	DESCRIPTION
265	BRANCH	HORIZ	SINGLE PUMP COLD LEG SEGMENT 1 OF 4
270	BRANCH	HORIZ	SINGLE PUMP COLD LEG SEGMENT 2 OF 4
271	MTRVLV		SINGLE LOOP LOCA VALVE
275	BRANCH	HORIZ	SINGLE PUMP COLD LEG SEGMENT 3 OF 4
280	BRANCH	HORIZ	SINGLE PUMP COLD LEG SEGMENT 4 OF 4

TRIPLE LOOP HOT AND COLD LEGS

#	TYPE	ORIENTATION	DESCRIPTION
100	BRANCH	HORIZ	TRIPLE LOOP HOT LEG SEGMENT 1 OF 4
105	BRANCH	HORIZ	TRIPLE LOOP HOT LEG SEGMENT 2 OF 4
110	BRANCH	HORIZ	TRIPLE LOOP HOT LEG SEGMENT 3 OF 4
115	BRANCH	ANGLE	TRIPLE LOOP HOT LEG SEGMENT 4 OF 4
120	BRANCH	+90	TRIPLE RSG INLET PLENUM
125	PIPE	+90/-90	TRIPLE RSG PRIMARY TUBES (8 SEGMENTS)
130	BRANCH	-90	TRIPLE RSG OUTLET PLENUM
135	BRANCH	ANGLE	TRIPLE RSG COLD LEG SEGMENT 1 OF 5
140	BRANCH	-90	TRIPLE RSG COLD LEG SEGMENT 2 OF 5

TABLE 6.2. RELAP5 RSG PLANT STEAM LINE BREAK (LOW POWER) MODEL NODE & JUNCTION DESCRIPTION (CONT'D)

TRIPLE LOOP HOT AND COLD LEGS

#	TYPE	ORIENTATION	DESCRIPTION
145	BRANCH	ANGLE	TRIPLE RSG COLD LEG SEGMENT 3 OF 5
150	BRANCH	HORIZ	TRIPLE.RSG COLD LEG SEGMENT 4 OF 5
155	BRANCH	ANGLE	TRIPLE RSG COLD LEG SEGMENT 5 OF 5
160	PUMP		ANGLETRIPLE LOOP RC PUMP
165	BRANCH	HORIZ	TRIPLE PUMP COLD LEG SEGMENT 1 OF 4
170	BRANCH	HORIZ	TRIPLE PUMP COLD LEG SECMENT 2 OF 4
175	BRANCH	HORIZ	TRIPLE PUMP COLD LEG SEGMENT 3 OF 4
180	BRANCH	HORIZ	TRIPLE PUMP COLD LEG SEGMENT 4 OF 4

PRESSURIZER

#	TYPE	ORIENTATION	DESCRIPTION
400	BRANCH	-90	PRESSURIZER SURGE LINE
410	PIPE	-90	PRESSURIZER (8 SEGMENTS)
415	TMDPJUN		PRESSURIZER SAFETY VALVE
420	TMDPVOL		PRESSURIZER STEAM SINK

FILL SYSTEMS

#	TYPE	ORIENTATION	DESCRIPTION
910	TMDPVOL		TRIPLE LOOP SAFETY INJECTION TANK (ECCS HIGH PRESSURE INJ)
911	TMDPJUN		TRIPLE LOOP SI FLOW CONTROL

TABLE 6.2. RELAP5 RSG PLANT STEAM LINE BREAK (LOW POWER) MODEL NODE & JUNCTION DESCRIPTION (CONT'D)

FILL SYSTEMS

#	TYPE	ORIENTATION	DESCRIPTION
930	TMDPVOL		SINGLE LOOP SAFETY INJECTION TANK (ECCS HIGH PRESSURE INJ)
931	TMDPJUN		SINGLE LOOP SI FLOW CONTROL

7. COMPARATIVE ANALYSES

The purpose of the comparative analyses presented in this section is to show that the method to be used by B&W to apply RELAP5 in safety analyses of recirculating steam generator (RSG) plants is valid. The approach is straightforward: A representative plant model, in this case based upon the Westinghouse four-loop design, is used to perform RELAP5 analyses of a selection of limiting safety analysis transients as presented in plant Safety Analysis Reports. The RELAP5 analyses incorporate the same assumptions and boundary conditions as those performed for the SAR's, and the results are directly comparable to the SAR results. The validity of the B&W model and application is demonstrated when the results obtained using essentially identical inputs and assumptions are in agreement with those produced by methods that have been reviewed and accepted for the same applications.

The Safety Analysis Report data used in these comparisons were taken from four sources: the Catawba, McGuire, and Trojan FSAR's (References 1-4) and the Westinghouse RESAR (Reference 5). The purpose in comparing the B&W model to a variety of results is twofold: First, the SAR results will differ among themselves because of significant differences in methods, assumptions used in the analyses, or boundary conditions. When these significant differences in inputs or assumptions are included in the RELAP5 analyses, the effects upon the results can be compared to the SAR results. Second, the comparison of results from several RSG plants shows the transient results are not sensitive to minor changes in plant parameters. The B&W modeling approach using

RELAP5 can thus be shown to be broadly applicable for licensing cases.

Plant parameters for the four reference SAR's and for the B&W plant model are presented for comparison in Table 7.0.1. Tables 7.0.2.A through 7.0.2.D list the significant transient analysis assumptions used in each of the analyses. Examples of differences among the analyses and plants are the tripped rod insertion times assumed for the SAR cases and the steam generator designs. The Trojan and RESAR analyses were done with 2.2 seconds to 85 percent insertion tripped rod worth whereas the Catawba cases used 3.05 seconds and the McGuire analyses either 3.3 seconds or 2.7 seconds to 85 percent insertion. As will be shown, these differences have a noticeable effect upon the results.

Comparative analyses were performed for five transients:

1. 75 pcm/sec Rod Withdrawal (Full Power)
2. Four Pump Coastdown (Full Power)
3. Turbine Trip (Full Power)
4. Locked Rotor (Full Power)
5. Steam Line Break (Low Power).

The bases for the selection of these transients is presented in Section 4. In order to provide detailed comparison of the B&W model results to the SAR results for the four plants, the SAR results were digitized (where possible) and the SAR results were normalized to the B&W model results. It should be noted that B&W calculations presented for or designated by particular plant names are specific to those plants <u>only</u> in terms of selected assumptions and parameters used in the analyses. The plant model used is a single, generic model not specific to any one of these plants.

7.0.1. Plant Characteristics and Initial Conditions Assumed in the Accident Analysis

For all accidents nominal values of initial conditions were used. Table 7.0.1 lists the values of the pertinent parameters for the B&W application model and the four plants that were used for comparison. The initial power level for the B&W model was 3411 MWt plus 2%, for a total of 3479 MWt. When the DNBR was calculated the following conservative steady state errors were assumed in the DNBR analysis;

1. Core power	+ 2 percent allowance for calorimetric error
2. Average reactor coolant system temperature	+4 F allowance for controller deadband and measurement error
3. Pressurizer pressure	-30 pounds per square inch (psi) allowance for steady state fluctuations and measurement error

7.0.2. Power Distribution

The transient response of the reactor system is dependent on the initial power distribution. The nuclear design of the reactor core minimizes adverse power distributions through operating instructions and the placement of control rods. Power distribution may be characterized by the radial factor ($F_{\Delta H}$) and the total peaking factor (F_Q). The peaking factor limits are given in the Technical Specifications.

For transients which may be DNB limited, the radial peaking factor is of importance. The radial peaking factor increases with decreasing power level due to rod insertion. This increase in $F_{\Delta H}$ is included in the core limits. All transients that may be DNB limited are assumed to begin with a $F_{\Delta H}$ consistent with the initial power level defined in the Technical Specifications. For the full power cases a $F_{\Delta H}$ of 1.38 and a F_Q of 2.38 were used as shown in Table 7.0.1.

7.0.3. Reactivity Coefficients Assumed in the Accident Analyses

The transient response of the reactor system is dependent on reactivity feedback effects, in particular the moderator temperature coefficient and the Doppler power coefficient. In the analysis of certain events, conservatism requires the use of large reactivity coefficient values, whereas in the analysis of other events, conservatism requires the use of small reactivity coefficient values. Tables 7.0.2.A, 7.0.2.B, 7.0.2.C, and 7.0.2.D list the reactivity coefficients used in the transients. Figure 7.0.1 shows, as functions of power, the upper and lower bound Doppler power coefficients used in the transient analysis. In some cases conservative combinations of parameters are used to bound the effects of core life, although these combinations may not represent possible realistic situations.

The B&W analysis was done using the initial value of the Doppler or moderator feedback coefficient throughout the transient. The SAR analysis was done with a variety of reactivity feedback methods. Reference 8 states there are three possible methods of using the Doppler coefficient. Most SARs show curves of Doppler power coefficient versus power level and moderator temperature coefficient versus average RCS temperature change. In most SARs a moderator density coefficient is presented, however in the McGuire SAR only a moderator temperature coefficient is presented when a positive moderator coefficient is used. The effect of this variety of Doppler and moderator coefficient feedback methods is to produce variations in the post trip neutron power versus time curves for the different transients. This variation shows up as a slight difference in results between RELAP5 and the SAR, because the RELAP5 analysis was done using the initial value of the Doppler or moderator feedback coefficient throughout the transient. The analysis comparisons show the B&W method is conservative.

7.0.4. Rod Cluster Control Assembly Insertion Characteristics

The negative reactivity insertion following a reactor trip is a function of the position versus time characteristic of the rod cluster control assemblies and of the variation in rod worth as a function of rod position. With respect to accident analyses, the critical parameter is the time of insertion up to the dashpot entry, approximately 85 percent of the rod cluster travel. The rod cluster control assembly position versus time characteristic assumed in the accident analyses is shown in Figure 7.0.2 for Trojan, RESAR, Catawba, and McGuire. The rod cluster control assembly insertion time to dashpot entry is taken as 2.2 seconds for Trojan/RESAR, 3.05 seconds for Catawba and 3.3/2.7 seconds for McGuire. The 2.7 seconds to 85% insertion for McGuire was used with the four pump coastdown and locked rotor flow coastdown transients. Real drop times are dependent on the type of rod cluster control assemblies actually used in the plant, whether Ag-In-Cd or B_4C. However, accidents are conservatively analyzed using the longer drop time.

Figure 7.0.3 shows the fraction of total negative reactivity insertion versus normalized rod position for a core where the axial distribution is skewed to the lower region of the core for Trojan, RESAR, Catawba, and McGuire. An axial distribution which is skewed to the lower region of the core can arise from an unbalanced xenon distribution. This curve is used to compute the negative reactivity insertion versus time following a reactor trip, which is input to all point kinetics core models used in transient analyses. The bottom skewed power distribution itself is not input into the point kinetics core model.

There is inherent conservatism in the use of the curves in Figure 7.0.2 for Trojan, RESAR, Catawba, and McGuire in that they are based on a skewed flux distribution which would exist relatively infrequently. For cases other than those associated with unbalanced xenon distributions, significant negative reactivity would have been inserted due to the more favorable axial distribution existing prior to trip.

The normalized rod cluster control assembly negative reactivity insertion versus time is shown in Figure 7.0.4 for Trojan, RESAR, Catawba, and McGuire. The curves shown in this figure were obtained from Figure 7.0.2. A total negative reactivity insertion following a trip of 4 percent $\Delta k/k$ is assumed in the transient analyses except where specifically noted otherwise. This assumption is conservative with respect to the calculated trip reactivity worth available. Figure 7.0.5 shows the reactivity versus time for these four cases. The Trojan, Catawba and McGuire curves were used for all of the B&W model analyses.

The normalized rod cluster control assembly negative reactivity insertion versus time curve for an axial power distribution skewed to the bottom (Figure 7.0.4) is used in those transient analyses for which a point kinetics core model is used.

Where special analyses require use of three dimensional or axial one dimensional core models, the negative reactivity insertion resulting from the reactor trip is calculated directly by the reactor kinetics code and is not separable from the other reactivity feedback effects. In this case, the rod cluster control assembly position versus time of Figure 7.0.2 is used as code input.

7.0.5. Trip Points and Time Delays to Trip Assumed in Accident Analyses

A reactor trip signal acts to open two trip breakers connected in series feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanisms to release the rod cluster control assemblies which then fall by gravity into the core. There are various instrumentation delays associated with each trip function, including delays in signal actuation, in opening the trip breakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. Limiting trip setpoints assumed in accident analyses and the time delay assumed for each trip function are given in Table 7.0.3. Further information about Reactor Protection Setpoints is given in Section 5.2.

The difference between the limiting trip point assumed for the analysis and the nominal trip point represents an allowance for instrumentation channel error and setpoint error. Nominal trip setpoints are specified in the plant Technical Specifications. During plant startup tests, it is demonstrated that actual instrument time delays are equal to or less than the assumed values. Additionally, protection system channels are calibrated and instrumentation response times are determined periodically in accordance with the Technical Specification.

7.0.6. Instrumentation Drift and Calorimetric Errors -- Power Range Neutron Flux

The instrumentation drift and calorimetric errors used in establishing the power range high neutron flux setpoint are presented in Table 7.0.4.

The calorimetric error is the error assumed in the determination of core thermal power as obtained from secondary plant measurements. The total ion chamber current (sum of the top and bottom sections) is calibrated (set equal) to this measured power on a periodic basis.

The secondary power is obtained from measurements of feedwater flow, feedwater inlet temperature to the steam generators, and steam pressure. High accuracy instrumentation is provided for these measurements with accuracy tolerances much tighter than those which would be required to control feedwater flow.

7.0.7. Computer Codes Utilized

The RELAP5/MOD2 computer code (Reference 6) was used for the B&W evaluation model. The RELAP5/MOD2 code is a best-estimate transient simulation of pressurized water reactors and associated systems. The modeling capability includes postulated accident simulation for large and small break loss-of-coolant accidents as well as operational transients such as anticipated transients without SCRAM, loss-of-offsite power, loss of feedwater, and loss of flow. A generic modeling approach is utilized which permits as much of a particular system to be modeled as necessary. Control system and secondary system components are included to permit modeling of plant controls, turbines, condensers, and secondary feedwater conditioning systems.

The computer codes equivalent to RELAP5 used by Westinghouse for the transients to be compared were LOFTRAN (References 7 and 8) or MARVEL (Reference 9). A major difference in the computer codes is the use of only a single radial fuel node in LOFTRAN. For these comparisons the B&W model was changed to a single radial fuel node model, however for future analysis the radial pin model for the B&W model will use five nodes for the fuel, one node for the gap and two nodes for the clad. Also, Westinghouse uses several different techniques for Doppler feedback (Figure 7.0.1) and the future analysis using the B&W application model will be done using Doppler feedback as a function of fuel temperature. The LOFTRAN code is also a constant flow model where the flow changes are input and not calculated by the code itself as in RELAP5/MOD2. The B&W model was not altered to produce constant flow, thus there is usually a slightly higher fluid temperature and system pressure calculated by the B&W model due to a slight reduction in flow during the transients. Other minor pressurizer pressure differences may be due to the LOFTRAN pressurizer being a two-node equilibrium model, while the B&W model is an eight-node nonequilibrium model.

The DNBR results shown are calculated by a RELAP5/MOD2 control variable algorithm that considers thermal power, fluid flow, fluid temperature, and system pressure at constant values of power peaking. These DNBR results are for comparison purposes only. The DNB calculations for design analysis will be done using detailed thermal-hydraulics codes where the system parameters will be from the B&W application model.

TABLE 7.0.1.

INPUT PARAMETERS AND INITIAL CONDITIONS FOR TRANSIENTS

PARAMETER		FULL POW EVAL MOD	LOW POW EVAL MOD	CATAWBA	MCGUIRE	RESAR	TROJAN
NEUTRON POWER (10+6 WATTS)		3479 (102%)	17 (0.5%)	3411 (100%)	3411 (100%)	3411 (100%)	3411 (100%)
MODERATOR TEMP	BOL	0 TO 7.0	--	0.0	0 TO 5.0	0.0	0.0
COEFFICIENT (10-5 ΔK/K/F)	EOL	--	--	--	--	--	--
MODERATOR DENSITY	BOL	--	--	--	--	--	--
COEFFICIENT	EOL	--	(1)	--	--	--	--
DOPPLER COEFF	BOL(MIN)	-6.5	--	-6.0	-6.0	-6.0	-6.5
(10-5 ΔK/K/%)	EOL(MAX)	-12.5	(2)	-12.5	-12.5	-12.5	-12.5
EFFECTIVE NEUTRON	BOL	19.40	19.40	19.40	19.40	19.40	19.40
LIFETIME (10-6 SEC)	EOL	18.10	18.10	18.10	18.10	18.10	18.10
DELAYED NEUTRON	BOL	0.0075	0.0075	0.0075	0.0075	0.0075	0.0075
FRACTION	EOL	0.0044	0.0044	0.0044	0.0044	0.0044	0.0044
AVERAGE HEAT FLUX (10+3 BTU/HR-FT2)		198.3	0.972	197.2	197.2		189.8
MAXIMUM HEAT FLUX (10+3 BTU/HR-FT2)		440.9	1.567				
MINIMUM DNBR		2.38(3)	--				
AXIAL POWER DISTRIBUTION		1.68	1.68				
RADIAL POWER DISTRIBUTION		1.38	1.38				
CORE COOLANT FLOW RATE (10+6 LB/HR)		132.5	132.8	138.3	146.1		132.7
CORE COOLANT INLET TEMP (F)		553.28	555.78	560.6	559.2		552.5
CORE AVERAGE COOLANT TEMP (F)		586.05	555.91	590.8	590.1	592.0	584.9

(TABLE 7.0.1. Continued)

CORE COOLANT EXIT TEMP (F)	618.8	556.03				617.2
HOT ASSEMBLY COOLANT EXIT TEMPERATURE (F)	639.53	--				
AVERAGE FUEL TEMP (F)	1162.39	558.2				
MAXIMUM FUEL CENTERLINE TEMP (F)	3230.7	561.48				
REACTOR COOLANT SYSTEM INVENTORY (10+6 LB)	0.5040	0.4987				
COOLANT VOLUME IN PRESSURIZER (FT3)	1080	500	1080	1175	1080	1080
REACTOR COOLANT PRESSURE (PSIA)	2250	2250	2250	2250	2250	2250
STEAM FLOW RATE (10+6 LB/HR)	15.29	0.059	15.15	15.14		15.07
STEAM PRESSURE (PSIA)	874.76	1080	1000	1000		910
STEAM QUALITY	1.0	1.0				
FEEDWATER FLOW RATE (10+6 LB/HR)	15.36	0.075				
FEEDWATER TEMPERATURE (F)	440	79.8	440	440	440	440
TOTAL CONTROL ROD WORTH (%ΔK/K)	4	--	4	4	4	4

(1) FUNCTION OF MODERATOR DENSITY

(2) FUNCTION OF POWER LEVEL

(3) BASED ON RELAP5 CONTROL VARIABLE ALGORITHM

TABLE 7.0.2.A.

75 PCM/SEC BANK WITHDRAWAL
TRANSIENT ANALYSIS ASSUMPTIONS

PARAMETER/COMPARISON	TROJAN	RESAR	CATAWBA	MCGUIRE	B&W
FUEL PIN NODES					
GAP K					
TRIPPED ROD WORTH					
TIME TO 85% INSERTION					
TRW BETA					
MODERATOR COEFFICIENT					
MODERATOR COEFF BETA					
DOPPLER COEFFICIENT					
DOPPLER COEFF BETA					
DOPPLER FEEDBACK METHOD					
DOPPLER FEEDBACK BETA					
ROD WITHDRAWAL RATE					
ROD WITHDRAWAL BETA					
PRZ PORV SETPOINT					
NUMBER PORV VALVES					
PRZ SV SETPOINT					
STEAM SV SETPOINT					
STEAM DUMP ACTUATION					
TURBINE TRIP					
FEEDWATER TRIP					
RSG HTC ADJUSTMENT					

c,d,e

TABLE 7.0.2.B.

FOUR PUMP COASTDOWN
TRANSIENT ANALYSIS ASSUMPTIONS

PARAMETER/COMPARISON	TROJAN	RESAR	CATAWBA	MCGUIRE	B&W
FUEL PIN NODES					
GAP K					
TRIPPED ROD WORTH					
TIME TO 85% INSERTION					
TRW BETA					
MODERATOR COEFFICIENT					
MODERATOR COEFF BETA					
DOPPLER COEFFICIENT					
DOPPLER COEFF BETA					
DOPPLER FEEDBACK METHOD					
DOPPLER FEEDBACK BETA			c, d, e		
ROD WITHDRAWAL RATE					
ROD WITHDRAWAL BETA					
PRZ PORV SETPOINT					
NUMBER PORV VALVES					
PRZ SV SETPOINT					
STEAM SV SETPOINT					
STEAM DUMP ACTUATION					
TURBINE TRIP					
FEEDWATER TRIP					
RSG HTC ADJUSTMENT					

TABLE 7.0.2.C.

TURBINE TRIP
TRANSIENT ANALYSIS ASSUMPTIONS

PARAMETER/COMPARISON	TROJAN	RESAR	CATAWBA	MCGUIRE	B&W
FUEL PIN NODES					
GAP K					
TRIPPED ROD WORTH					
TIME TO 85% INSERTION					
TRW BETA					
MODERATOR COEFFICIENT					
MODERATOR COEFF BETA					
DOPPLER COEFFICIENT					
DOPPLER COEFF BETA					
DOPPLER FEEDBACK METHOD					
DOPPLER FEEDBACK BETA			c,d,e		
ROD WITHDRAWAL RATE					
ROD WITHDRAWAL BETA					
PRZ PORV SETPOINT					
NUMBER PORV VALVES					
PRZ SV SETPOINT					
STEAM SV SETPOINT					
STEAM DUMP ACTUATION					
TURBINE TRIP					
FEEDWATER TRIP					
RSG HTC ADJUSTMENT					

TABLE 7.0.2.D.

LOCKED ROTOR
TRANSIENT ANALYSIS ASSUMPTIONS

PARAMETER/COMPARISON	TROJAN	RESAR	CATAWBA	MCGUIRE	B&W
FUEL PIN NODES					
GAP K					
TRIPPED ROD WORTH					
TIME TO 85% INSERTION					
TRW BETA					
MODERATOR COEFFICIENT					
MODERATOR COEFF BETA					
DOPPLER COEFFICIENT					
DOPPLER COEFF BETA					
DOPPLER FEEDBACK METHOD					
DOPPLER FEEDBACK BETA			c,d,e		
ROD WITHDRAWAL RATE					
ROD WITHDRAWAL BETA					
PRZ PORV SETPOINT					
NUMBER PORV VALVES					
PRZ SV SETPOINT					
STEAM SV SETPOINT					
STEAM DUMP ACTUATION					
TURBINE TRIP					
FEEDWATER TRIP					
RSG HTC ADJUSTMENT					

TABLE 7.0.3.

TRIP POINTS AND TIME DELAYS TO TRIP

TRIP FUNCTION		EVAL MOD	CATAWBA	MCGUIRE	RESAR	TROJAN
HIGH NEUTRON FLUX TRIP (%)	SETPOINT	118	118	118	118	118
	DELAY	0.5 SEC	0.5 SEC	0.5 SEC	0.5 SEC	0.5 SEC
OVERTEMPERATURE ΔT TRIP (F)	K41	1.129				
	K42	0.01702/F				
	TAU41	21.0 SEC				
	TAU42	3.0 SEC				
	TAVGO	584.8 F				
	K43	86 E-5 PSI				
	PO	2250 PSIA				
	DELAY	6.0 SEC				
HIGH PRESSURIZER PRESS TRIP (PSIA)	SETPOINT	2425	2425	2425	2425	2425
	DELAY	2 SEC	2 SEC	2 SEC	2 SEC	2 SEC
LOW REACTOR COOLANT FLOW TRIP (%)	SETPOINT	87	87	87	87	87
	DELAY	1 SEC	1 SEC	1 SEC	1 SEC	1 SEC
UNDERVOLTAGE TRIP(%)	SETPOINT	68	68	68	68	68
	DELAY	1.5 SEC	1.5 SEC	1.5 SEC	1.5 SEC	1.2 SEC

Table 7.0.4.
Catawba SAR Table 15.0.7-1

Determination of Maximum Overpower Trip Point-Power Range Neutron Flux Channel - Based On Nominal Setpoint Considering Inherent Instrumentation Errors

Nominal setpoint (% of rated power) — 109

Calorimetric errors in the measurement of secondary system thermal power:

Variable	Accuracy of Measurement of Variable (% error)	Estimated Effect on Thermal Power Determination (% error)	
Feedwater temperature	±0.5	0.3	
Feedwater pressure (Small correction on enthalpy)	±0.5		
Steam pressure (Small correction on enthalpy)	±2		
Feedwater flow	±1.25	1.25	
Assumed calorimetric error (% of rated power)			2 (1)*
Axial power distribution effects on total ion chamber current			
Estimated error (% of rated power)		3	
Assumed error (% of rated power)			5 (2)*
Instrumentation channel drift and setpoint reproducibility			
Estimated error (% of rated power)		1	
Assumed error (% of rated power)			2 (3)*
*Total assumed error in setpoint (% of rated power) (1) + (2) + (3)			±9
Maximum overpower trip point assuming all individual errors are simultaneously in the most adverse direction (% of rated power)			118

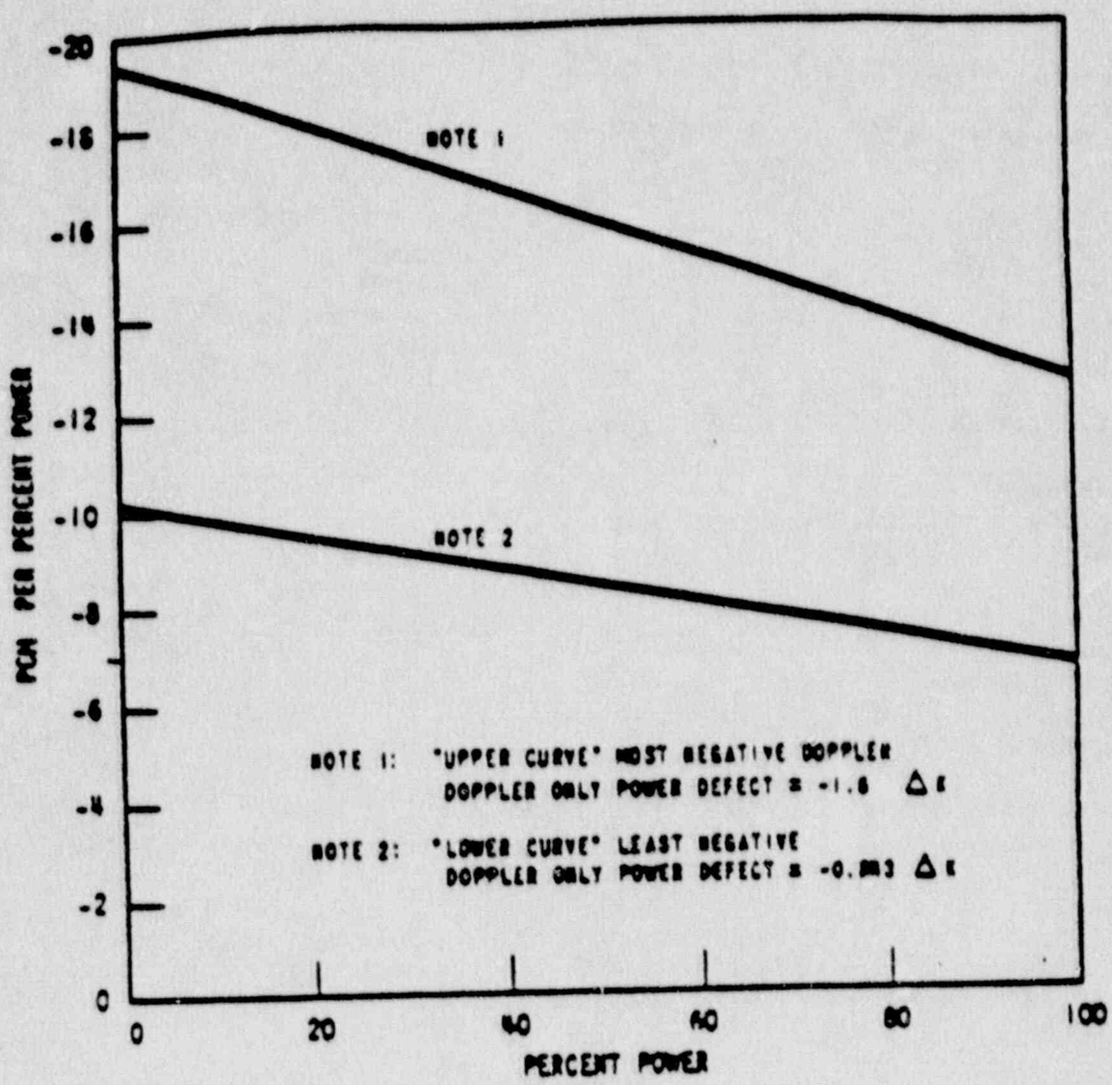


Figure 7.0.1. Trojan SAR Figure 15.0-5 Doppler Power Coefficient Used in Accident Analysis

FIGURE 7.0.2. RCCA POSITION VERSUS TIME TO DASHPOT

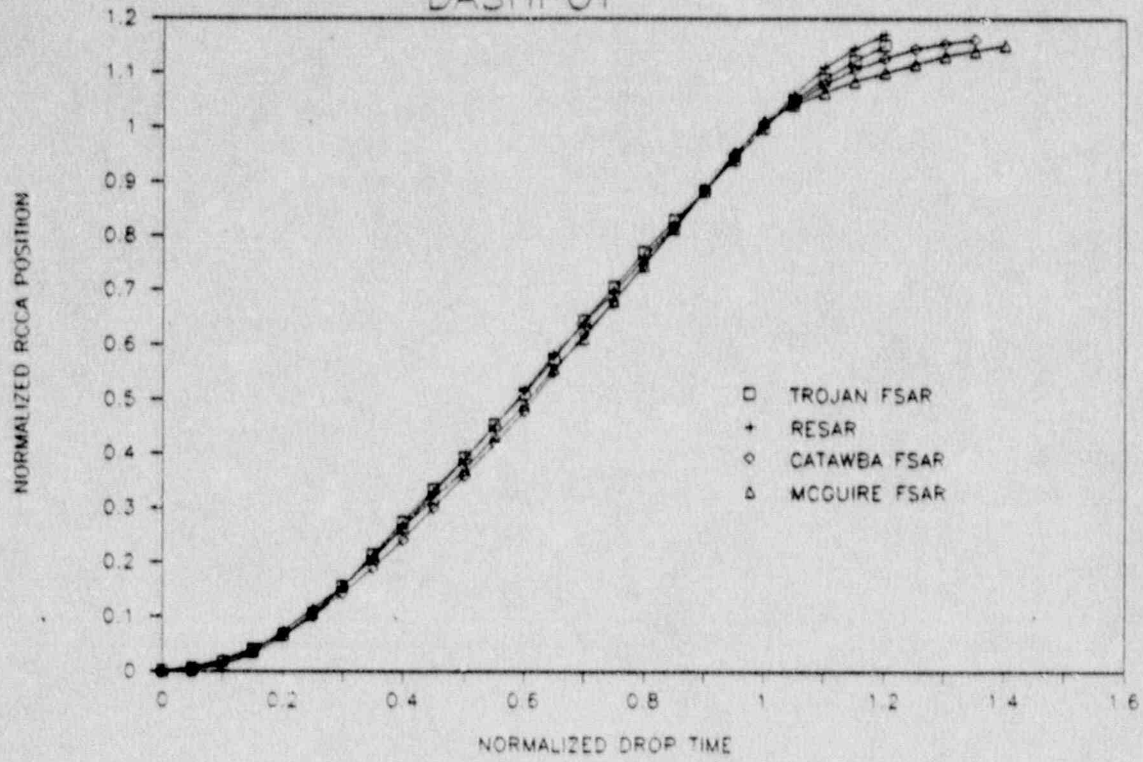


FIGURE 7.0.3. NORMALIZED ROD WORTH VERSUS PERCENT INSERTED

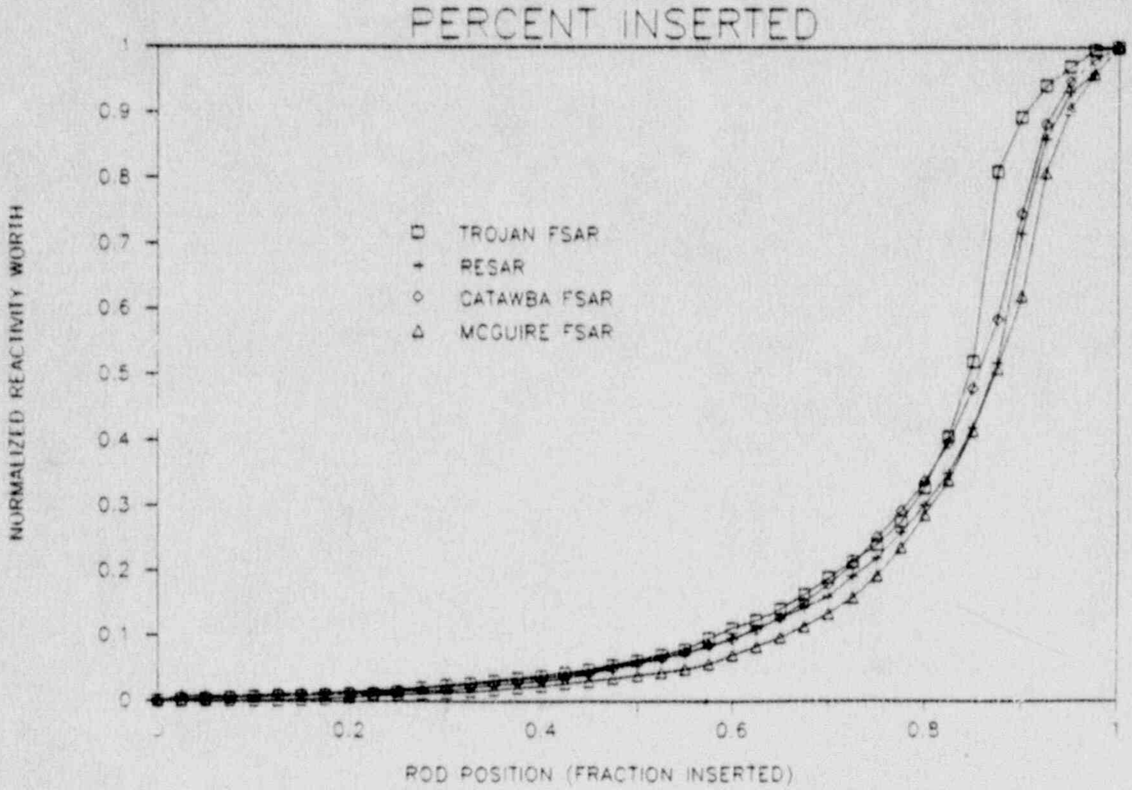


FIGURE 7.0.4. NORMALIZED RCCA BANK REACTIVITY WORTH VERSUS NORMALIZED DROP TIME

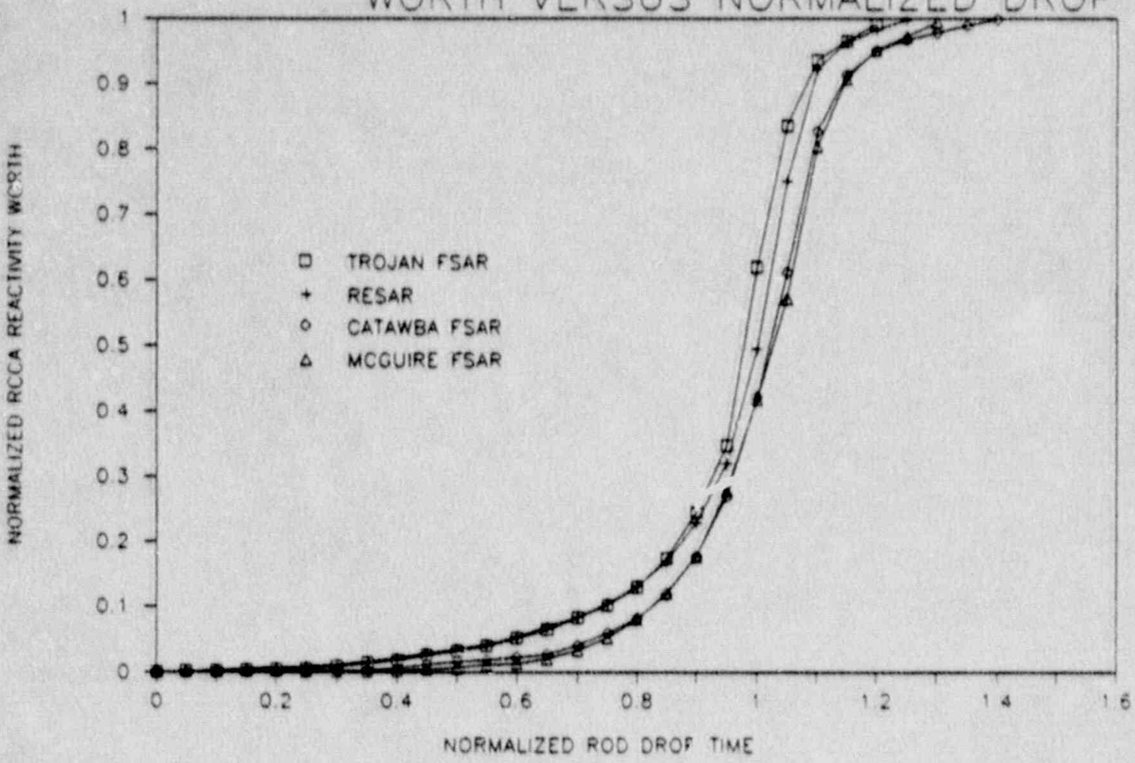
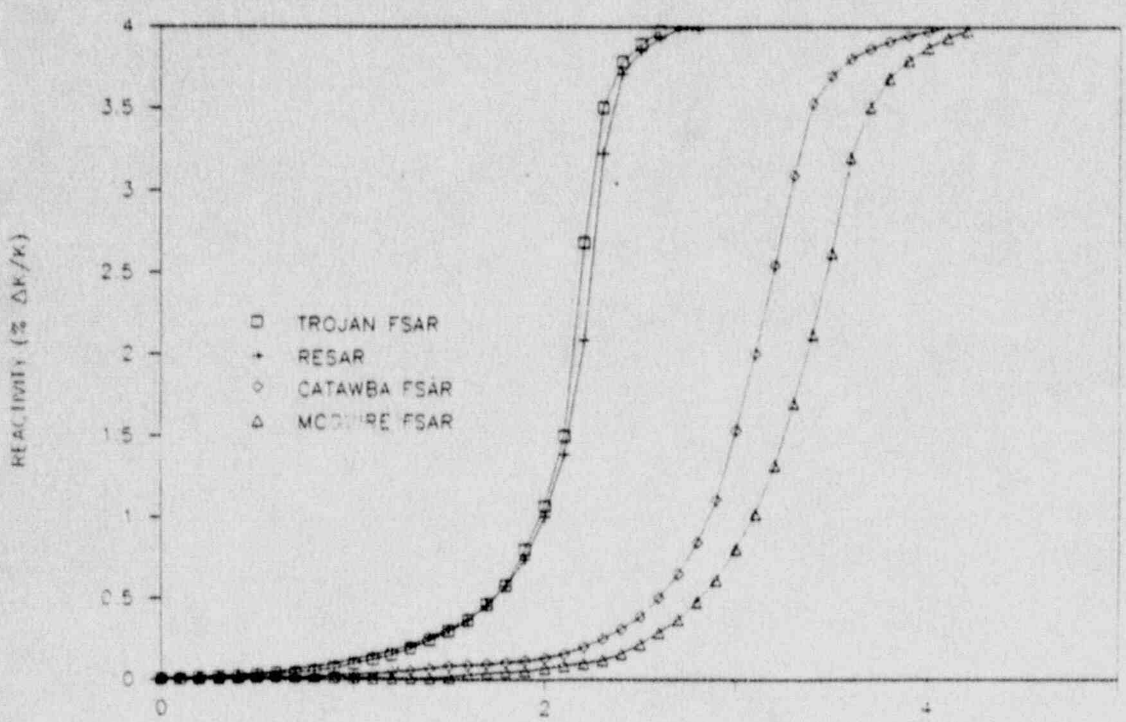


FIGURE 7.0.5. TRIPPED ROD WORTH VERSUS TIME



7.1. 75 pcm/sec Bank Withdrawal

7.1.1. Identification of Causes and Accident Description

The rod withdrawal transient is classified as an ANS Condition II incident (an incident of moderate frequency) as defined in Section 4.0.

Uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power results in an increase in the core heat flux. Since the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in DNB. Therefore, in order to avert damage to the fuel clad the reactor protection system is designed to terminate any such transient before the DNBR falls below the limit value.

The automatic features of the reactor protection system that prevent core damage following the postulated accident include the following:

1. Power range neutron flux instrumentation actuates a reactor trip if two out of four channels exceed an overpower setpoint.

2. Reactor trip is actuated if any two out of four ΔT channels exceed an over-temperature ΔT setpoint. This setpoint is automatically varied with axial power imbalance, coolant temperature and pressure to protect against DNB.

3. Reactor trip is actuated if any two out of four ΔT channels exceed an overpower ΔT setpoint. This

setpoint is automatically varied with axial power imbalance to ensure that the allowable heat generation rate (kw/ft) is not exceeded.

4. A high pressurizer pressure reactor trip actuated from any two out of four pressure channels which is set at a fixed point. This set pressure is less than the set pressure for the pressurizer safety valves.

5. A high pressurizer water level reactor trip actuated from any two out of three level channels when the reactor power is above approximately 10 percent.

In addition to the above listed reactor trips, there are the following RCCA withdrawal blocks:

1. High neutron flux (one out of four power range)
2. Overpower ΔT (two out of four)
3. Over-temperature ΔT (two out of four)

7.1.2. Analysis of Effects and Consequences

Method of Analysis

The transient was analyzed for the safety analysis reports by the LOFTRAN code (Reference 7). This code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The code computes pertinent plant variables including temperature, pressures, and power level.

The RELAP5/MOD2 computer code (Reference 6) was used for the B&W analysis. The RELAP5/MOD2 code is a best-estimate transient simulation of pressurized water reactors and associated systems. A generic modeling approach is utilized which permits as much of a particular system to be modeled as necessary. Control system and secondary system components are included to permit modeling of plant controls, turbines, condensers, and secondary feedwater conditioning systems. The code simulates the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, steam generator, and steam generator safety valves. Through the use of control variables any plant parameter can be computed.

Initial and Boundary Conditions

Initial reactor power, pressure and RCS temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR analysis as described in Section 7.0.

The maximum positive reactivity insertion rate is greater than that for the simultaneous withdrawal of the combination of the two control banks and the maximum combined worth at maximum speed.

A conservatively small (in absolute magnitude) value Doppler power coefficient is used in the comparison analysis. For the B&W application model analysis the Doppler reactivity feedback consistent with the rod withdrawal time in life was used. A moderator coefficient of zero was used for all analysis.

The reactor trip on high neutron flux is assumed to be actuated at a conservative value of 118 percent of nominal full power. The ΔT trips include all adverse instrumentation and setpoint errors; the delays for trip actuation are assumed to be the maximum values.

The effect of RCCA movement on the axial core power distribution is accounted for by causing a decrease in over-temperature ΔT trip setpoint proportional to a decrease in margin to DNB. The RCCA trip insertion characteristic is based on the assumption that the highest worth assembly is stuck in its fully withdrawn position.

7.1.3. Results

From the safety analysis standpoint, the rod withdrawal event is a relatively straightforward transient. The key phenomena are the core power increase accompanying the rod withdrawal and the heatup and expansion of the primary coolant. The most significant parameters are the pre- and post-trip power, the core fluid temperature, and pressurizer pressure. The important output is departure from nucleate boiling (DNB) ratio, affected during the event by increased local core heat flux and fluid temperature.

Safety Analysis Report Data

Each of the Safety Analysis Reports presents the results in terms of plots of the key system parameters: neutron power fraction, thermal power fraction, system pressure, pressurizer water volume, core average temperature, and DNB ratio. Figures 7.1.1 through 7.1.6 show SAR results for a 75 pcm/sec rod withdrawal for the Catawba, McGuire, and Trojan plants and for the Westinghouse RESAR. The Trojan data are presented only to 5 seconds, and no plots for pressurizer water volume or thermal power are available from the Trojan SAR or from the RESAR. The results can be differentiated readily in terms of the tripped rod insertion times used for the analyses. The Catawba/McGuire cases used 3.05/?.3 seconds to 85 percent insertion versus 2.2 seconds for the Trojan and RESAR safety analyses. This grouping is obvious in the neutron power fraction curves of Figure 7.1.1. The other parameters align directly according to tripped rod insertion times, with one exception. The core average temperatures, Figure 7.1.5, show the Trojan and RESAR temperatures peaking later than those for Catawba and McGuire. This relationship is at odds with the other parameters, most notably the DNBR curves. A possible explanation is that the core average temperature plots for the Trojan and RESAR results are based upon averaging the cold leg inlet and hot leg outlet temperatures whereas the Catawba and McGuire temperatures are literally the core volume temperatures.

Comparison Cases

Results of the RELAP5 comparison analyses for RESAR/Trojan and Catawba/McGuire are presented in Figures 7.1.7 through 7.1.18. The SAR analyses are based upon constant loop flow during the transient, but the RELAP5 calculations included active loop flows using the pump model. The relatively small changes in flow due to changes in fluid density are shown in Figures 7.1.11 and 7.1.18. The DNBR comparisons were done using an algorithm to generate DNBR as a RELAP5 control variable as opposed to the detailed thermal-hydraulic analyses done for the safety analysis reports using the Westinghouse CHF correlations. Thus, the DNBR results are presented only for comparison of trends and timing.

The RELAP5 results for the Trojan and RESAR cases follow the SAR curves directly. Because of the similarity between the conditions and assumptions for Trojan and the RESAR, a single RELAP5 case was run. The RELAP5 cases were analyzed assuming both actuation of the steam dump system and without steam dump. As the results show, there is virtually no effect associated with the steam dump for this event.

In examining Figures 7.1.1 and 7.1.3, it is noteworthy that the Trojan post-trip power drops off more steeply than the RESAR and RELAP5 curves. This is likely due to a difference in reactivity feedback modeling specific to that analysis for Trojan. There is a corresponding effect upon peak pressure, shown in the second figure, but the difference in DNBR is not significant.

Separate RELAP5 analyses were run for the Catawba and McGuire comparisons shown in Figures 7.1.12 through 7.1.18. The Catawba case and the McGuire case reflect the different tripped rod insertion times. The RELAP5 calculation for McGuire also included a +5.0 pcm moderator coefficient as was used in the SAR analysis. The figures show generally excellent agreement between the RELAP5 and SAR results. Furthermore, the effect of tripped rod insertion time indicated in the original safety analyses is reflected correspondingly in the RELAP5 cases. This is a small effect, and it is discernible after the point of minimum DNBR. The RELAP5 calculations allowed simulation of the effect of fluid temperature upon pump performance, and this shows in the core flow fraction plot of Figure 7.1.18. The flow variation is small -- it is not accounted for in the SAR analyses -- and has some effect upon the calculated thermal power after about three seconds. This accounts for the greater net expansion to the pressurizer calculated by RELAP5 (Figure 7.1.15) and for the slightly less rapid dropoff in core average temperature at the end of the transient.

B&W Application Model

Figures 7.1.19 through 7.1.25 show RELAP5 analysis results for the B&W model application. The only differences between the comparison RELAP5 runs and the B&W application are:

1) The B&W model has a five region fuel pin, a 1 region gap and a 2 region clad model.

2) The B&W model uses a fuel temperature Doppler reactivity feedback and a Doppler coefficient consistent with the rod withdrawal time in life.

Reactor trip on high neutron flux occurs shortly after the start of the accident. Since this is rapid with respect to the thermal time constants of the plant, only small changes in average temperature and pressure occur, thus margin to DNB is maintained. The neutron power results are significantly different from the SAR analysis because the Doppler coefficient was input as a fuel temperature coefficient instead of a power coefficient. The fuel temperature lags the neutron power and less negative reactivity feedback occurs early in the t sient, thus the neutron power rises faster and the overpower trip occurs sooner.

After the reactor trip, the fuel temperature feedback adds to the negative rod drop reactivity to produce a very rapid power reduction.

Figure 7.1.20 shows the core heat flux does not exceed 118% of its nominal value. Figure 7.1.24 shows the DNBR versus time for the 75 pcm/sec rod withdrawal as calculated by the RELAP5 code algorithm. The minimum DNBR is never less than the limit value.

7.1.4. Conclusions

The RELAP5 model has shown agreement with SAR rod withdrawal analysis results for several different plants and is therefore valid for the analysis of any specific plant when the appropriate plant geometry and plant parameters are modeled.

FIGURE 7.1.1. 75 PCM/SEC ROD WITHDRAWAL

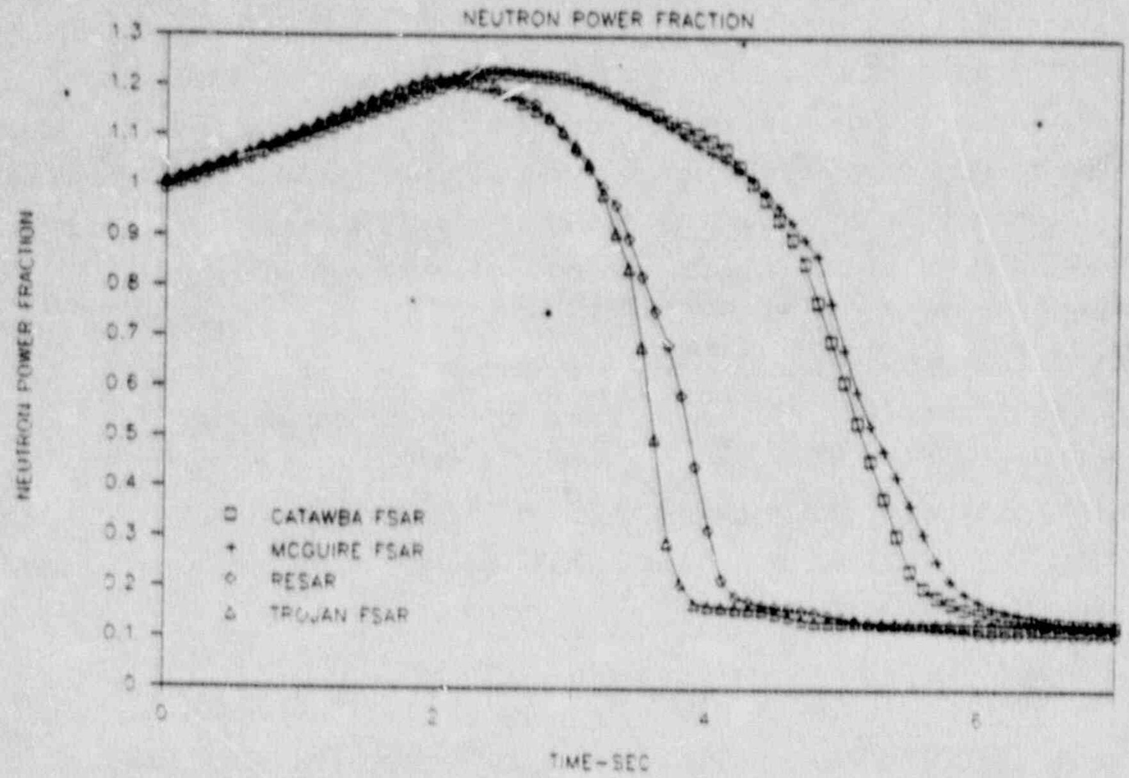


FIGURE 7.1.2. 75 PCM/SEC ROD WITHDRAWAL

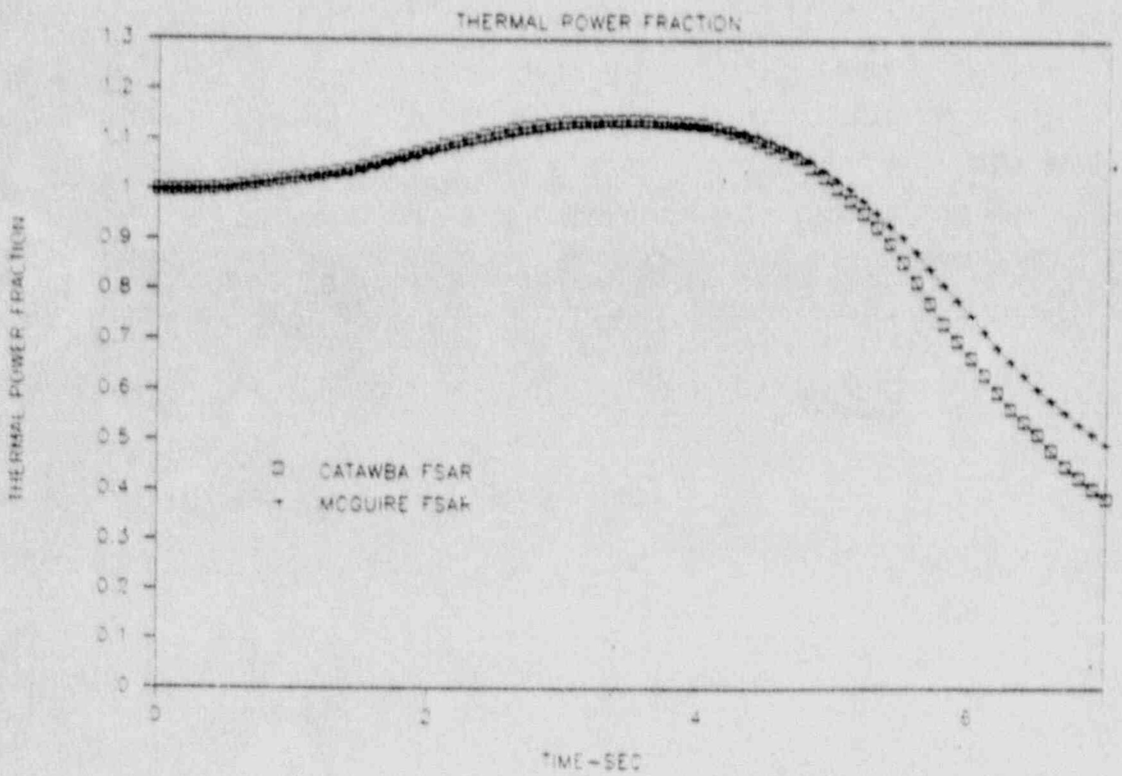


FIGURE 7.1.3. 75 PCM/SEC ROD WITHDRAWAL

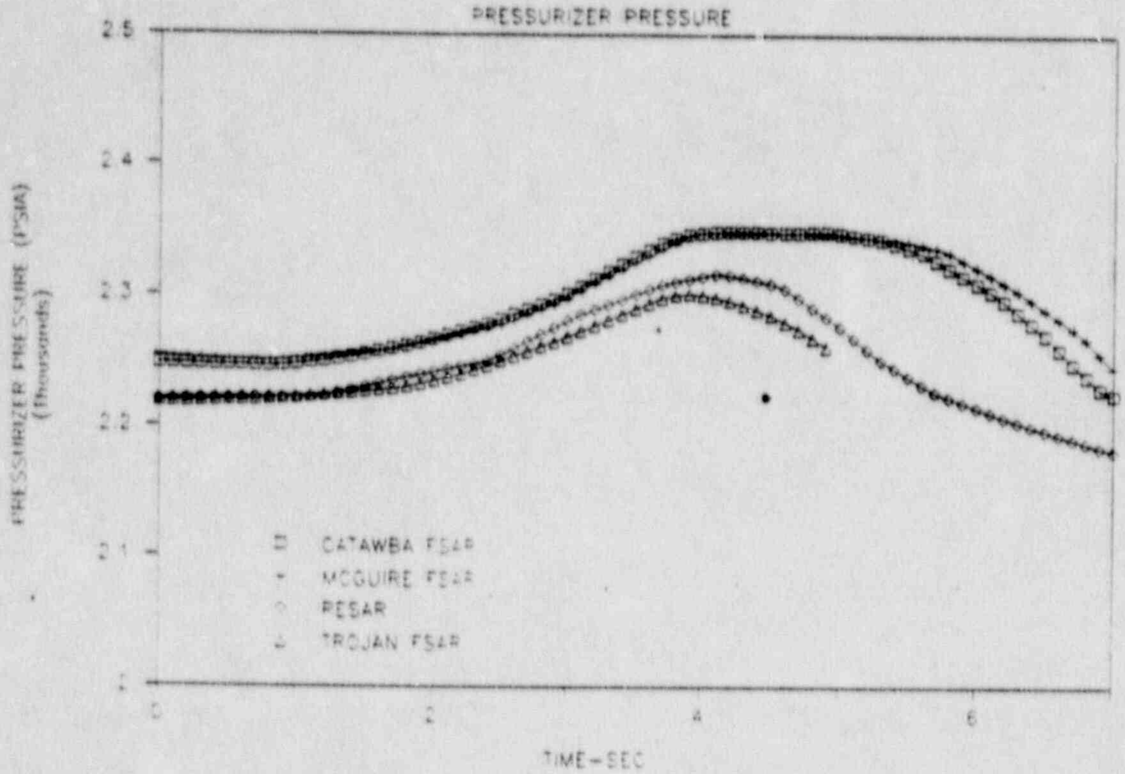


FIGURE 7.1.4. 75 PCM/SEC ROD WITHDRAWAL

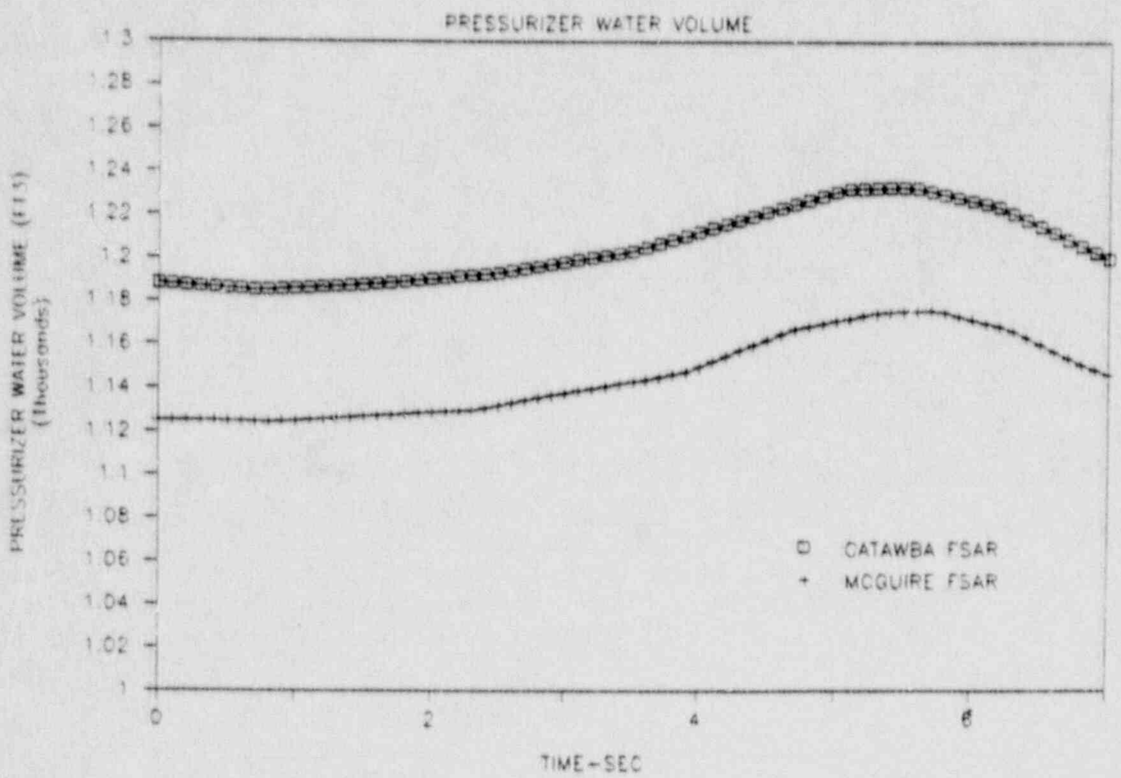


FIGURE 7.1.5. 75 PCM/SEC ROD WITHDRAWAL

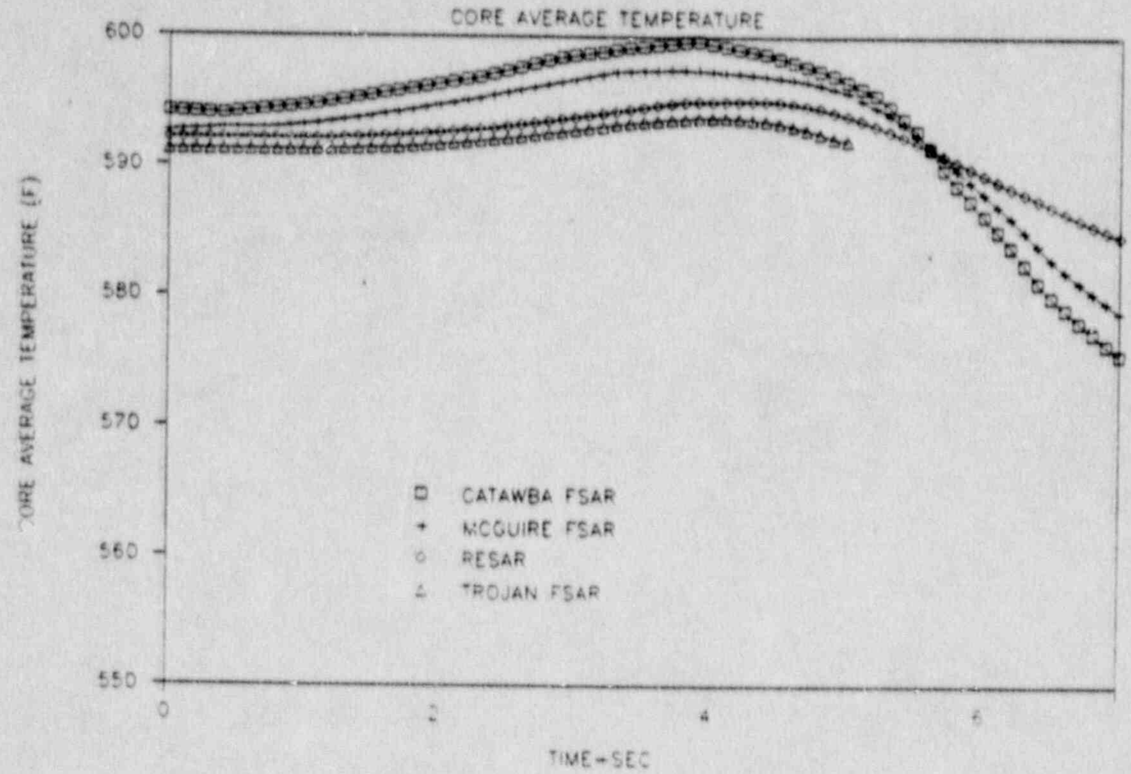


FIGURE 7.1.6. 75 PCM/SEC ROD WITHDRAWAL

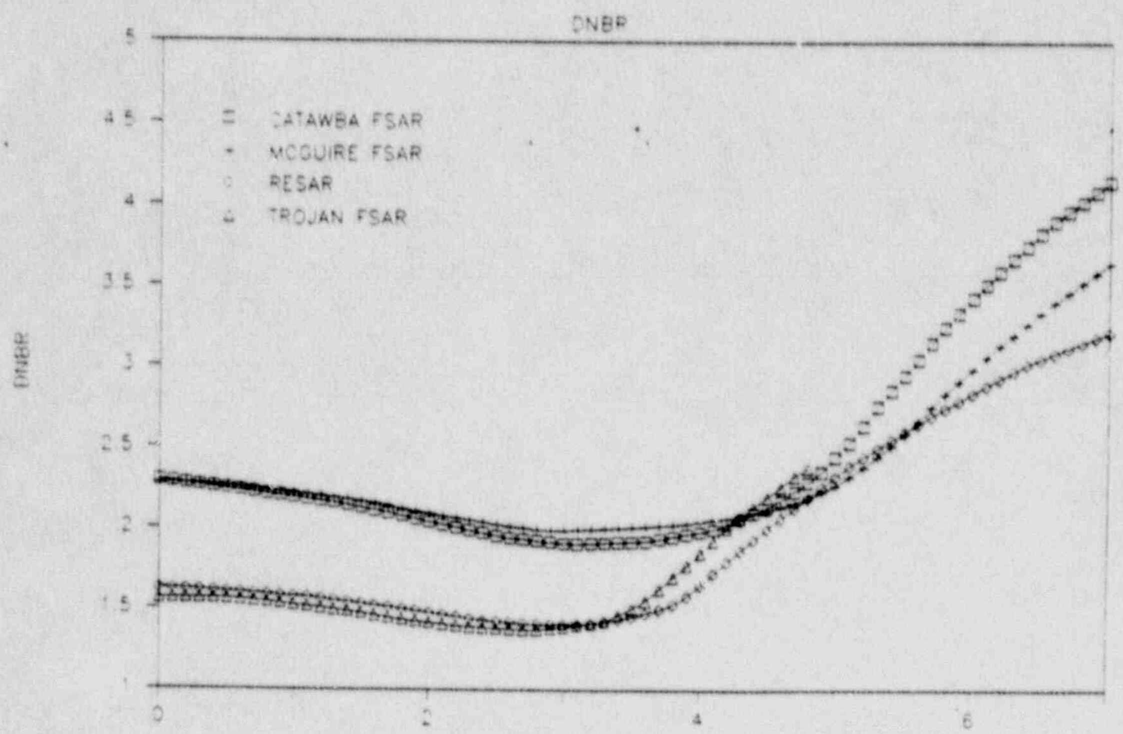


FIGURE 7.1.7. 75 PCM/SEC ROD WITHDRAWAL

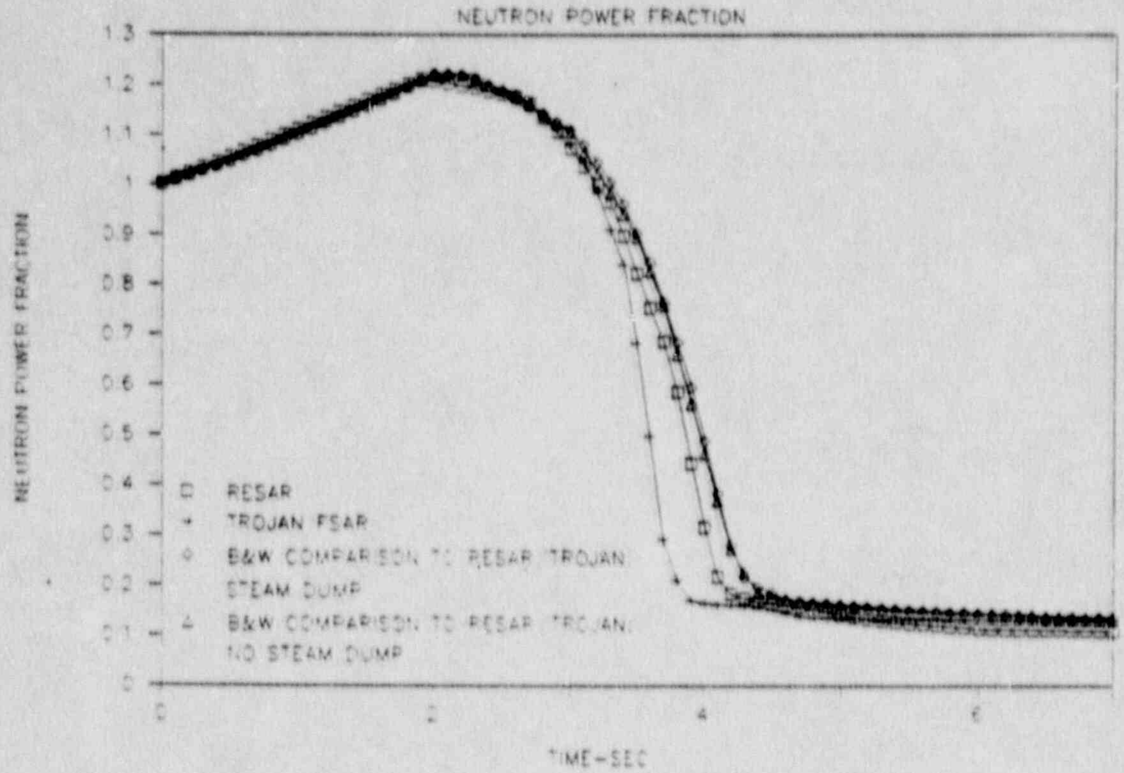


FIGURE 7.1.8. 75 PCM/SEC ROD WITHDRAWAL

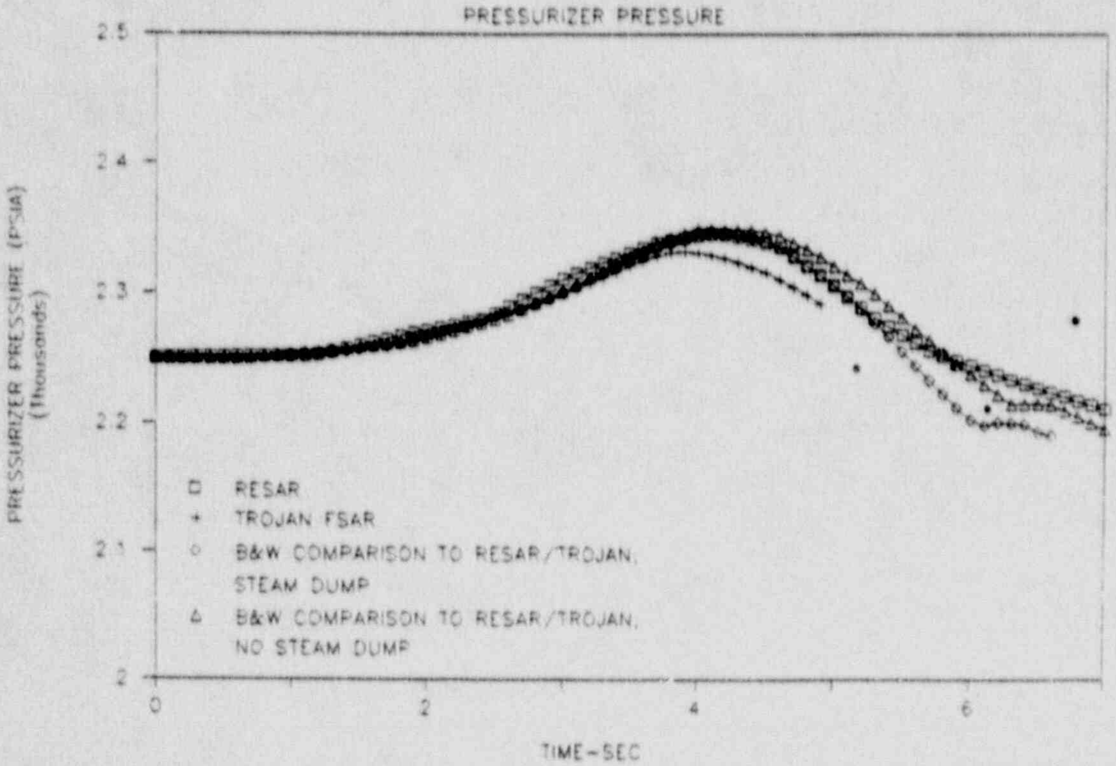


FIGURE 7.1.9. 75 PCM/SEC ROD WITHDRAWAL

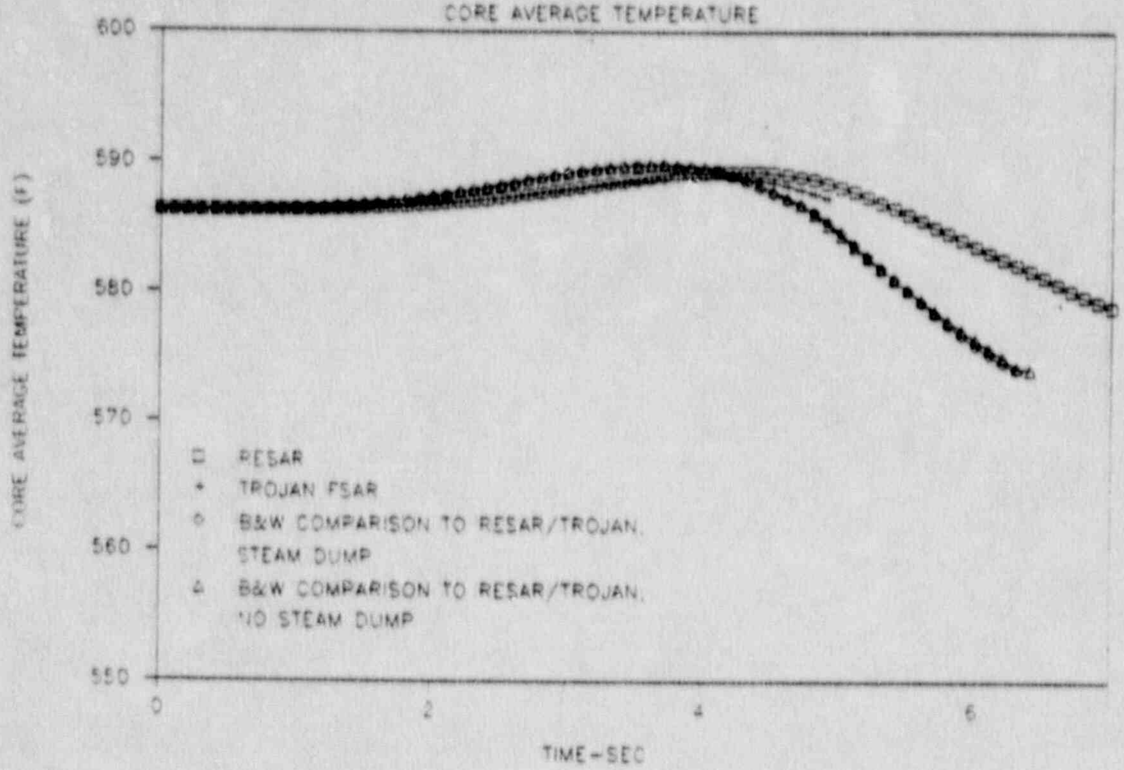


FIGURE 7.1.10. 75 PCM/SEC ROD WITHDRAWAL

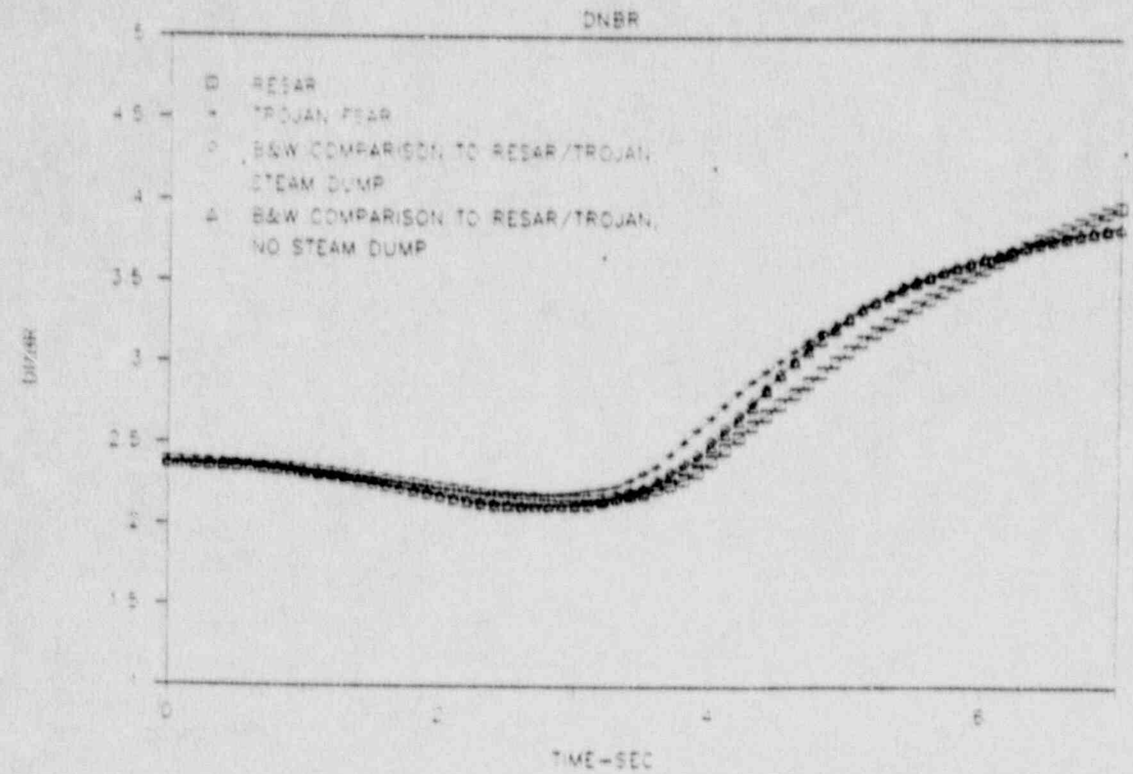


FIGURE 7.1.11. 75 PCM/SEC ROD WITHDRAWAL

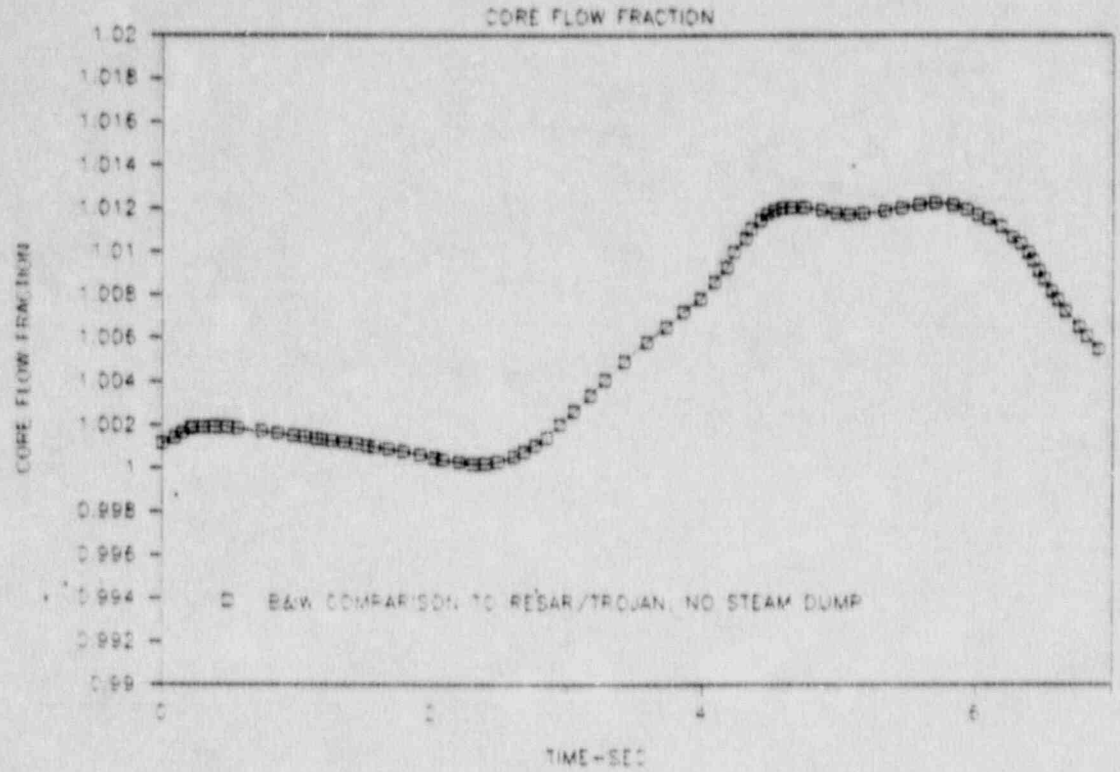


FIGURE 7.1.12. 75 PCM/SEC ROD WITHDRAWAL

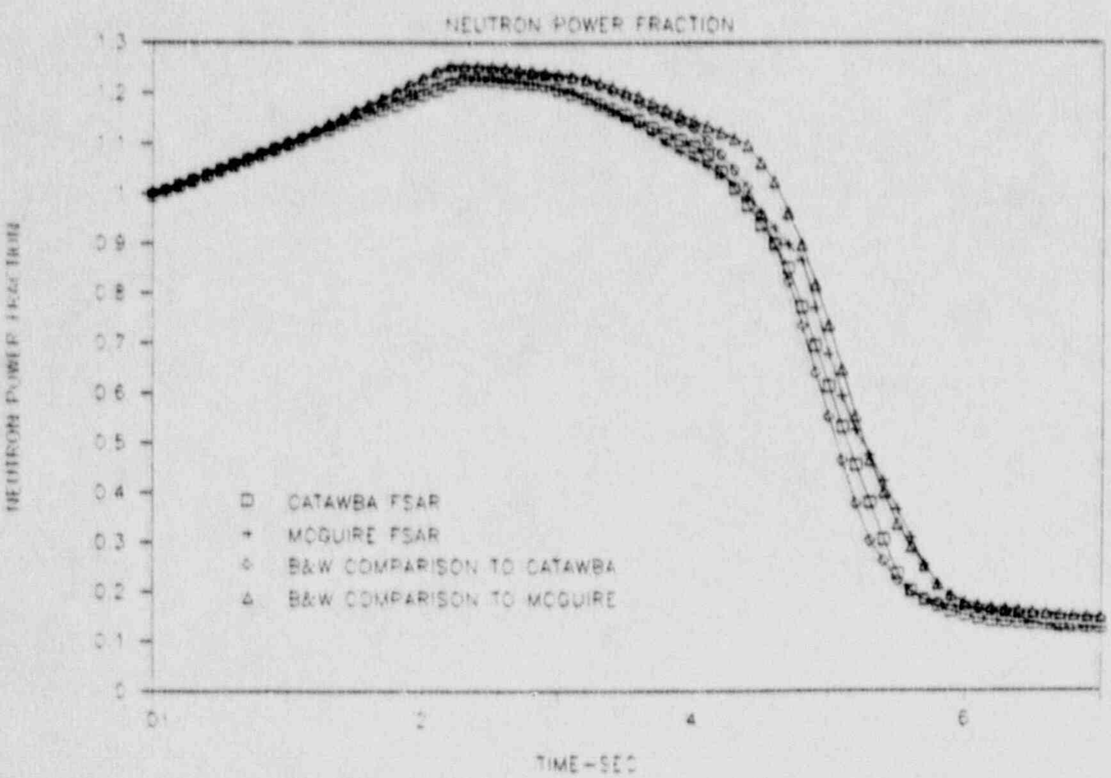


FIGURE 7.1.13. 75 PCM/SEC ROD WITHDRAWAL

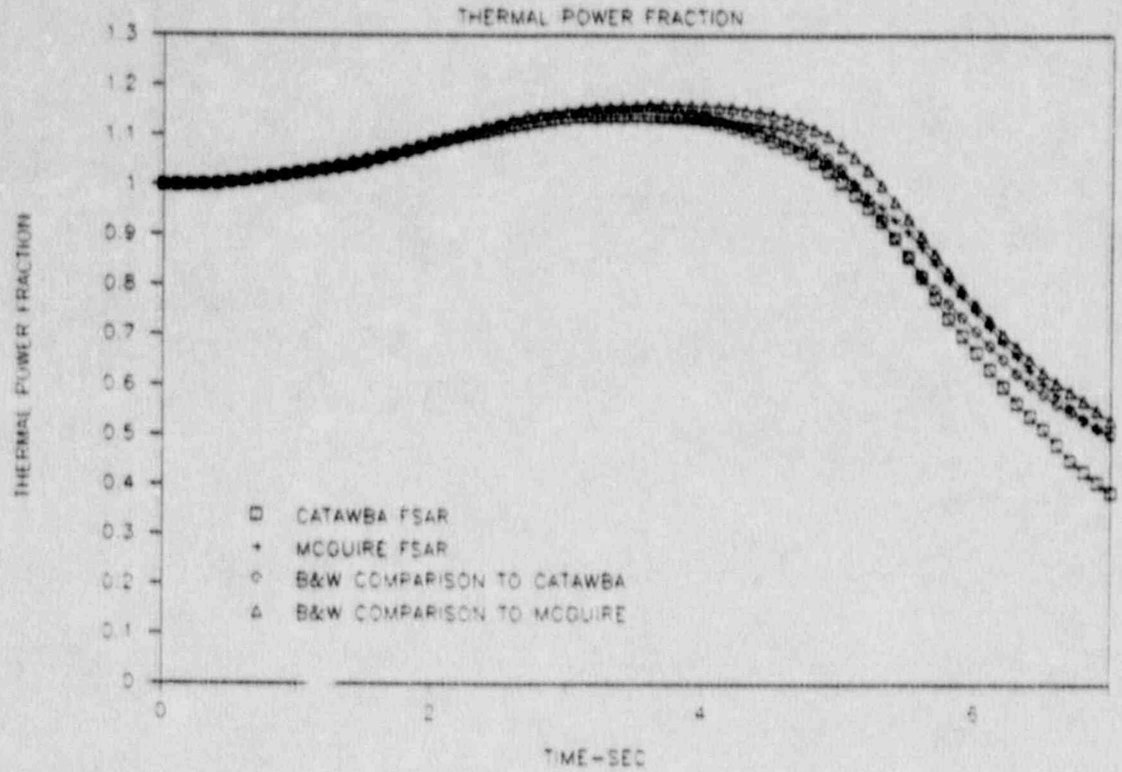


FIGURE 7.1.14. 75 PCM/SEC ROD WITHDRAWAL

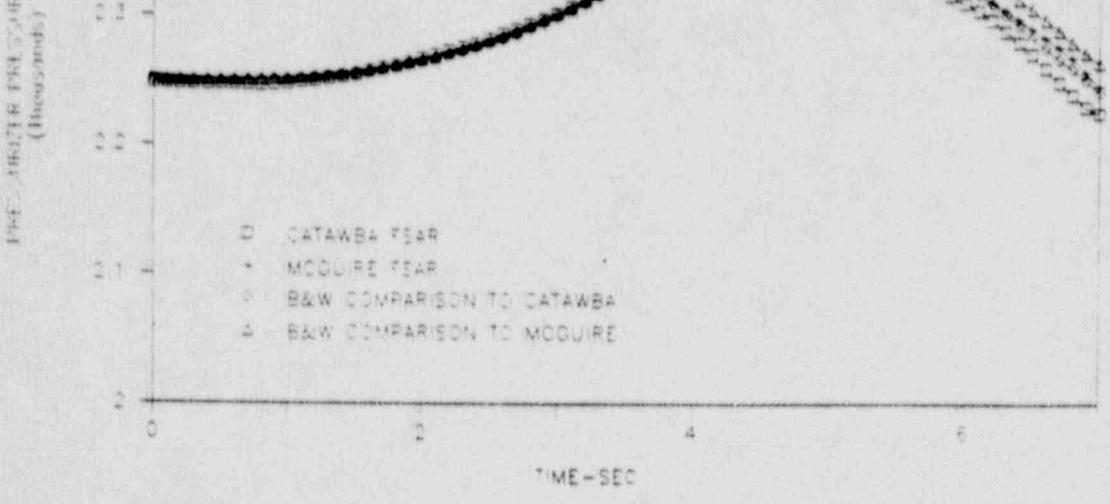


FIGURE 7.1.15. 75 PCM/SEC ROD WITHDRAWAL

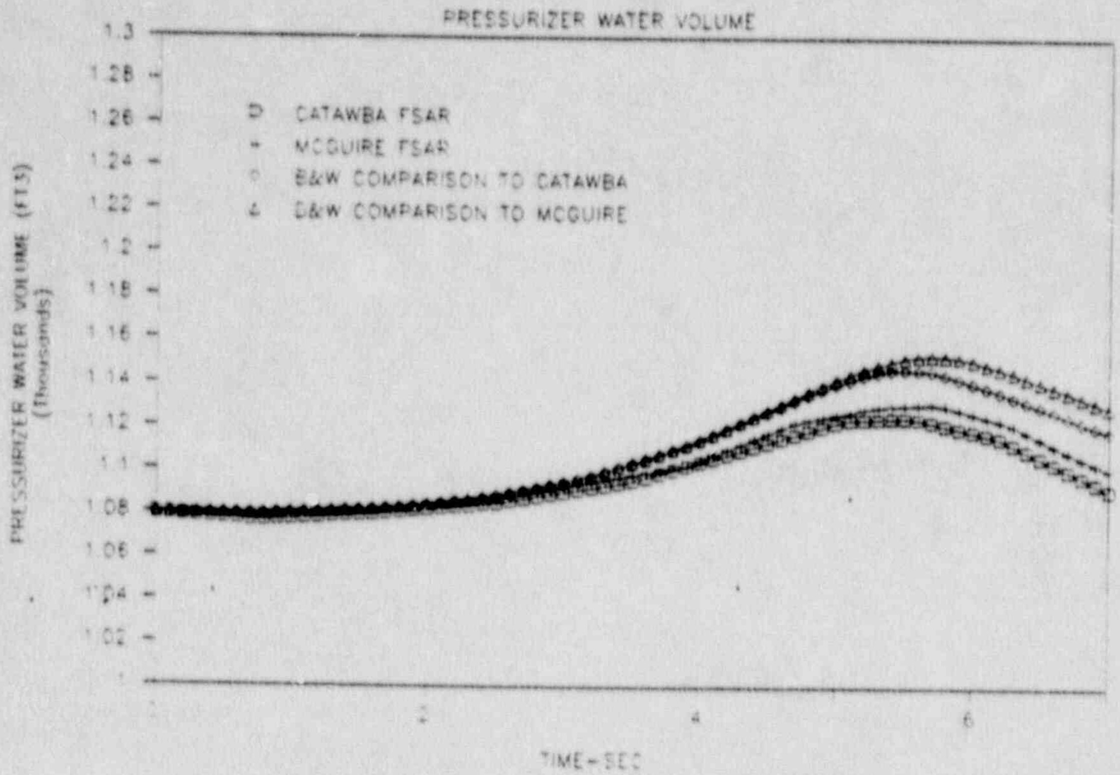


FIGURE 7.1.16. 75 PCM/SEC ROD WITHDRAWAL

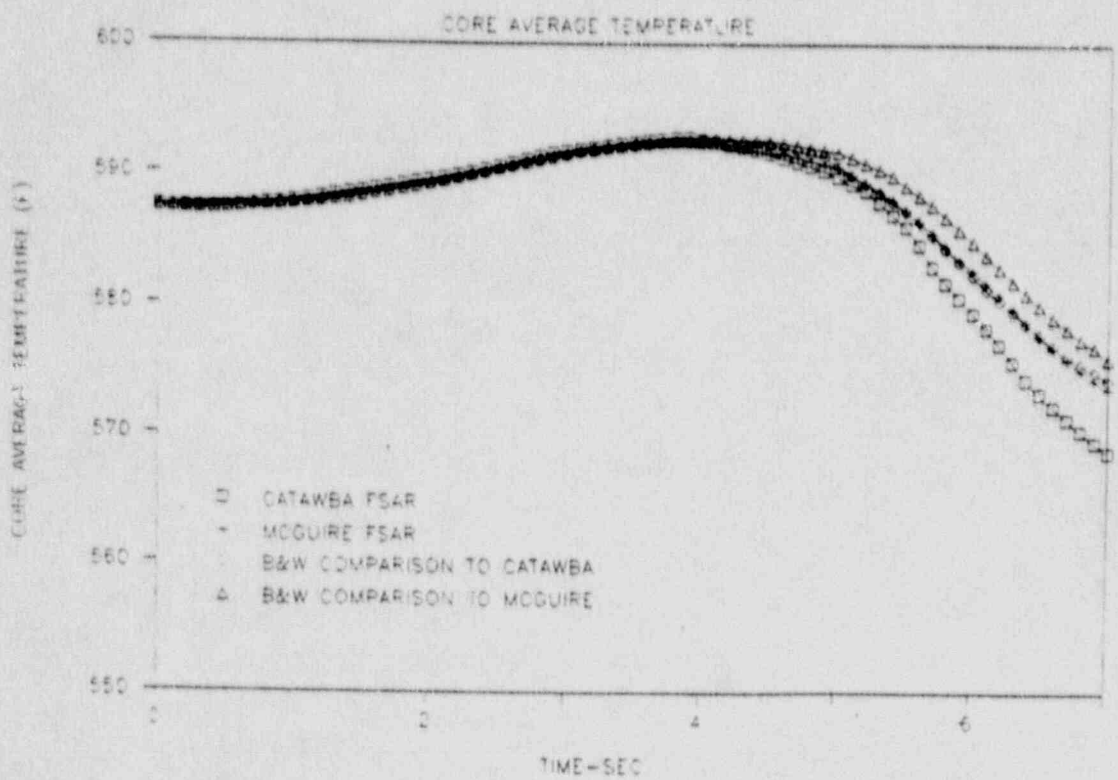


FIGURE 7.1.17. 75 PCM/SEC ROD WITHDRAWAL

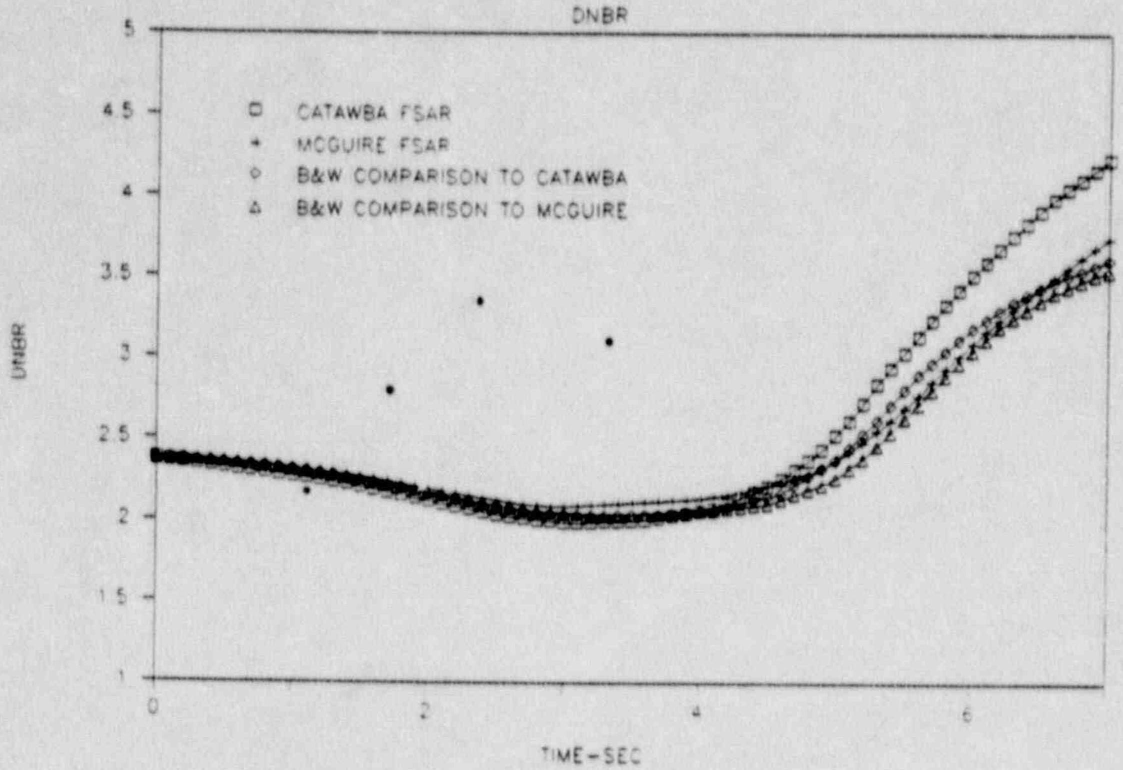


FIGURE 7.1.18. 75 PCM/SEC ROD WITHDRAWAL

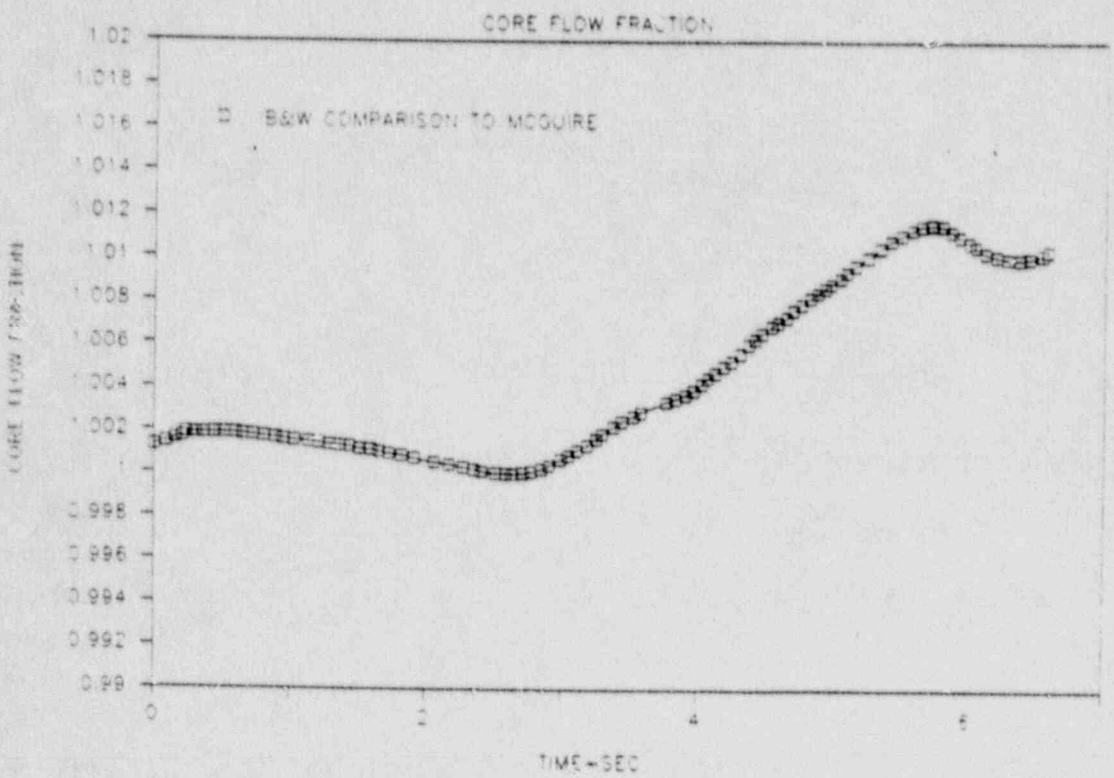


FIGURE 7.1.19. 75 PCM/SEC ROD WITHDRAWAL

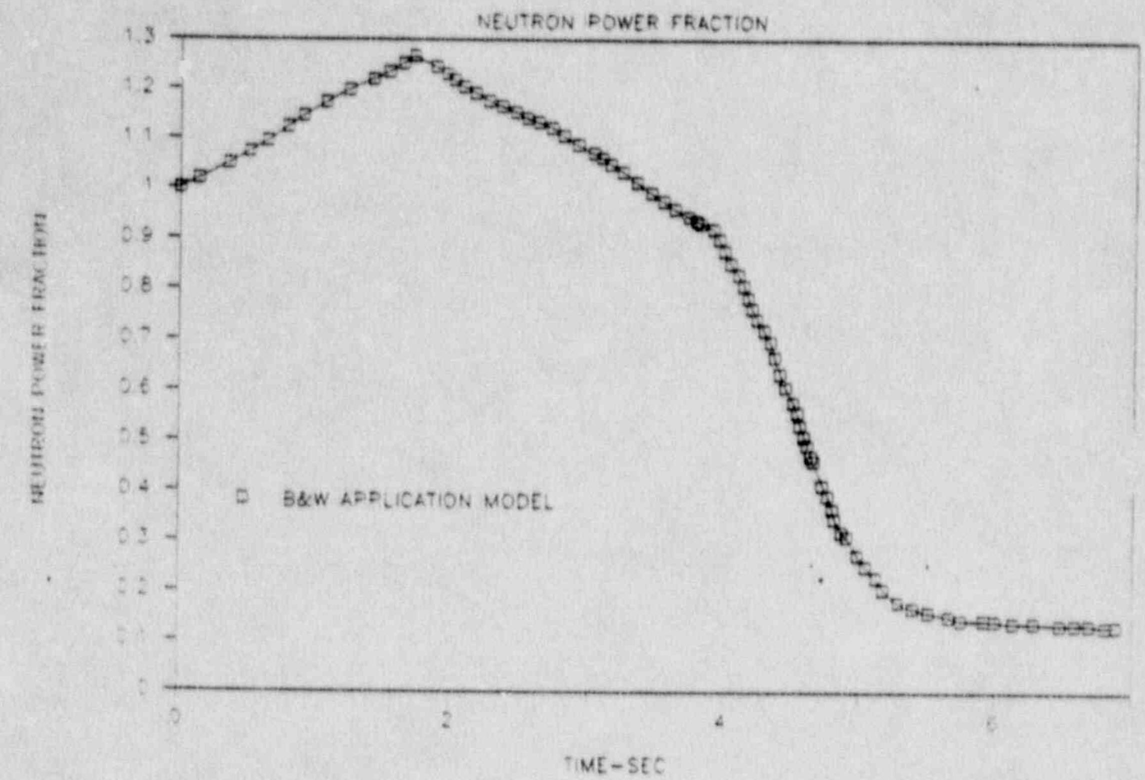


FIGURE 7.1.20. 75 PCM/SEC ROD WITHDRAWAL

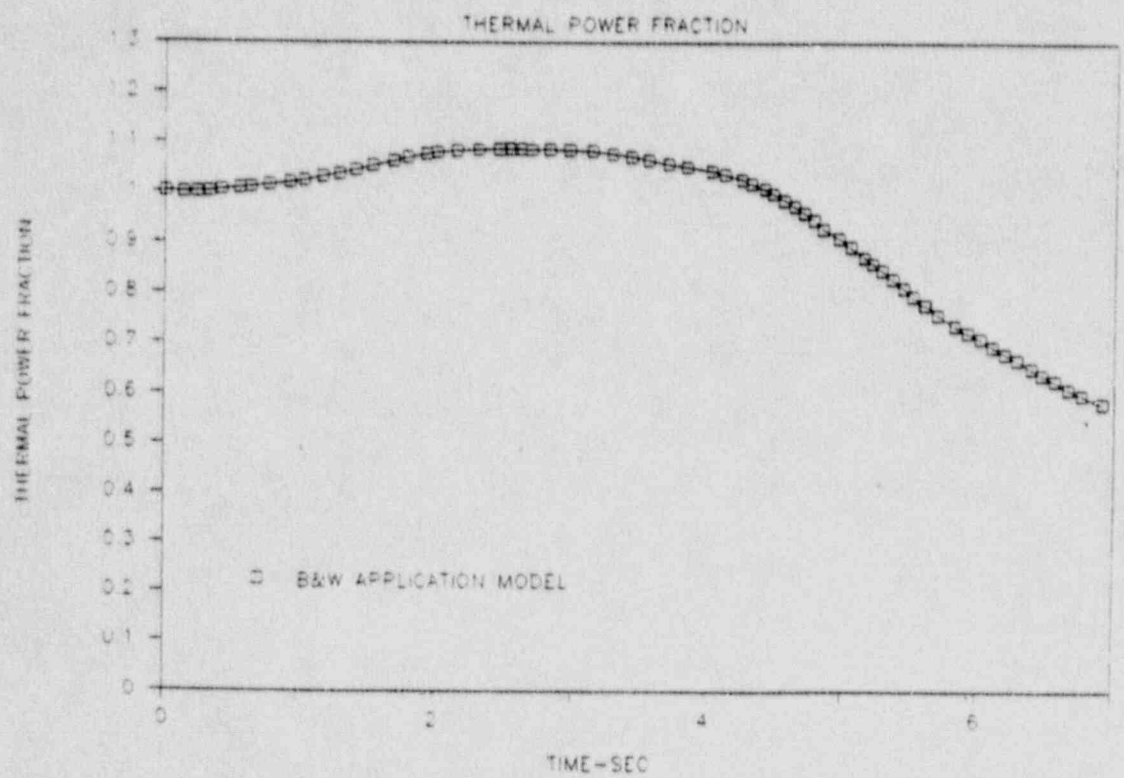


FIGURE 7.1.21. 75 PCM/SEC ROD WITHDRAWAL

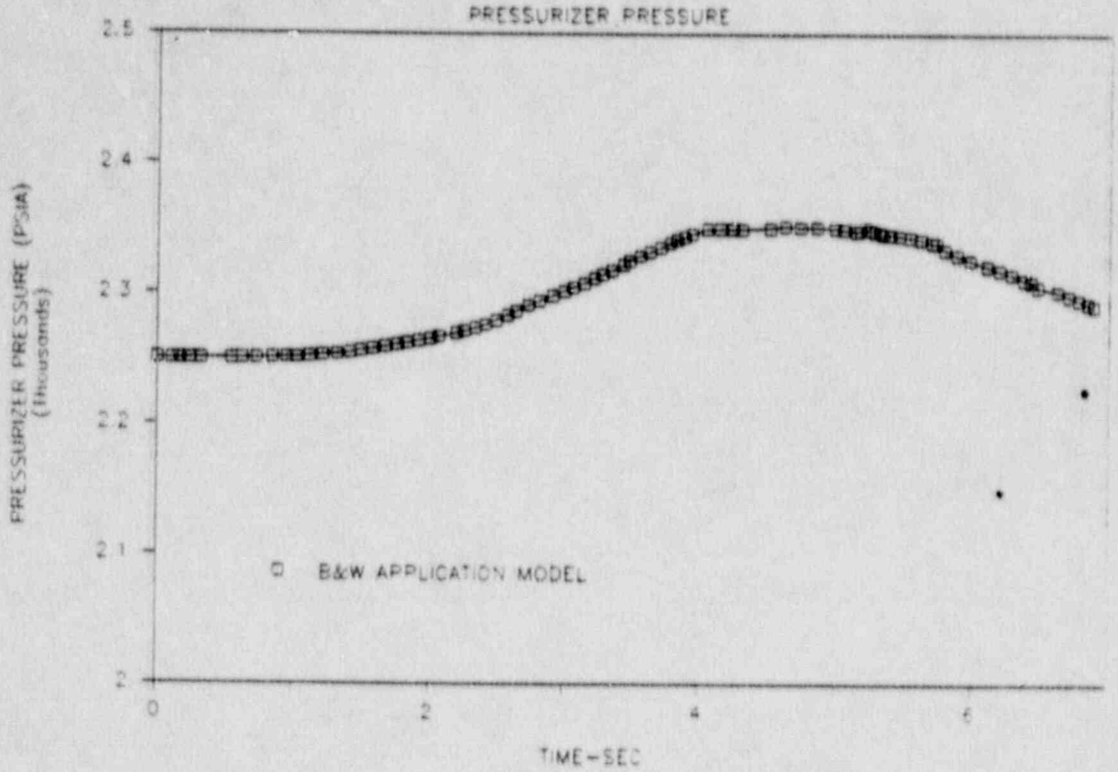


FIGURE 7.1.22. 75 PCM/SEC ROD WITHDRAWAL

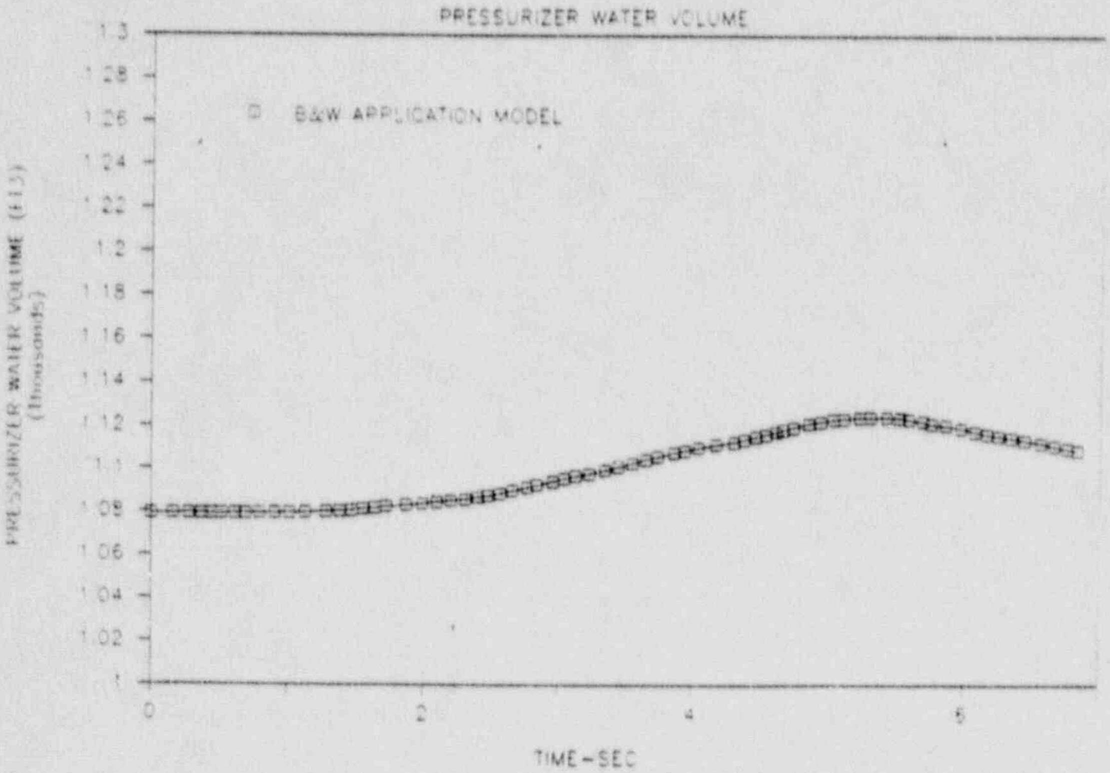


FIGURE 7.1.23. 75 PCM/SEC ROD WITHDRAWAL

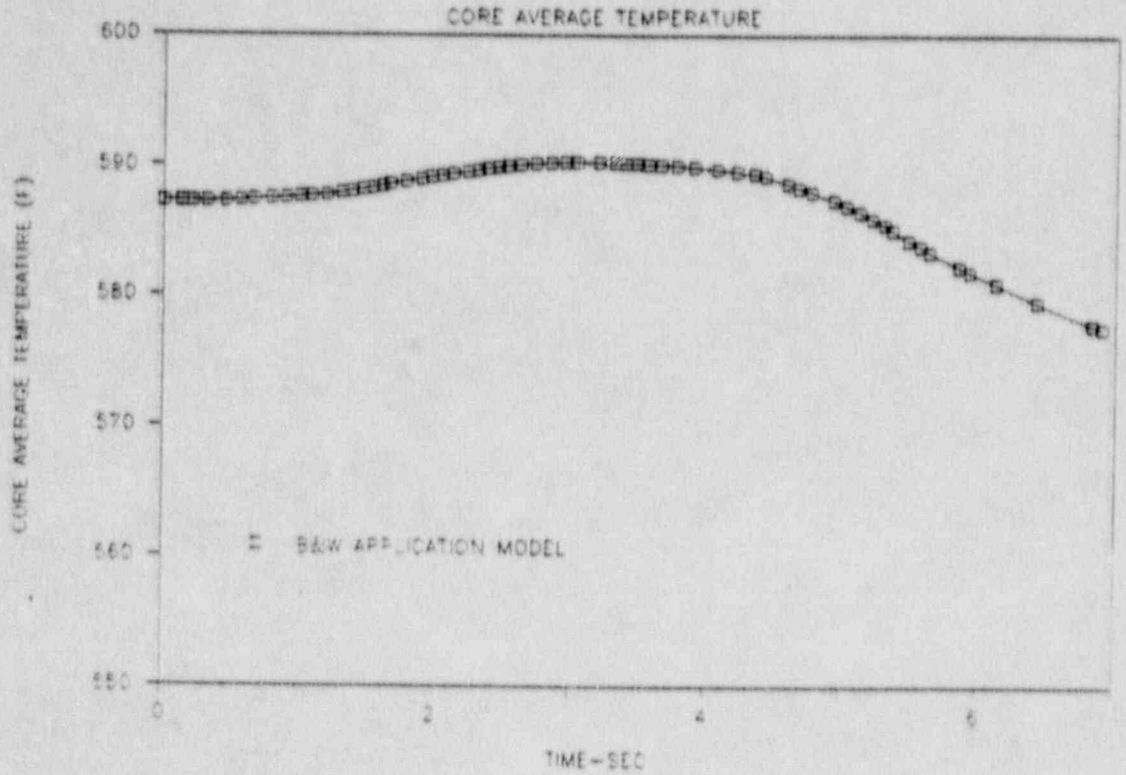


FIGURE 7.1.24. 75 PCM/SEC ROD WITHDRAWAL

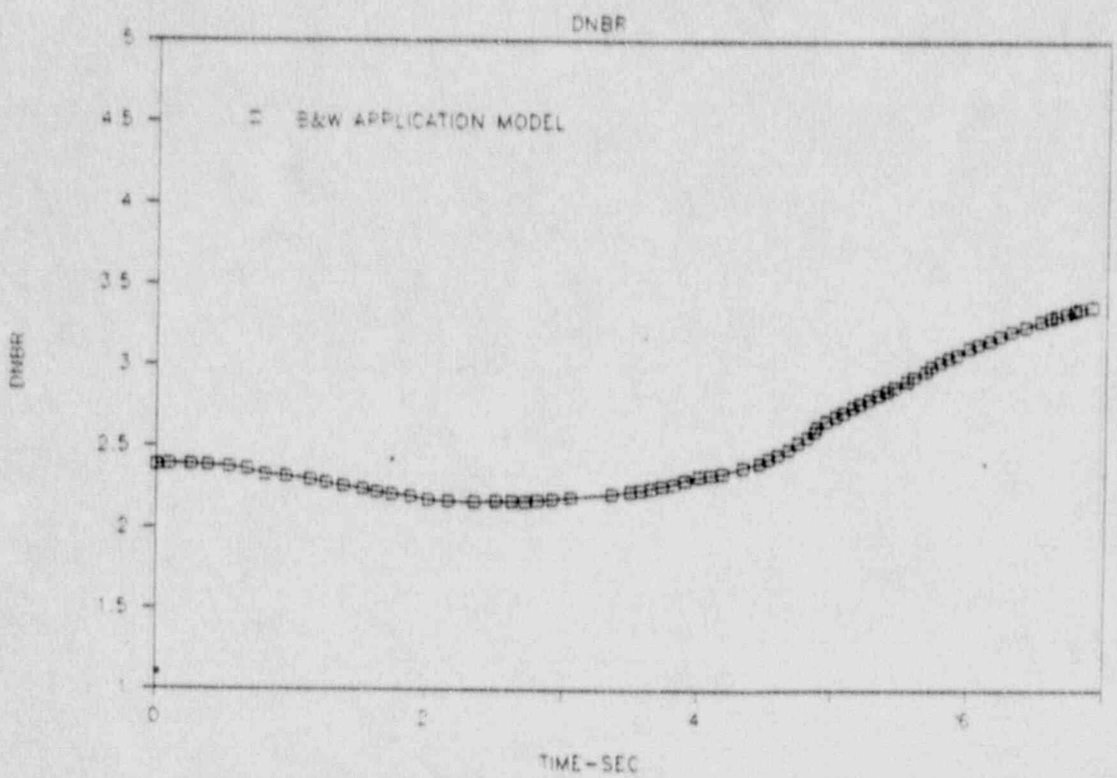
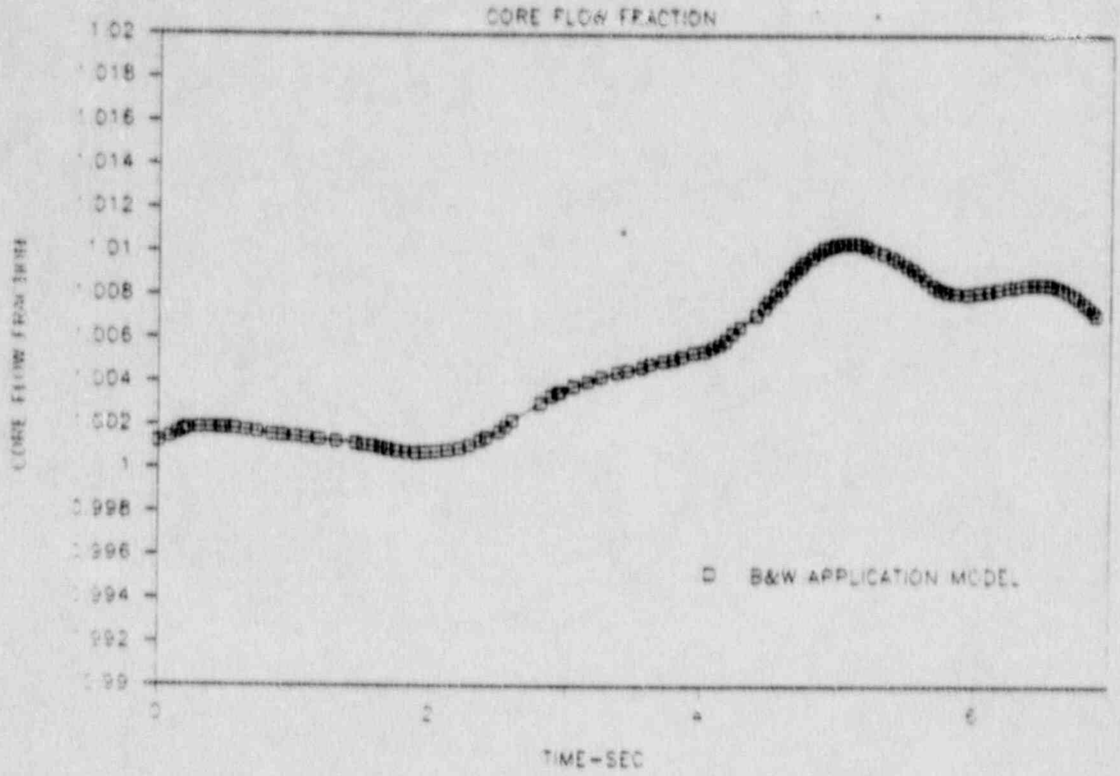


FIGURE 7.1.25. 75 PCM/SEC ROD WITHDRAWAL



7.2. Complete Loss Of Forced Reactor Coolant Flow

The complete loss of flow event, or four-pump coastdown, is the limiting DNBR event in the Condition III reduced flow category. The key phenomena are the primary system flow coastdown and interaction with core power. Reactor trip is actuated effectively at time zero, so the core power response is determined by trip delay time and rod insertion rate. The drop in thermal power, surface heat flux at the fuel rods, lags the neutron power, and DNBR decreases due to reduced flow and increased core inlet temperature.

A complete loss of forced reactor coolant flow is classified as an ANS Condition III event, an infrequent fault. See Section 4.0 for a discussion of Condition III events.

7.2.1. Identification of Causes and Accident Description

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical supplies to all reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of loss of coolant flow is a rapid increase in the coolant temperature. This increase could result in DNB within the core with subsequent fuel damage were the reactor not tripped in time.

The following reactor signals provide the necessary protection against a complete loss of flow accident: (1) reactor coolant pump power supply undervoltage or underfrequency and (2) low reactor coolant loop flow.

The reactor trip on reactor coolant pump undervoltage is provided to protect against conditions which can cause a loss of voltage to all reactor coolant pumps, i.e., station blackout. This function is blocked below approximately 10 percent power.

The reactor trip on reactor coolant pump underfrequency is provided to trip the reactor for an underfrequency condition, resulting from frequency disturbances on the power grid.

7.2.2. Analyses of Effects and Consequences

Methods of Analysis

This transient was analyzed in the SARs by several digital computer codes. For the McGuire SAR the loop and core flows are calculated by the LOFTRAN code (Reference 7) based on the measured flow coastdown data taken during startup tests. LOFTRAN is also used to calculate the time of reactor trip based on the flows, the nuclear power transient, and the primary system pressure and temperature transients. The FACTRAN code (Reference 10) is then used to calculate the heat flux transient based on the nuclear power and flow from LOFTRAN. Finally, the THINC code (Section 4.4 of References 1 and 2) is used to calculate the DNBR during the transient based on the heat flux from FACTRAN and flow from LOFTRAN. The DNBR transients presented represent the minimum of the typical or thimble cell.

The B&W analysis was done using the RELAP5/MOD2 code (Reference 6). A pump speed versus time curve was used to produce a core flow coastdown equivalent to the flow coastdown of the SAR analysis. For design analysis measured flow coastdown data taken during startup tests will be used. The B&W model DNBR results shown were calculated by a RELAP5/MOD2 control variable algorithm that considers thermal power, fluid flow, fluid temperature, and system pressure at constant values of power peaking. These DNBR results are for comparison purposes only. The DNB calculations for design analysis will be done using detailed thermal-hydraulics codes where the system parameters will be from RELAP5/MOD2.

Initial and Boundary Conditions

Initial reactor power, pressure and RCS temperature are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limit DNBR analysis as described in Section 7.0.

A conservatively large absolute value of the Doppler-only power coefficient is used in the SAR analysis. This is equivalent to a total integrated Doppler reactivity from 0 to 100 percent of 0.016 $\Delta k/k$. For the B&W application analysis a Doppler fuel temperature coefficient consistent with the time in life for this analysis was used.

A zero moderator coefficient was used for the Trojan, RESAR and Catawba analyses. A positive moderator temperature coefficient of +5 pcm/F was used for the McGuire analysis. A positive moderator temperature coefficient of +7.0 pcm/F was used for the B&W application analysis. The use of a positive moderator temperature coefficient is very conservative since this results in the maximum core power during the initial part of the transient when the minimum DNBR is reached.

Following the loss of power supply to all pumps at power, a reactor trip can be actuated by either reactor coolant pump power supply undervoltage or underfrequency. For these analyses the reactor is assumed to be tripped on an undervoltage signal.

7.2.3. Results

Safety Analysis Report Data

The results of the analyses performed for the four-pump coastdown transient as reported in the plant SAR's and in the RESAR are presented in Figures 7.2.1 through 7.2.5. No pressurizer pressure curves were reported for Trojan or for the RESAR.

The most significant differences among the results are attributable to differences in tripped rod worth insertion times and in trip delays. The Trojan and RESAR cases assumed 2.2 seconds to 85 percent insertion, whereas the Catawba and McGuire results are based upon 3.05 seconds and 2.7 seconds, respectively. This effect is immediately discernible in the neutron power versus time plots of Figure 7.2.1.

There are other differences among the SAR neutron power calculations that are not apparent in the SAR plots but, nonetheless, affect the comparative RELAP5 analyses. For example, different methods of using the power Doppler coefficient may have been used as were different moderator coefficients. For McGuire, a moderator density coefficient was not used, whereas a positive moderator temperature coefficient was included in the kinetics calculations. For consistency, the RELAP5 comparative analyses were all done with the initial feedback coefficient taken as a constant throughout the transient.

As in other cases, there are large variations in the magnitudes of the DNB ratios presented in the SAR's and in the RESAR. These are primarily due to the applications of different critical heat flux correlations -- the W-3 correlation versus WRB-1, for instance -- and different thermal design methods. The important indices of comparison are the magnitudes of DNB ratio reduction during the event and the relative times of occurrence of the minimum DNBR's.

Trojan/RESAR Comparison

RELAP5 results are compared to the Trojan and RESAR curves in Figures 7.2.6 through 7.2.9 for the major four-pump coastdown parameters. Again, pressurizer pressure plots were not available from these safety analysis reports. The distinction between the two cases as analyzed using the RELAP5 model is in the trip delay times. The effect of the steam dump system was included in the

RELAP5 analyses, but this has no effect upon the transient results through the time of minimum DNBR.

Comparison of the SAR plots to the RELAP5 outputs for the major parameters is straightforward. The agreement is quite good, and the relative magnitudes of DNBR suppression and times to minimum DNBR are consistent. The operative difference between the Trojan and RESAR cases is the trip delay time, and the RELAP5 results show the same effect as presented in the SAR's.

It is noteworthy that the neutron power calculation for the Trojan case produced by the RELAP5 model does not show the same drop-off just before 3 seconds as does the SAR curve. This is most probably due to the use of power or temperature dependent feedbacks in the Trojan calculation that were not included in the RELAP5 inputs. A constant initial value of minimum Doppler was applied throughout the transient. This approach did produce finer agreement with the RESAR neutron power fraction and did not affect the parameter of concern -- the DNBR -- in any noticeable way.

Catawba/McGuire Comparison

Figures 7.2.10 through 7.2.19 show the comparisons of RELAP5 analyses to the Catawba and McGuire SAR curves for the loss of flow event. The significant differences between the two plant analyses are in the times to 85 percent tripped rod worth insertion and in the moderator coefficients. The Catawba results are based upon 3.05 seconds to 85 percent insertion using a zero moderator coefficient, and the McGuire comparison is based upon 2.7 seconds for the tripped rod worth insertion with a +5 pcm/F moderator coefficient. The steam dump system is modeled as 70 percent of nominal steam flow for each case. In addition to the direct comparison cases, results are presented for separate sensitivity analyses: For the Catawba calculation, two additional curves are presented for each parameter showing the effect of the rate of steam release. For McGuire, the additional results are plotted to show the effect of the tripped rod insertion time.

The Catawba results are presented in Figures 7.2.10 through 7.2.14. The constant feedback modeling used for the RELAP5 calculation produces more rapid rolloff in neutron power fraction between about 2 seconds and 4 seconds than shown in the SAR plot. This effect is carried through the other parameters--the RELAP5 thermal power is slightly lower during that period, and the pressurizer peak pressure is later and about 20 psi lower--but the baseline comparison case (for 70 percent steam dump) agrees well with the Catawba SAR calculation.

Also shown for the Catawba case are RELAP5 results for the same event assuming zero steam dump and steam release at full flow. While the rate of steam release does affect the time and magnitude of peak pressurizer pressure, as shown in Figure 7.2.13, the most important output, the minimum DNBR, is unaffected. That is, the limiting DNB condition occurs before the steam dump system has any impact upon primary system conditions.

The McGuire comparative analyses are shown in Figures 7.2.15 through 7.2.19. The sensitivity studies on tripped rod worth insertion times are given for 3.05 seconds to 85 percent insertion and 3.3 seconds insertion. The matchup between the baseline case and the SAR curves is quite good; there is close agreement on the times to peak pressure--Figure 7.2.18--and to minimum DNBR--Figure 7.2.19--and the tripped rod worth insertion times show the expected sensitivities.

Both the Catawba and McGuire baseline comparison calculations show good agreement with the SAR results and are consistent with the differing assumptions applied in the SAR analyses. The DNBR algorithm used for RELAP5 produces similar DNB behavior to that reported for those plants. Quantitative agreement would require DNB calculation using a detailed thermal hydraulic analysis with the same thermal design procedure and CHF correlation.

B&W Application Model

Figures 7.2.20 through 7.2.24 show RELAP5 analysis results for the B&W application model. The differences between the RELAP5 comparison runs and these calculations are:

1. The B&W model has a five region fuel pin, a 1 region gap and a 2 region clad model.

2. The moderator temperature coefficient is +7 pcm/F and a fuel temperature Doppler reactivity feedback is used.

3. A 70% steam dump is used.

Figure 7.2.24 shows the DNBR to be always greater than the limit value. Since DNB does not occur, the ability of the primary coolant to remove heat from the fuel rod is not greatly reduced. Thus, the average fuel and clad temperatures do not increase significantly above their respective initial values.

The reactor coolant pumps will continue to coast down, and natural circulation flow will eventually be established. With the reactor tripped, a stable plant condition would be attained and normal plant shutdown may then proceed.

The analysis performed has demonstrated that for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the limit value at any time during the transient. Thus, no fuel or clad damage is predicted, and all applicable acceptance criteria are met.

7.2.4. Conclusions

The RELAP5 model has shown agreement with SAR complete loss of forced reactor coolant flow analysis results for several different plants and is therefore valid for the analysis of any specific plant when the appropriate plant geometry and plant parameters are modeled.

FIGURE 7.2.1. FOUR PUMP COASTDOWN

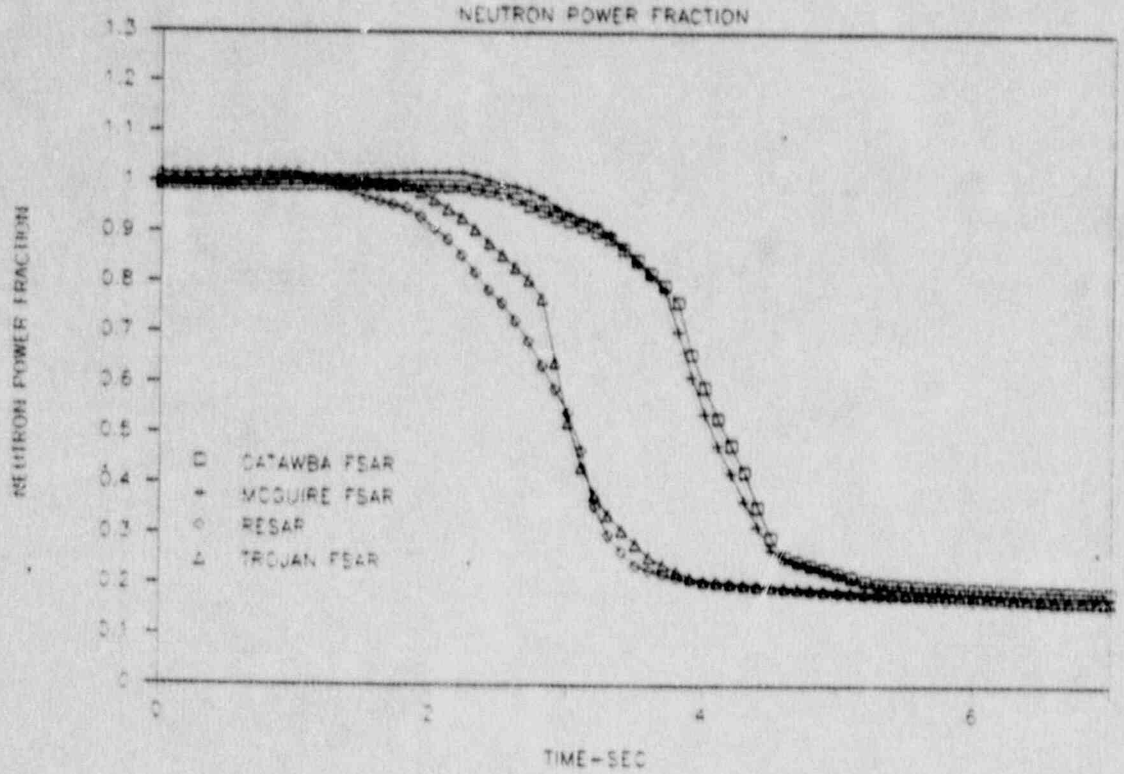


FIGURE 7.2.2. FOUR PUMP COASTDOWN

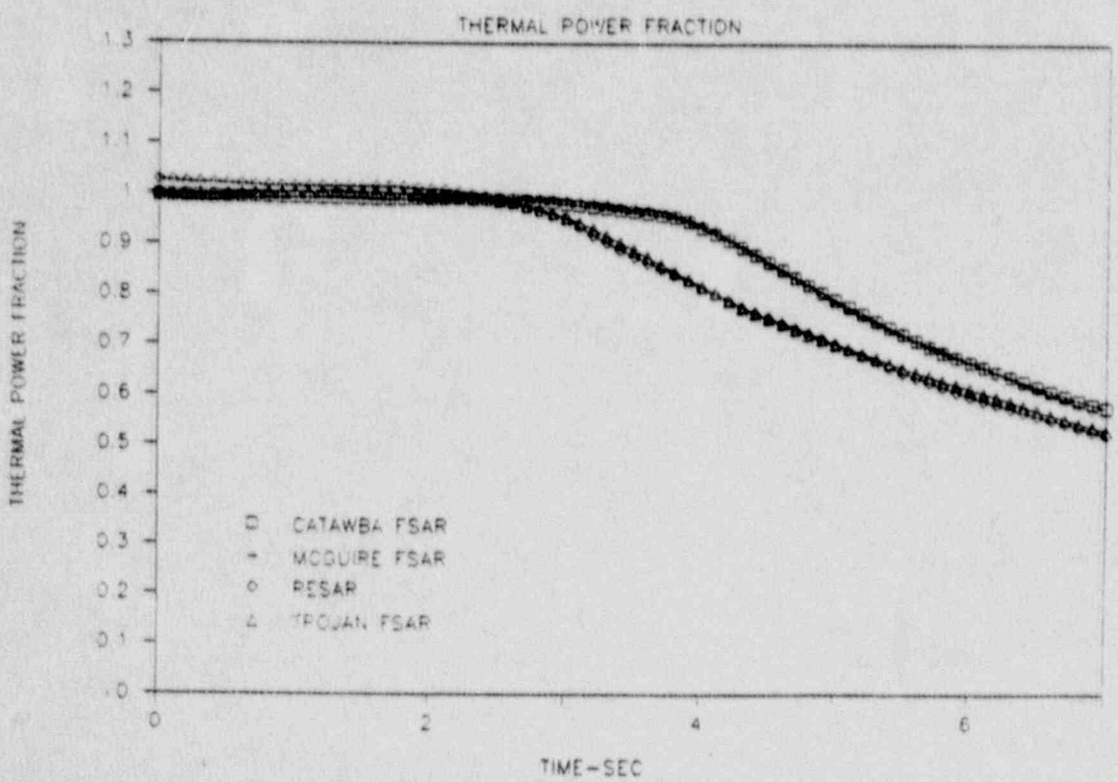


FIGURE 7.2.3. FOUR PUMP COASTDOWN

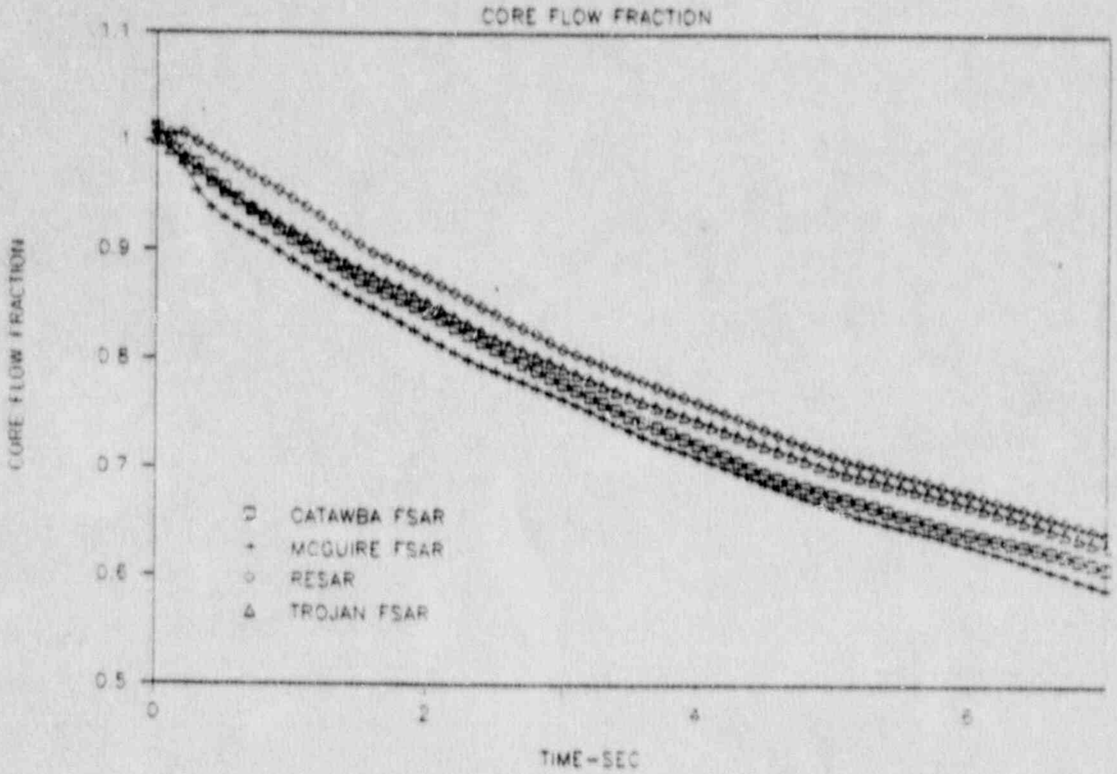


FIGURE 7.2.4. FOUR PUMP COASTDOWN

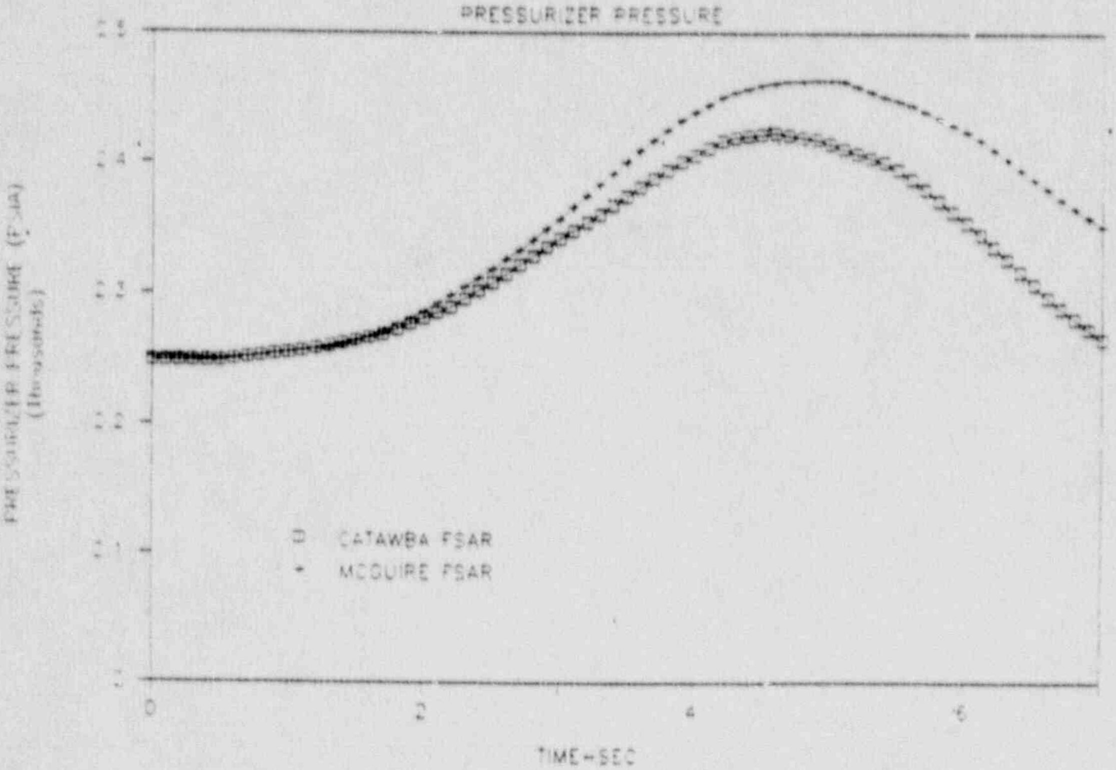


FIGURE 7.2.5. FOUR PUMP COASTDOWN

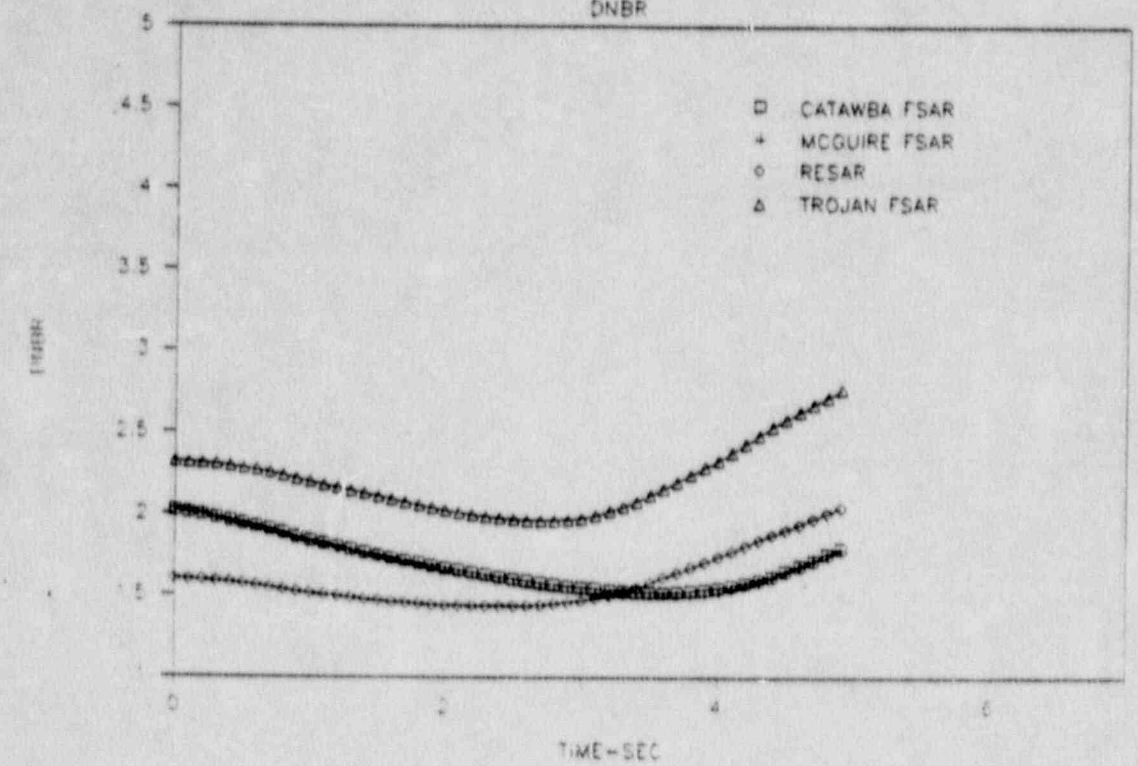


FIGURE 7.2.6. FOUR PUMP COASTDOWN

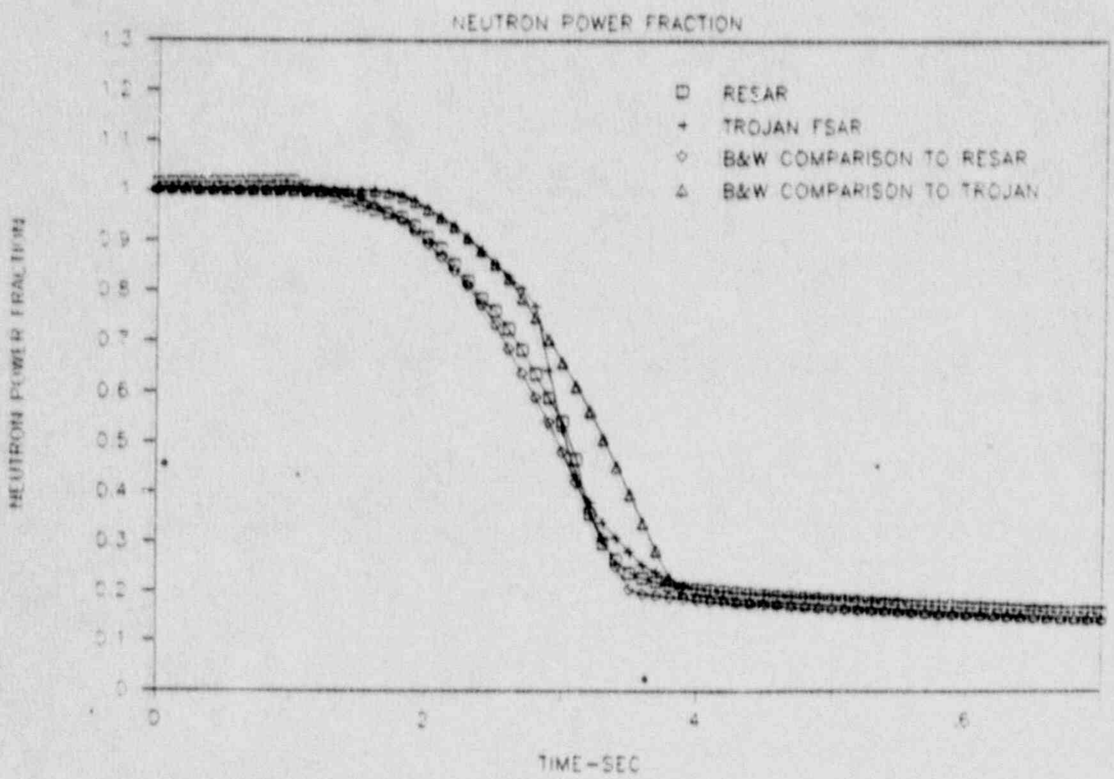


FIGURE 7.2.7. FOUR PUMP COASTDOWN

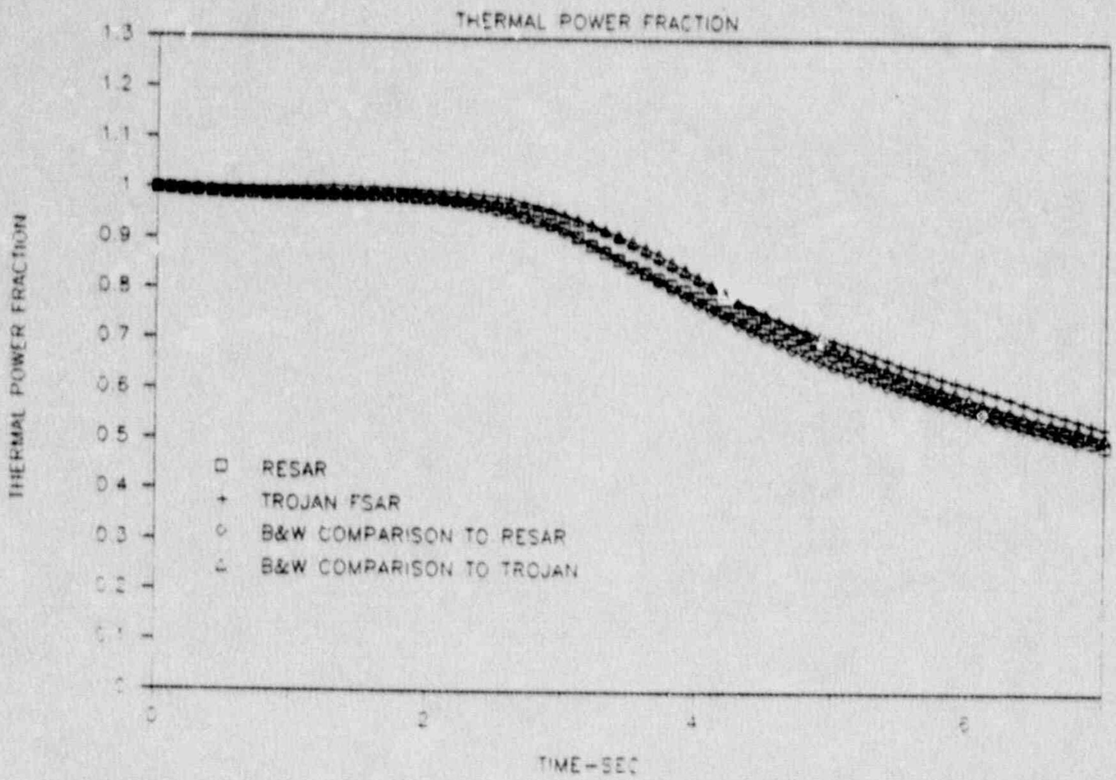


FIGURE 7.2.8. FOUR PUMP COASTDOWN

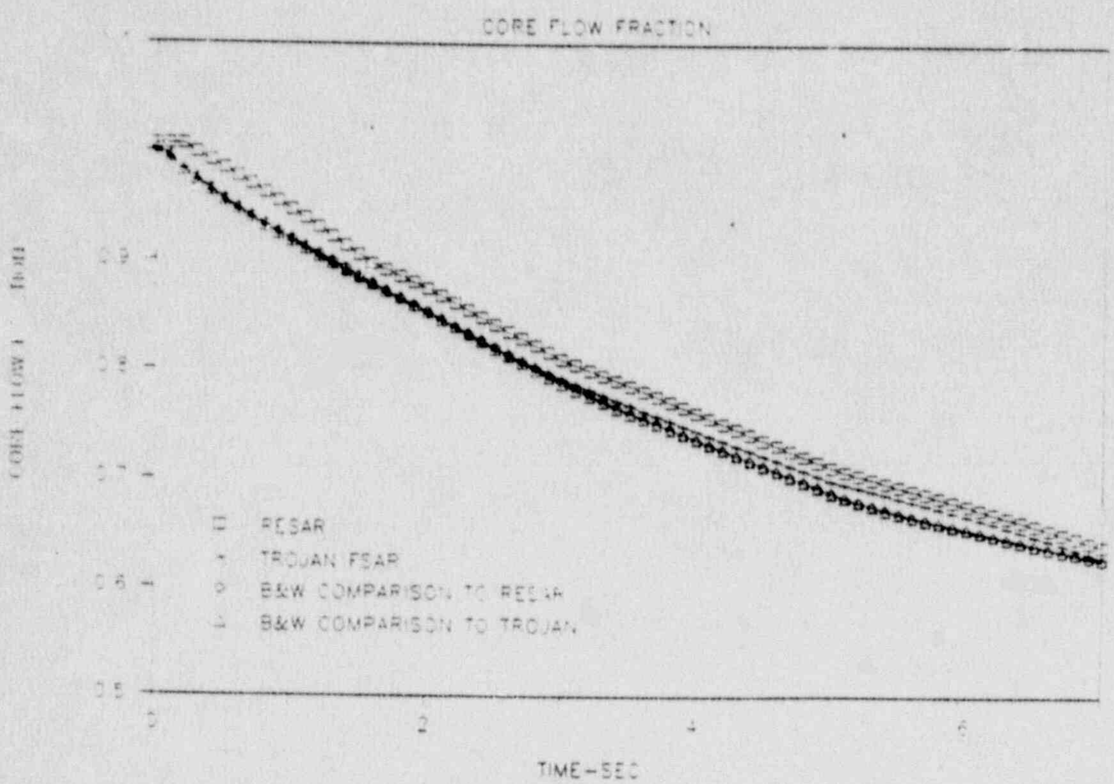


FIGURE 7.2.9. FOUR PUMP COASTDOWN

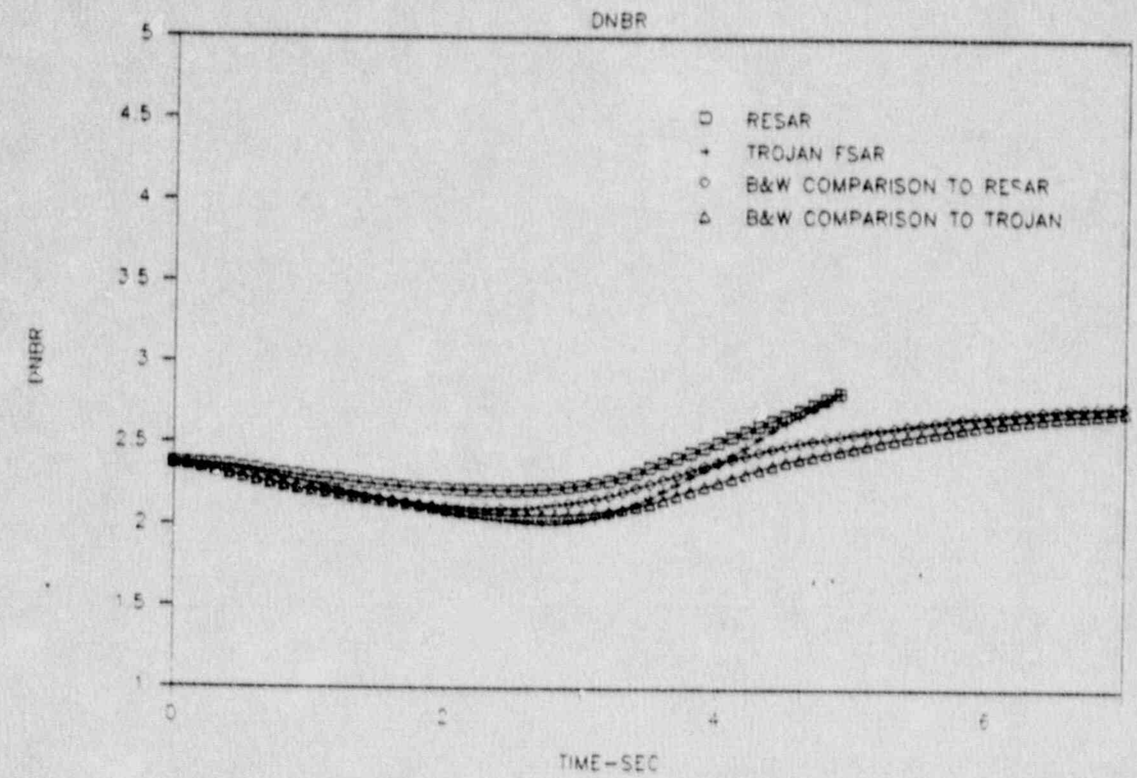


FIGURE 7.2.10. FOUR PUMP COASTDOWN

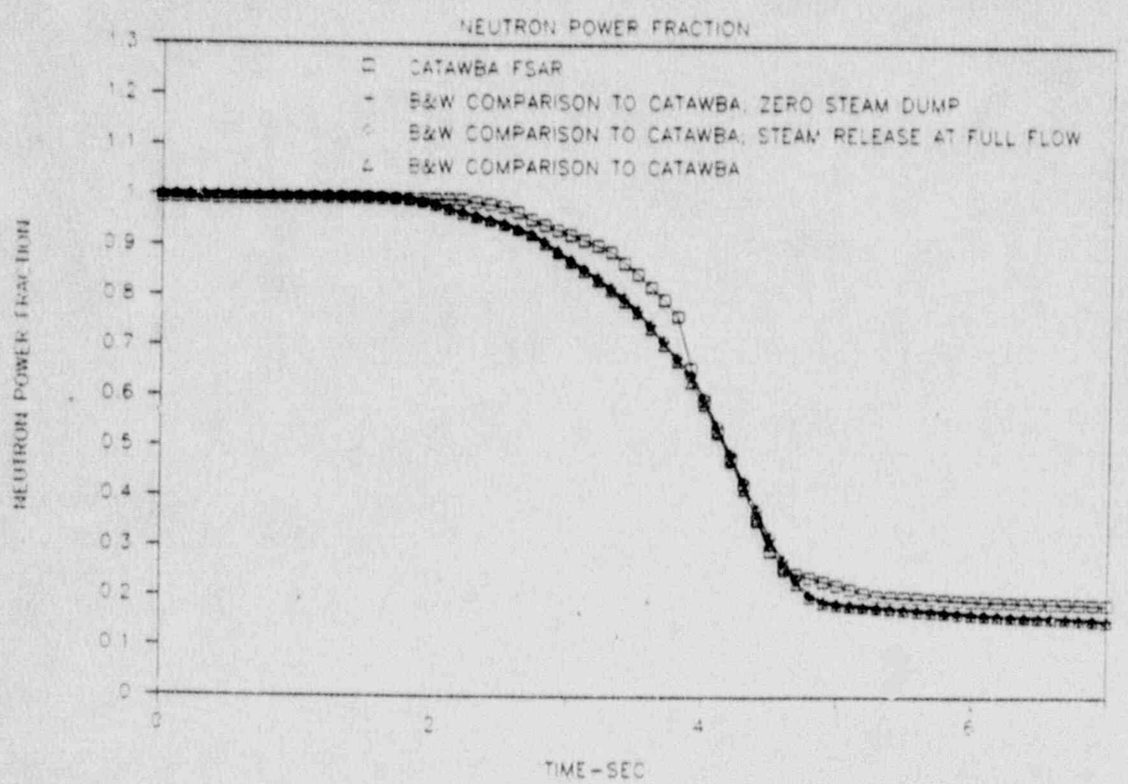


FIGURE 7.2.11. FOUR PUMP COASTDOWN

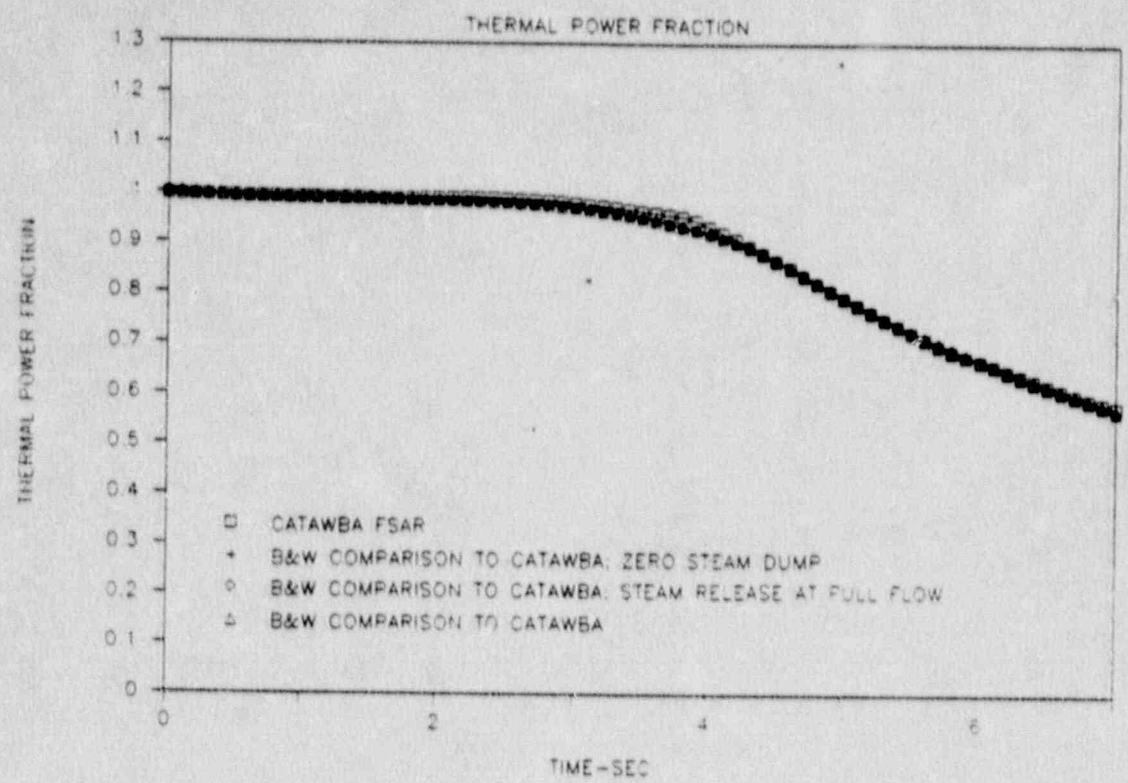


FIGURE 7.2.12. FOUR PUMP COASTDOWN

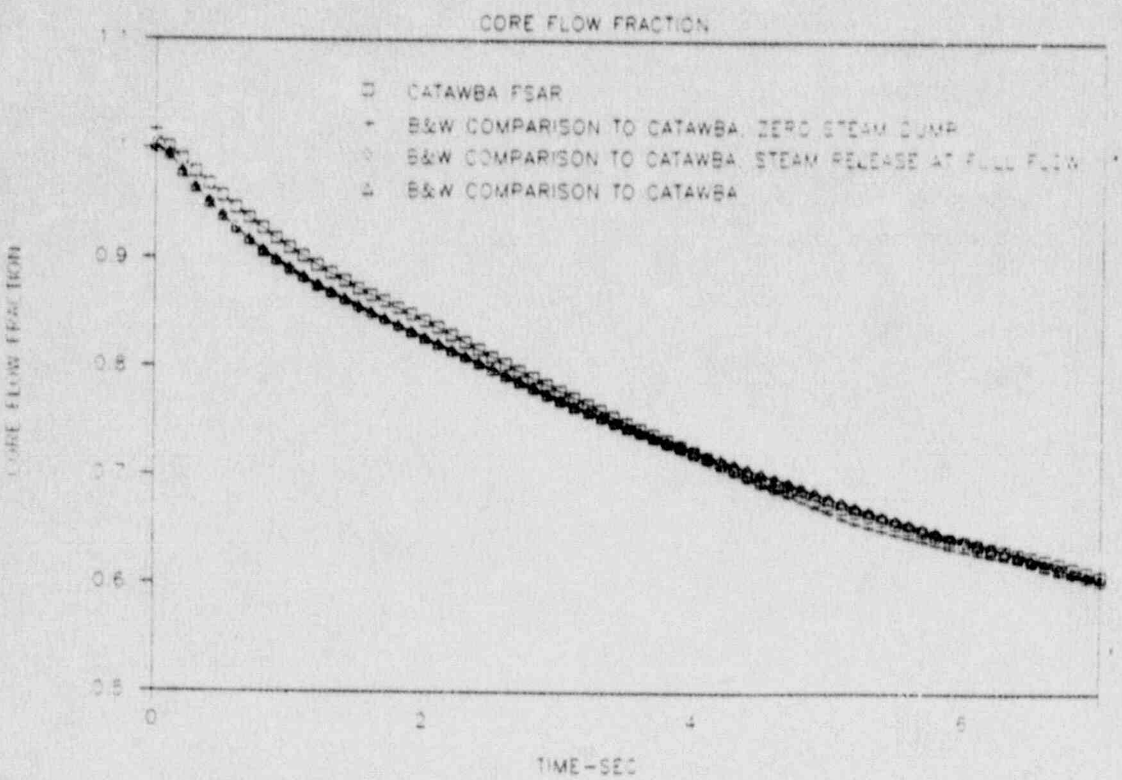


FIGURE 7.2.13. FOUR PUMP COASTDOWN

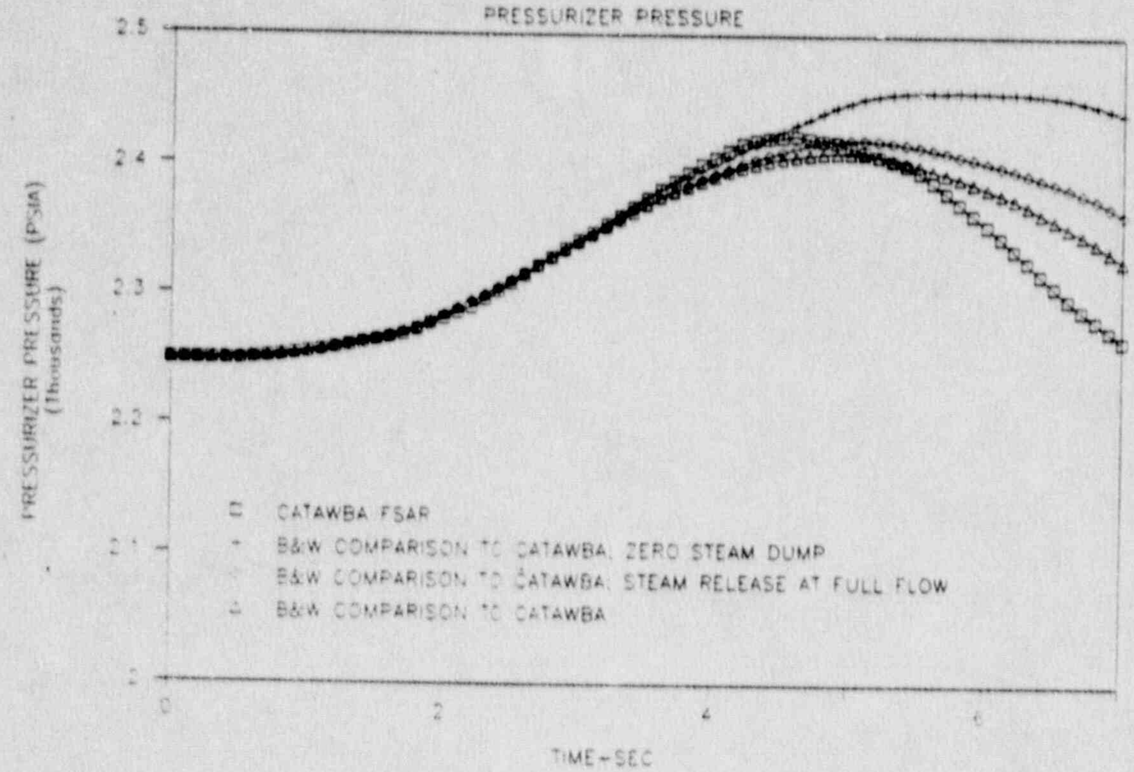


FIGURE 7.2.14. FOUR PUMP COASTDOWN

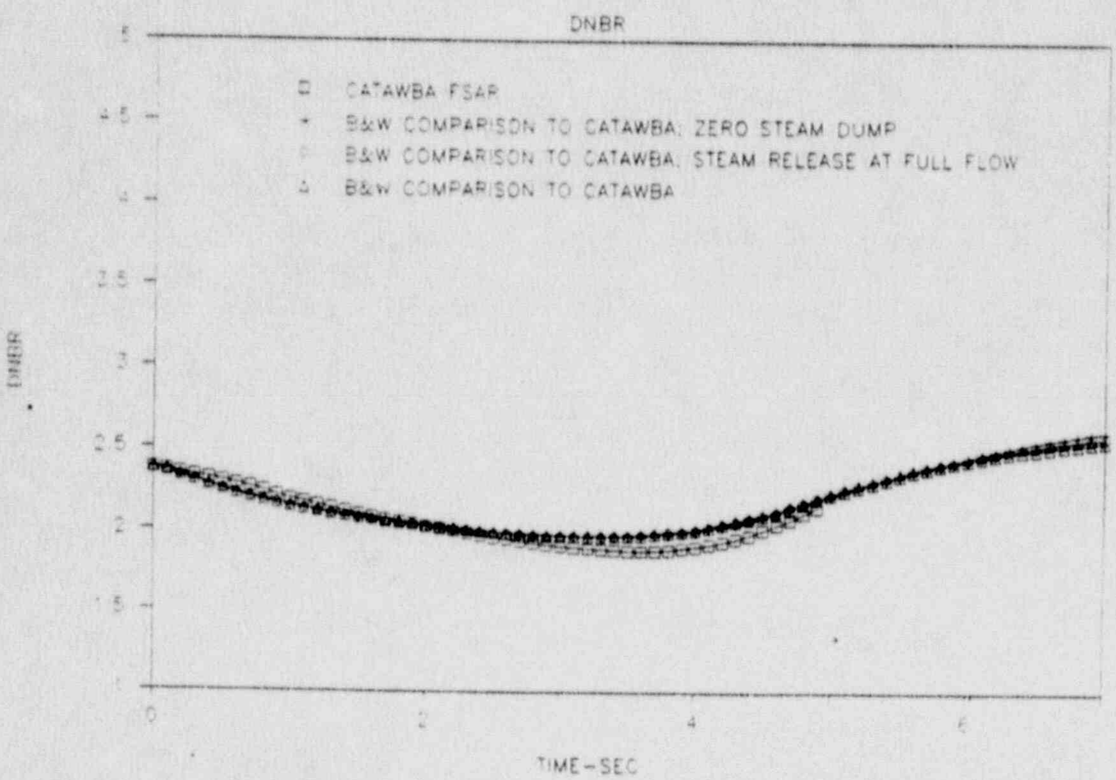


FIGURE 7.2.15. FOUR PUMP COASTDOWN

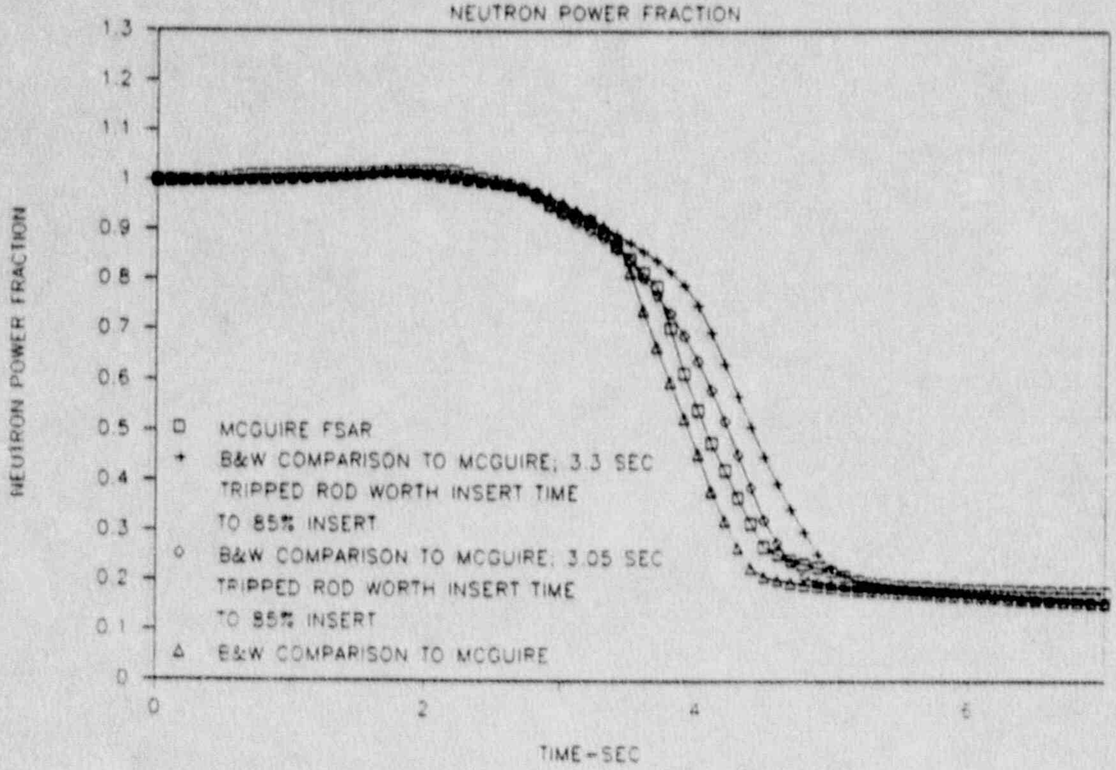


FIGURE 7.2.16. FOUR PUMP COASTDOWN

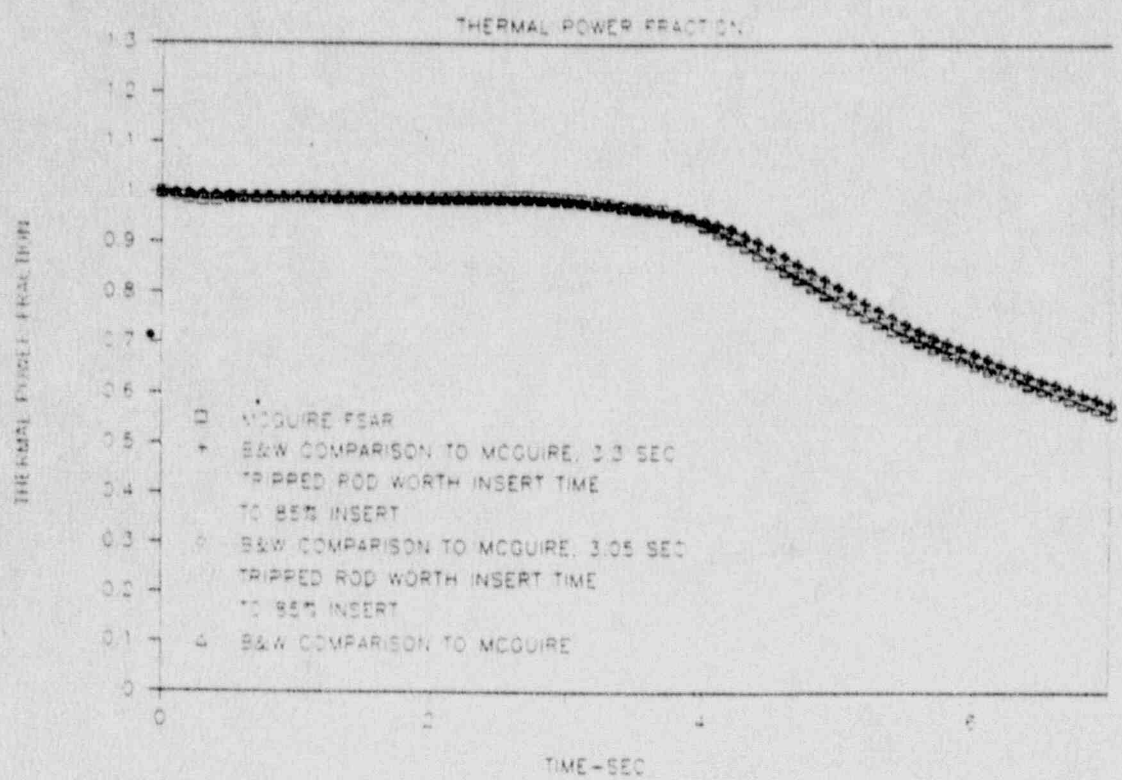


FIGURE 7.2.17. FOUR PUMP COASTDOWN

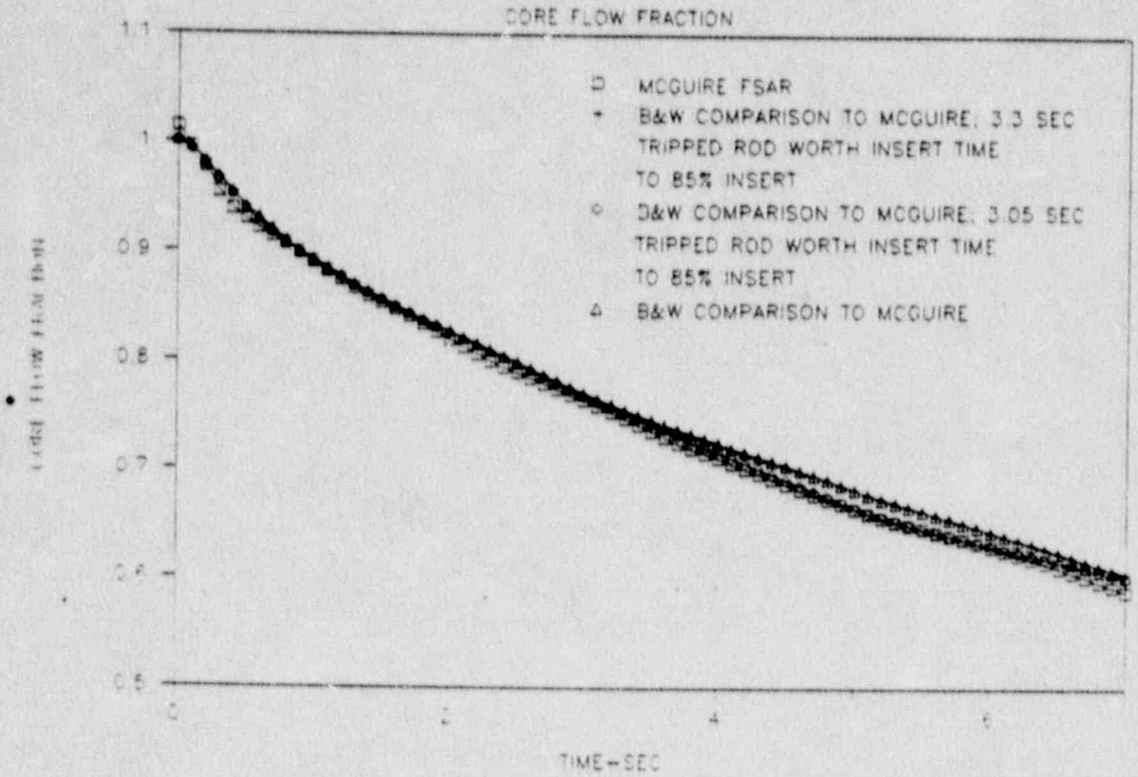


FIGURE 7.2.18. FOUR PUMP COASTDOWN

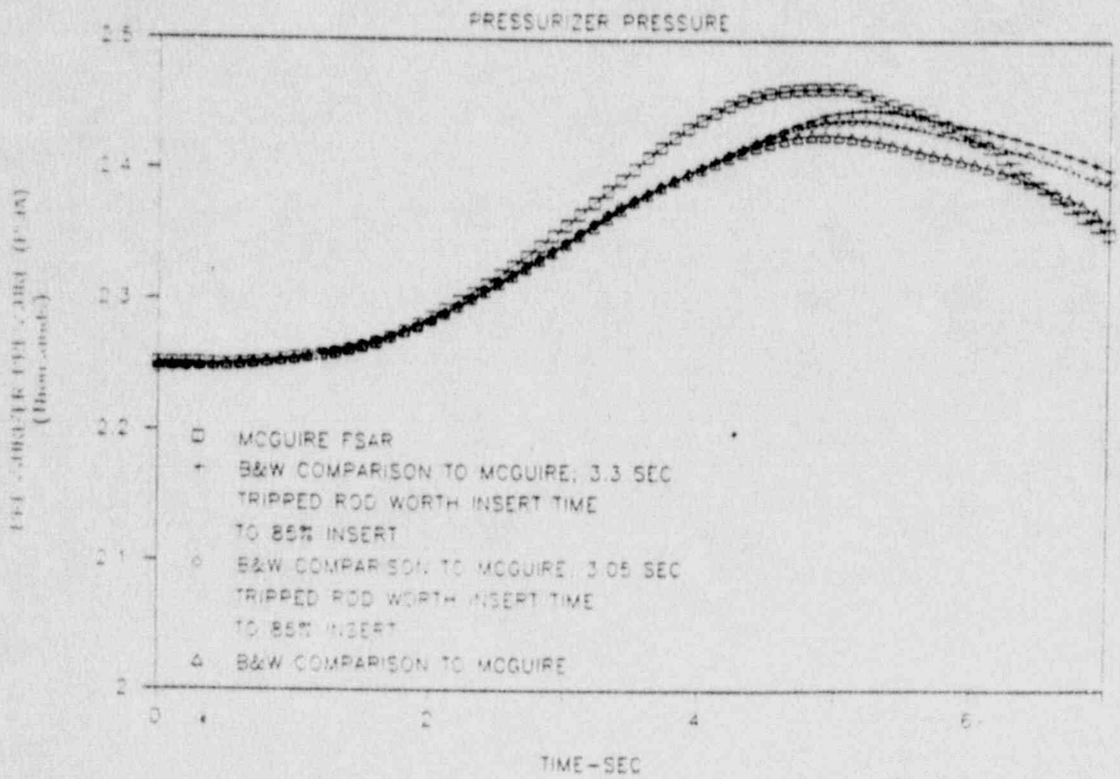


FIGURE 7.2.19. FOUR PUMP COASTDOWN

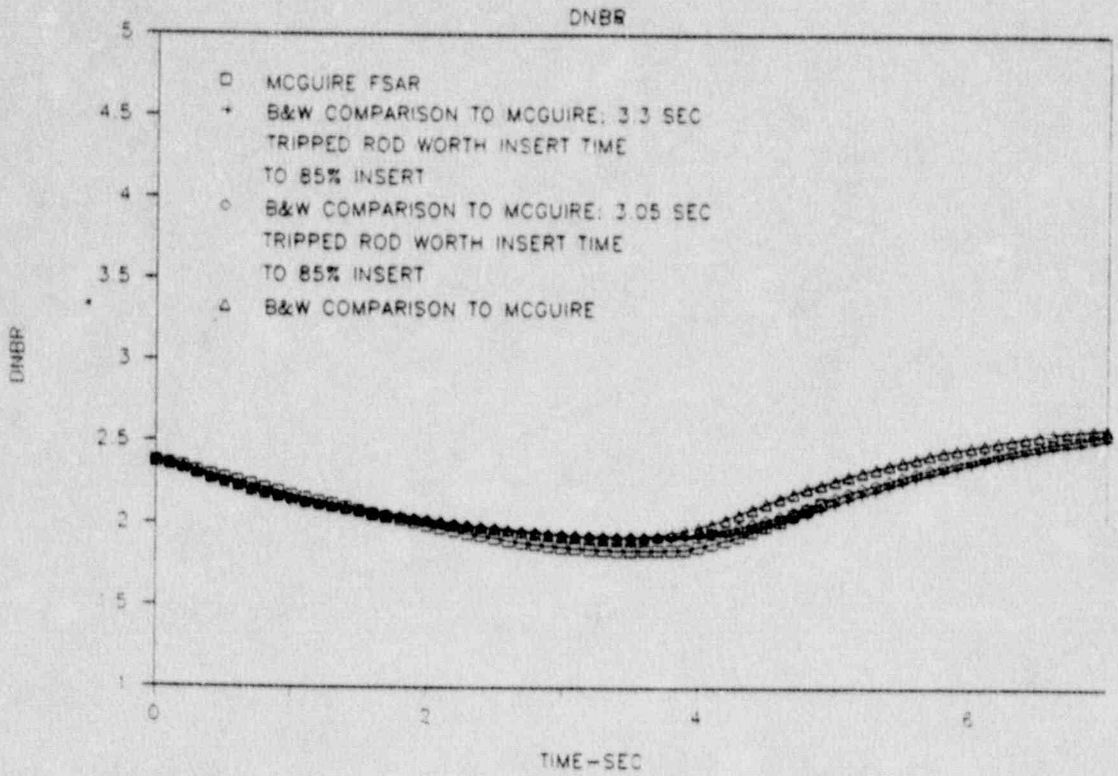


FIGURE 7.2.20. FOUR PUMP COASTDOWN

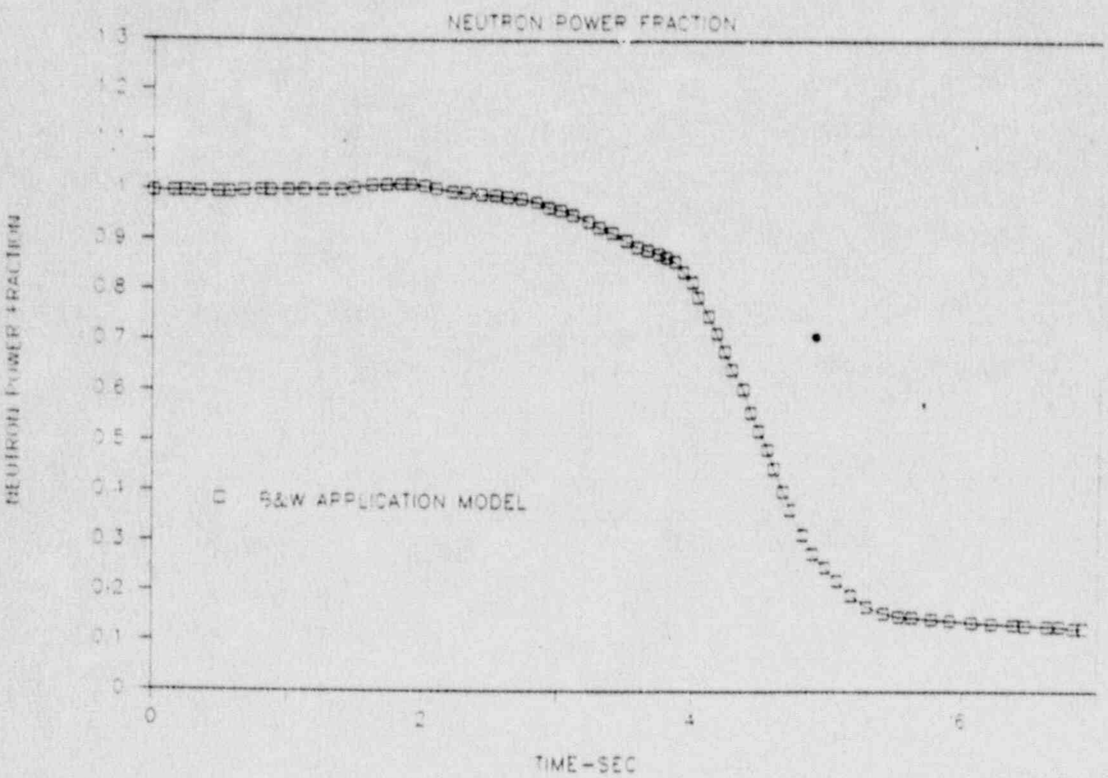


FIGURE 7.2.21. FOUR PUMP COASTDOWN

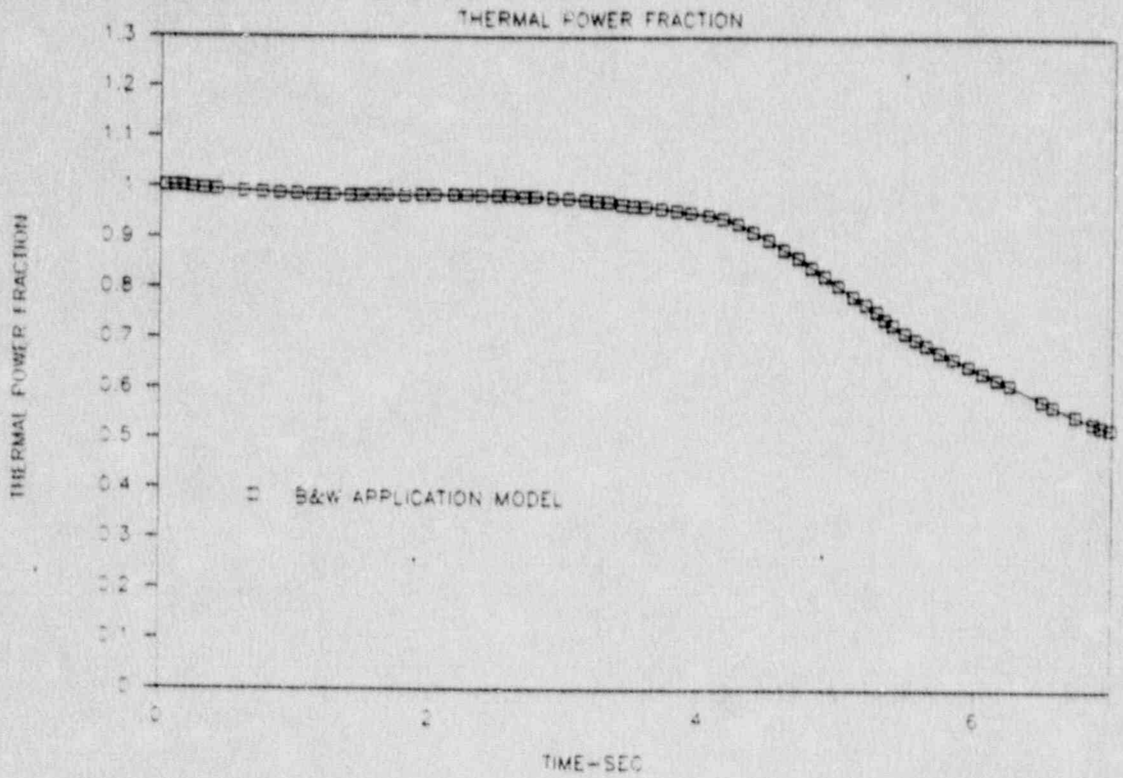


FIGURE 7.2.22. FOUR PUMP COASTDOWN

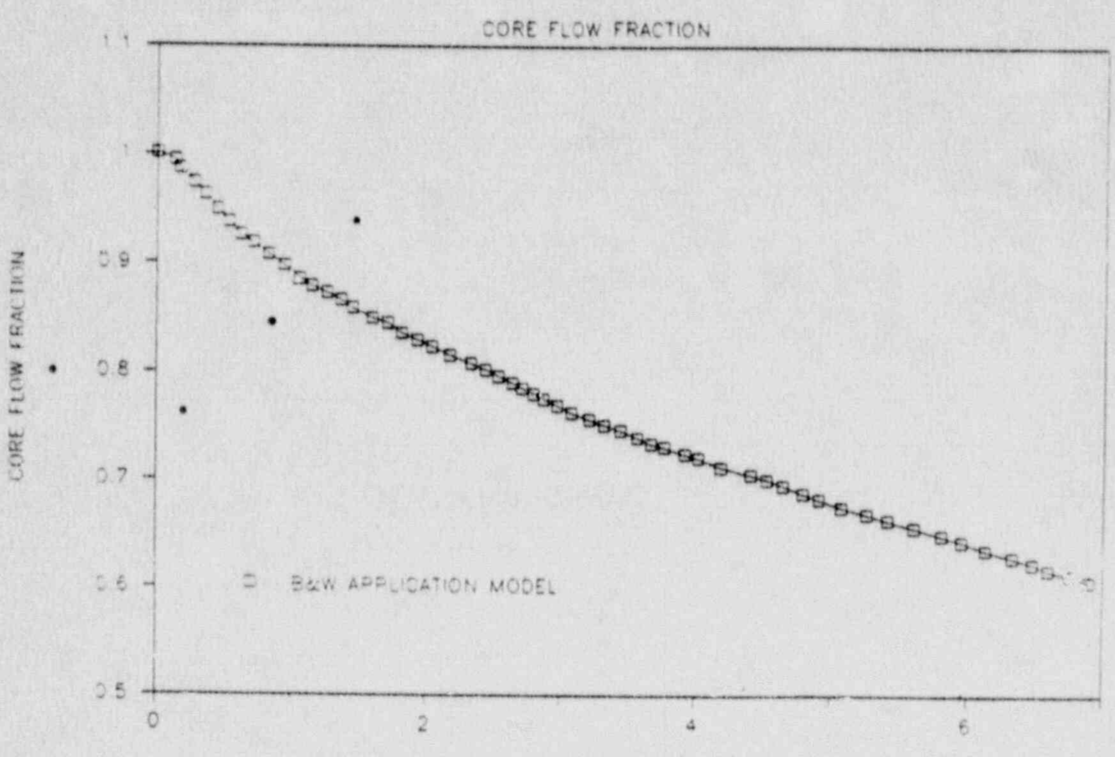


FIGURE 7.2.23. FOUR PUMP COASTDOWN

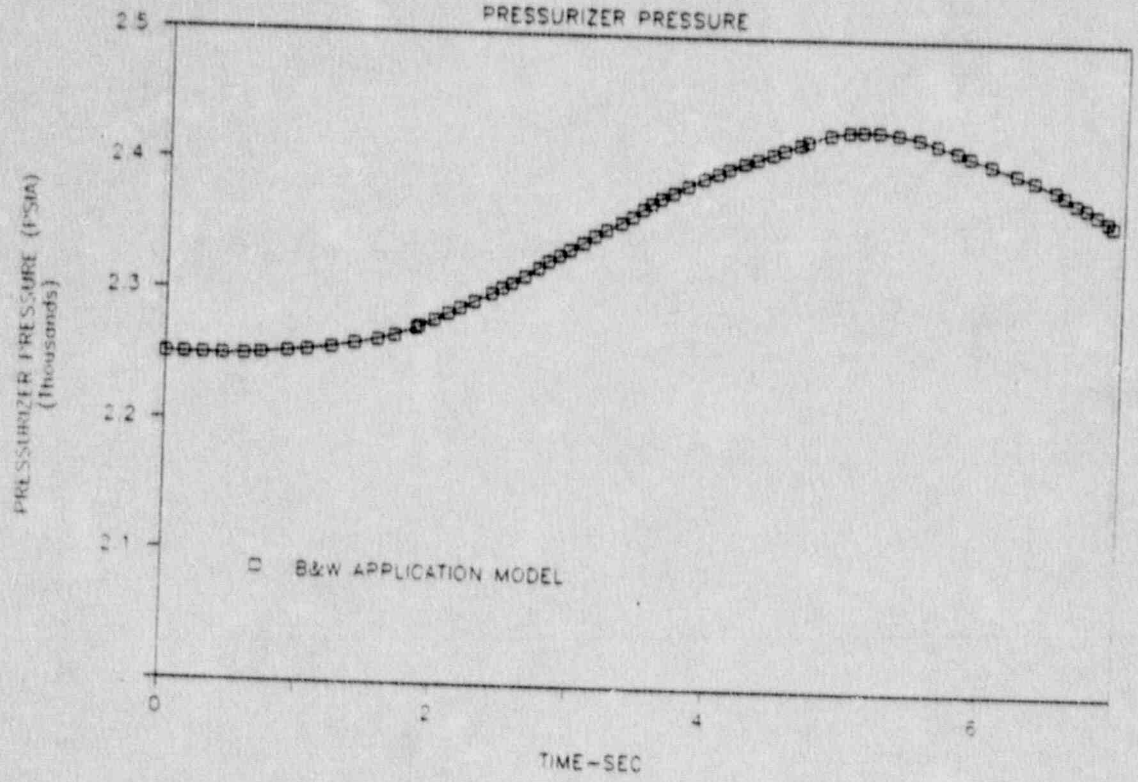
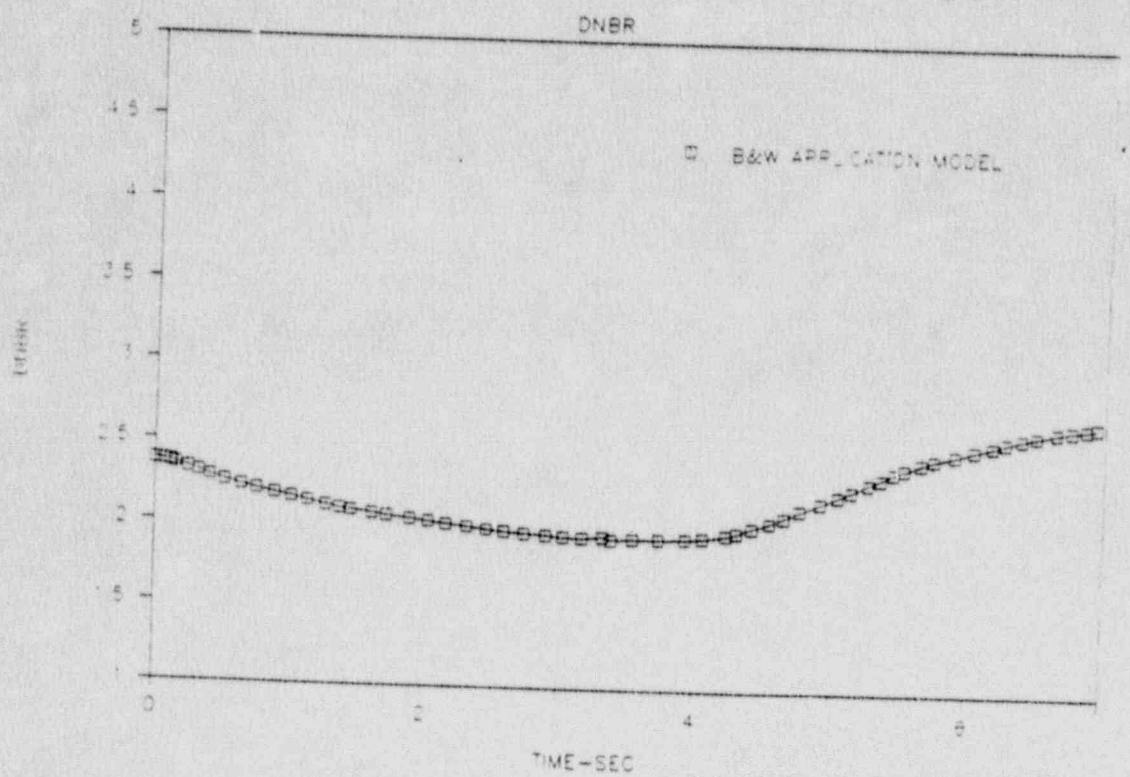


FIGURE 7.2.24. FOUR PUMP COASTDOWN



7.3. Turbine Trip

7.3.1. Identification of Causes and Accident Description

The turbine trip event is the limiting transient in the grouping of events resulting in a decrease in heat removal by the secondary system caused by a loss of external load. For a turbine trip, the turbine stop valves close rapidly (typically 0.1 sec.) on loss of trip-fluid pressure actuated by one of a number of possible turbine trip signals. (Turbine-trip initiation signals include generator trip, low condenser vacuum, loss of lubricating oil, turbine thrust bearing failure, turbine overspeed, main steam reheat high level, and manual trip.) Steam flow to the turbine stops abruptly when the stop valves close, and sensors on the stop valves detect the turbine trip and initiate steam dump. The loss of steam flow results in an almost immediate rise in secondary system temperature and pressure. It also reduces heat removal from the primary side, causing a pressurization transient. A more severe transient occurs for the turbine trip event than for the loss of external load event due to the more rapid loss of steam flow caused by the more rapid valve closure.

A turbine trip is classified as an ANS Condition II event, fault of moderate frequency. See Section 4.0 for a discussion of Condition II events.

The automatic steam dump system would normally accommodate the excess steam generation. Reactor coolant temperatures and pressure do not significantly increase if both the steam dump system and pressurizer pressure control system are functioning properly.

7.3.2. Analysis of Effects and Consequences

Method of Analysis

In this analysis, the behavior of the unit is evaluated for a complete loss of steam load from 100 percent power to show the adequacy of the pressure relieving devices and to demonstrate core protection margins. The reactor is not tripped until conditions in the RCS result in a trip. Main feedwater flow is terminated at the time of turbine trip, with no credit taken for auxiliary feedwater to mitigate the consequences of the transient.

The turbine trip transient was analyzed in the SARs by employing the digital computer program LOFTRAN (Reference 7). The program simulates the neutron kinetics, RCS pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generator, and steam generator safety valves. The program computes pertinent plant variables including temperatures, pressures, and power level.

The RELAP5/MOD2 computer code (Reference 6) was used for the B&W analysis. The RELAP5/MOD2 code is a best-estimate transient simulation of pressurized water reactors and associated systems. The code simulates the neutron kinetics, RCS pressurizer, pressurizer relief and safety valves, steam generators, and steam generator safety valves. Through the use of control variables any plant parameter can be computed.

Initial and Boundary Conditions

Initial reactor power, pressure and RCS temperatures are assumed to be at their nominal values. Uncertainties in initial conditions are included in the limiting DNBR analysis as described in Section 7.0. From the standpoint of the maximum pressures attained it is conservative to assume that the reactor is in manual control. If the reactor were in automatic control, the control rod banks would move prior to trip and reduce the severity of the transient.

The steam generator pressure rises with stop valve closure to the safety valve setpoint where steam release through safety valves limits secondary steam pressure at the setpoint value. Main feedwater flow to the steam generators is assumed to be lost at the time of turbine trip. No credit is taken for auxiliary feedwater flow since a stabilized plant condition will be reached before auxiliary feedwater initiation is normally assumed to occur; however, the auxiliary feedwater pumps would be expected to start on a trip of the main feedwater pumps. The auxiliary feedwater flow would remove core decay heat following plant stabilization. Also, no credit is taken for the effect of pressurizer spray and power operated relief valves in reducing or limiting the coolant pressure. Safety valves are operable.

Reactivity Coefficients

The turbine trip is analyzed with minimum reactivity feedback. The minimum feedback cases assume a least negative moderator temperature coefficient and the least negative Doppler coefficients. For the B&W model analysis a Doppler fuel temperature coefficient consistent with the time in life for this analysis was used.

A moderator coefficient of zero was used for the Trojan, and Catawba analyses. A positive moderator temperature coefficient of +5 pcm/F was used for the McGuire analysis. A positive moderator temperature coefficient of +7.0 pcm/F was used for the B&W model analysis.

Reactor Trips

Reactor trip is actuated by the first Reactor Protection System trip setpoint reached. Trip signals are expected due to high pressurizer pressure, overtemperature ΔT, high pressurizer water level, and low-low steam generator water level. No credit is assumed for a reactor trip due to a turbine trip for this analysis.

The reactor protection system may be required to function following a turbine trip. Pressurizer safety valves and/or steam generator safety valves may be required to open to maintain system pressures below allowable limits. No single active failure will prevent operation of any system required to function.

7.3.3. Results

The significant system responses to the turbine trip event are directly caused by the rise in secondary pressure and temperature following closure of the turbine stop valves. The reduced steam generator heat demand results in heatup and expansion of primary coolant; primary pressure rises through the high pressure trip setpoint to actuate the pressurizer safety valves. The relief flow through the safety valves and the reactor trip act to limit the increase in primary system pressure.

Safety Analysis Report Data

Plots of the key turbine trip parameters, digitized from the Trojan, Catawba, and McGuire SARs and from the Westinghouse RESAR, are shown in Figure 7.3.1 through 7.3.6. For all of these, data were presented for neutron power, pressurizer pressure, core average temperature, pressurizer water volume, and DNB ratio. Core inlet temperatures were reported for all but the Trojan SAR.

Examination of the RESAR curves, specifically the neutron power and primary pressure concluded that trip setpoints or rod motion delays were assumed in the original analysis that are incompletely described in the available documents. The almost two-second difference between the Trojan and RESAR power curves could not be accounted for in the as-documented analysis assumptions for the two cases. For this reason, the turbine trip analyses in RELAP5 were confined to the Trojan, McGuire, and Catawba cases. The Trojan SAR data are for both the original and a later SAR revision (Reference 4).

A point of interest is suggested by the neutron power fraction curves, comparing the response for Trojan to that calculated for the Catawba and McGuire units. Despite a difference in rod insertion times -- 2.2 seconds for Trojan, 3.05 seconds for Catawba, and 3.3 seconds for McGuire -- the neutron power response for Trojan is almost coincident with those of Catawba and McGuire. This is attributable to the difference in pressure rise response; the Catawba and McGuire analyses reach the reactor trip setpoint at about 4 seconds whereas the Trojan results indicate 5.5 seconds to reactor trip pressure. Thus, the differences in rod insertion times are offset by the more rapid occurrence of the high pressure trip for the Catawba and McGuire SAR cases. This point will be discussed further with the RELAP5 comparison analyses for Catawba and McGuire.

A comparison of the McGuire and Catawba SAR results indicates more rapid primary pressurization for the McGuire turbine trip analysis. Hence, the slightly earlier reactor trip for McGuire and tendency to lead or exceed the Catawba curves for the other parameters. These marginal differences--they are not material to the major results--can be attributed to different SAR analysis inputs in terms of pressurizer liquid volume, loop flow, and loop temperatures as initial conditions. A single RELAP5 initial condition was used, closer to the Catawba conditions, to show the effect of the difference in rod insertion times and moderator coefficients.

Trojan Comparison

Results of the RELAP5 comparative analysis for Trojan are shown in Figures 3.3.7 through 7.3.11. Two RELAP5 cases were considered and are identified in the plots: for the first case, the effect of steam dump during the event was not modeled, and for the second, the steam dump system was included in the analysis. For all of the key parameters, most significantly the primary pressure increase and reactor trip timing, the RELAP5 outputs match up well with the SAR results.

The neutron power fraction and pressurizer curves of Figures 7.3.7 and 7.3.8 are most indicative. Replication of the SAR assumptions and boundary conditions in the RELAP5 model produces virtually the same primary pressure response and time-to-trip as were generated for the SAR. The RELAP5 results are slightly ahead of the SAR curves, but not significantly. There is a noticeable difference at and beyond the time at which steam relief through the pressurizer safety valves begins. This can be attributed to a difference in the treatment of the safety valve setpoint pressure error in the analyses or in the modeling of the valve flow rates. The RELAP5 safety valve flow was fixed at rated capacity using a time-dependent junction for this calculation.

As in other of the comparisons, the core average temperature as taken from the RELAP5 calculation leads the plot presented in the SAR. The close correspondence between the pressure insurge plots, Figure 7.3.10, however, indicates the consistency in the calculated primary fluid expansion due to the heatup. The difference in the presented core temperatures indicates that the plots were derived from different bases in the two sets of code outputs.

Catawba/McGuire Comparisons

The RELAP5 comparison cases for the Catawba/McGuire analyses are plotted in Figures 7.3.12 through 7.3.19 for the major core and system parameters. The RELAP5 results for comparison with Catawba and for comparison with McGuire reflect the differences associated with the tripped rod insertion times and moderator coefficients used in the analyses. As already discussed, both cases were done from the same initial condition. In Figure 7.3.14, the effect of the steam dump system is shown for the Catawba turbine trip. Because the steam dump occurs with delay after reactor trip, only the post-trip depressurization results are affected.

In performing the turbine trip comparison analysis, the RELAP5 model was adjusted to include an additional effect, a loss of offsite power (LOOP) coincident with the turbine trip, in the primary pressure response. The further reduction in steam generator heat demand associated with the LOOP assumption accounts for the earlier, steeper pressure responses and earlier trip time in the McGuire and Catawba SAR analyses than was evidenced for Trojan.

As shown in the first two figures, the RELAP5 results practically coincide with the SAR calculations. The slightly steeper pressurization during the first few seconds is due to the fixed LOOP adjustment made for the RELAP5 cases. The effect of the steam dump system is shown for the Catawba case in Figure 7.3.14. As stated earlier, the effect of the steam dump is not present until well after reactor trip. Peak pressures calculated by RELAP5 are in early exact agreement with those presented in the SARs.

Examination of the other key parameters, the core average temperatures in Figure 7.3.15 and the pressurizer liquid volume and core inlet temperatures in the following two figures, indicates that the RELAP5 calculations produce the same timing and magnitude of the excursions in these variables as did the SAR system analyses. The better matchup is between RELAP5 and Catawba since the RELAP5 model was initialized closer to the Catawba initial conditions. This is best reflected by the timing of events in the core temperature and DNBR plots. Again, the DNBR results were calculated directly from the RELAP5 outputs using an algorithm as opposed to the detailed thermal-hydraulic analysis done for the SAR.

B&W Application Model

Figure 7.3.20 through 7.3.26 show results for the B&W application model. The differences between the comparison RELAP5 runs and the SAR model runs are:

1. The B&W model has a five-region fuel pin, a 1-region gap, and a 2-region clad model.

2. The moderator temperature coefficient is a +7.0 pcm/F modeled as a reactivity versus density feedback, and the fuel temperature Doppler reactivity feedback is used.

3. A 70% steam dump is used but no LOOP effect is included.

The turbine trip accident was evaluated assuming the plant to be initially operating at full power with no credit taken for the pressurizer spray or pressurizer power-operated relief valves. The reactor is tripped on the high pressurizer pressure signal. Figures 7.3.20 through 7.3.26 show the transient with a +7.0 pcm/F positive moderator coefficient. The neutron flux remains essentially constant at full power until the reactor is tripped. The DNBR never goes below the initial value throughout the transient. In this case the pressurizer safety valves are actuated, and maintain system pressure below design value.

7.3.4. Conclusions

The RELAP5 model has shown agreement with SAR turbine trip analysis results for several different plants and is therefore a valid code for the analysis of any specific plant when the appropriate plant geometry and plant parameters are modeled.

FIGURE 7.3.1. TURBINE TRIP

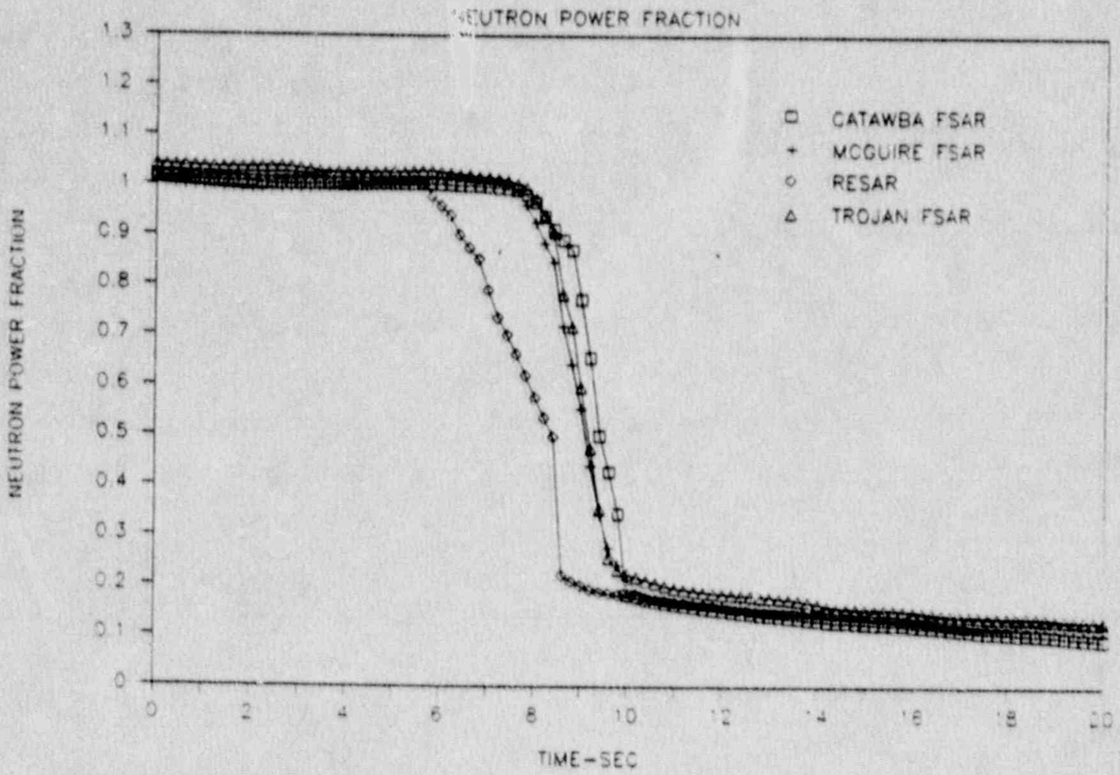


FIGURE 7.3.2. TURBINE TRIP

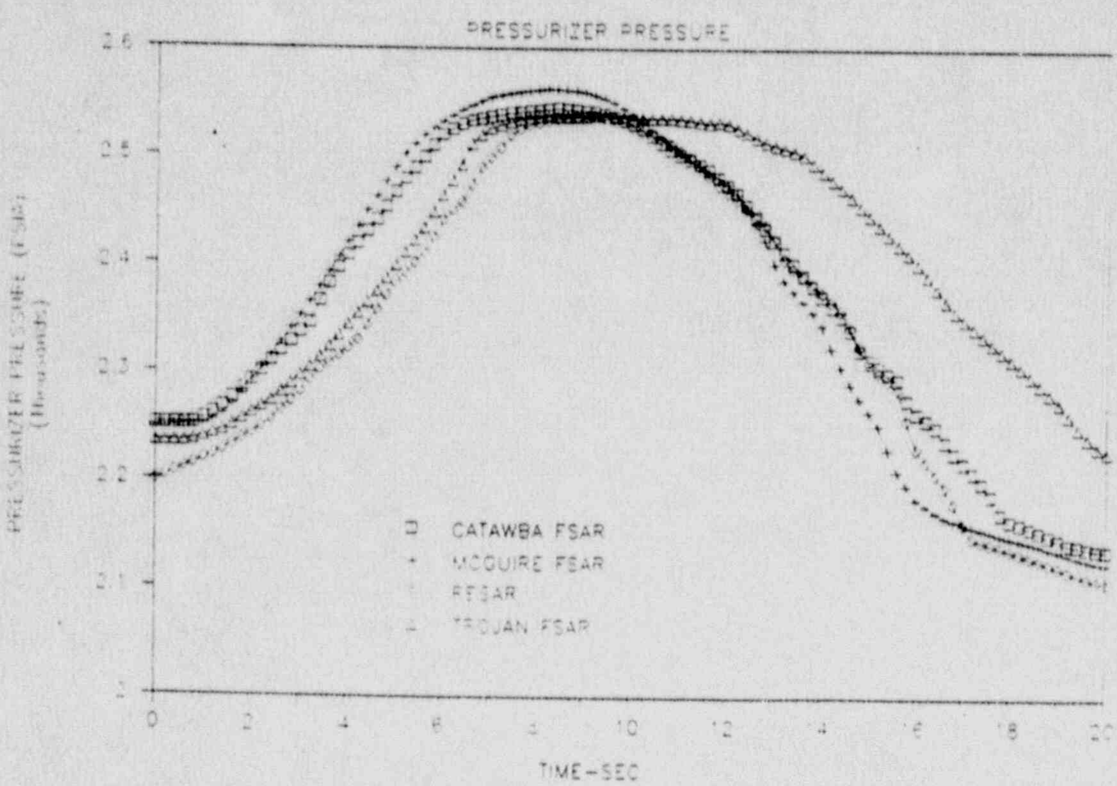


FIGURE 7.3.3. TURBINE TRIP

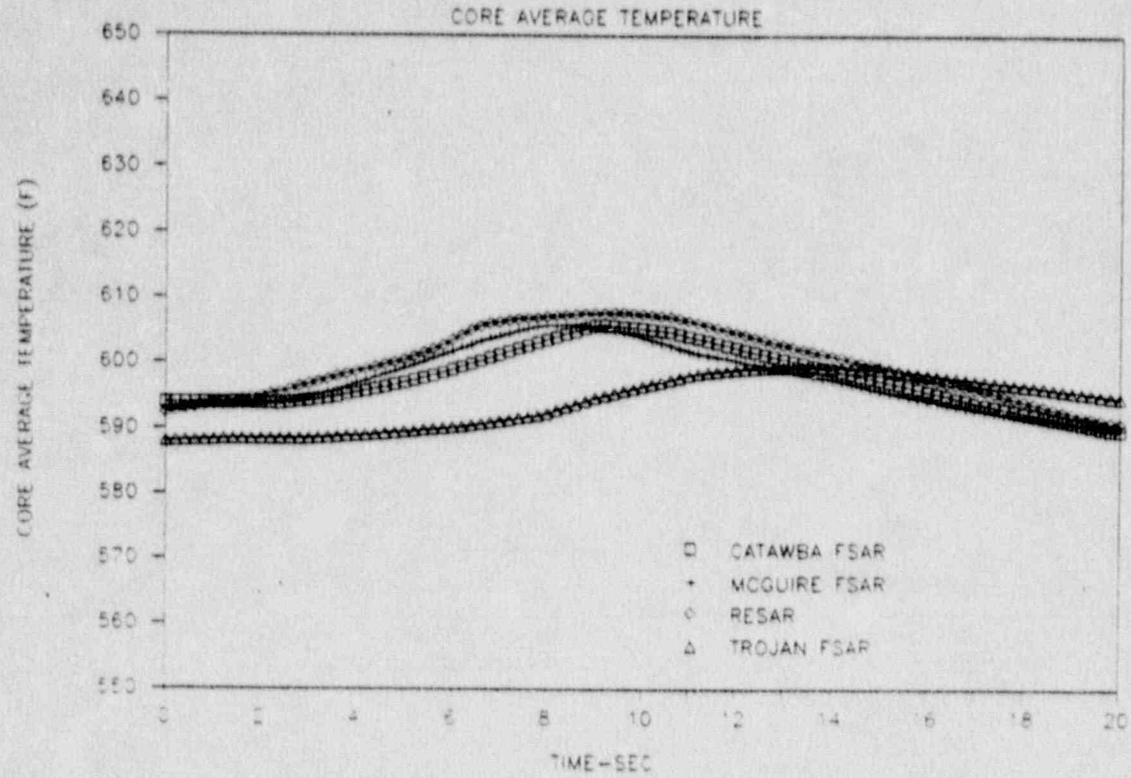


FIGURE 7.3.4. TURBINE TRIP

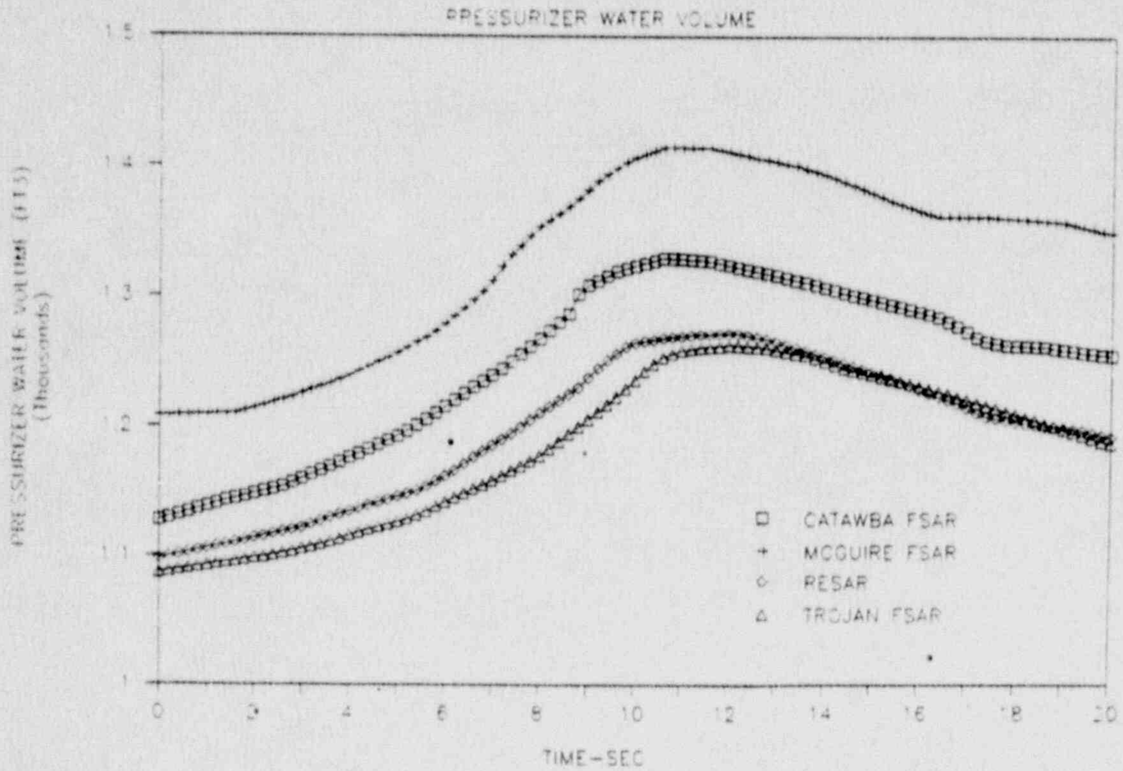


FIGURE 7.3.5. TURBINE TRIP

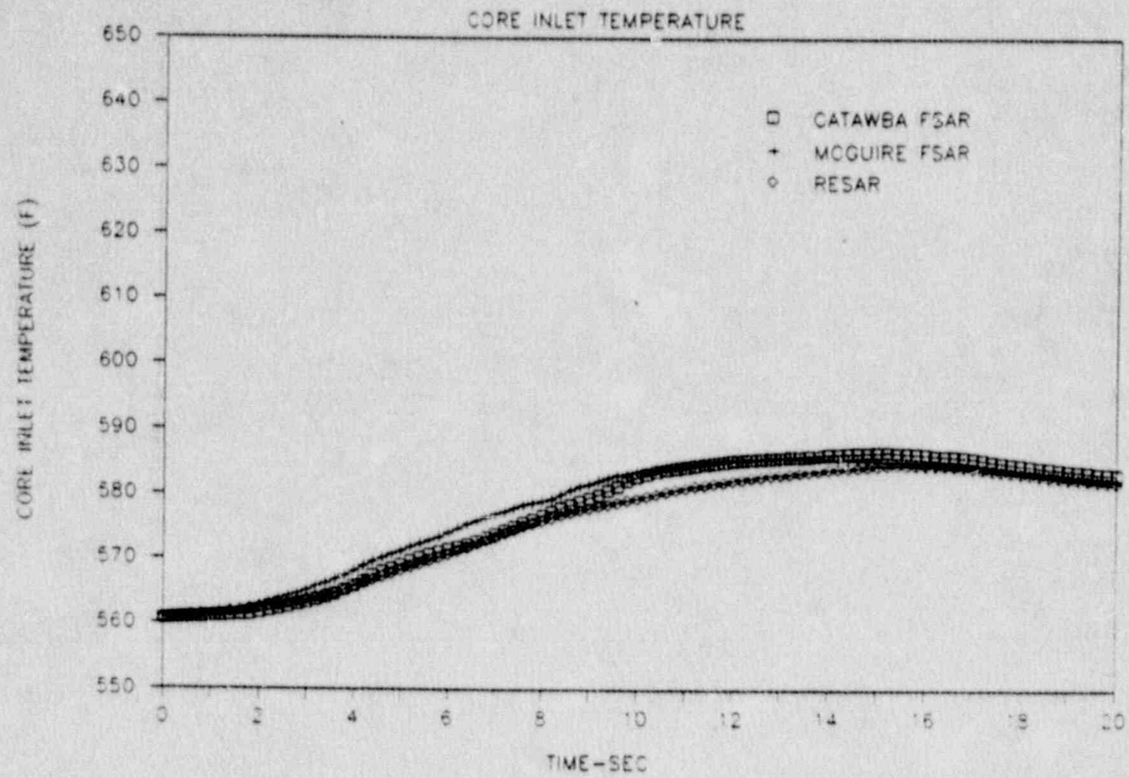


FIGURE 7.3.6. TURBINE TRIP

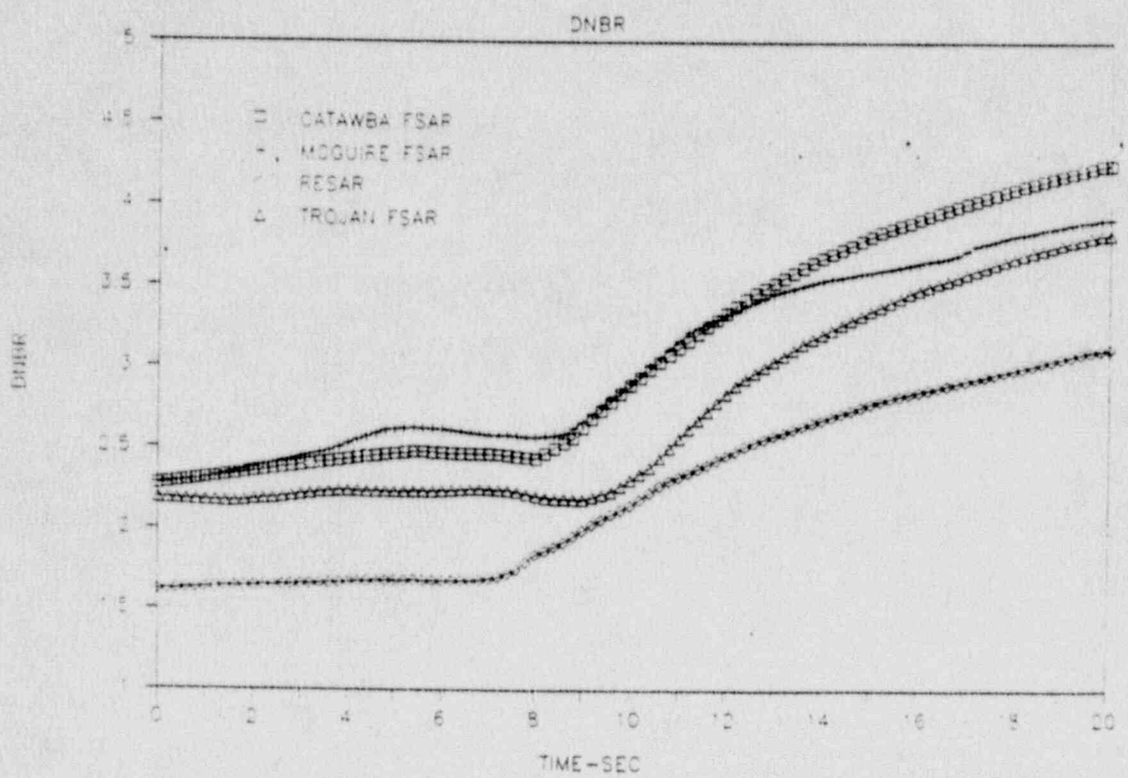


FIGURE 7.3.7. TURBINE TRIP

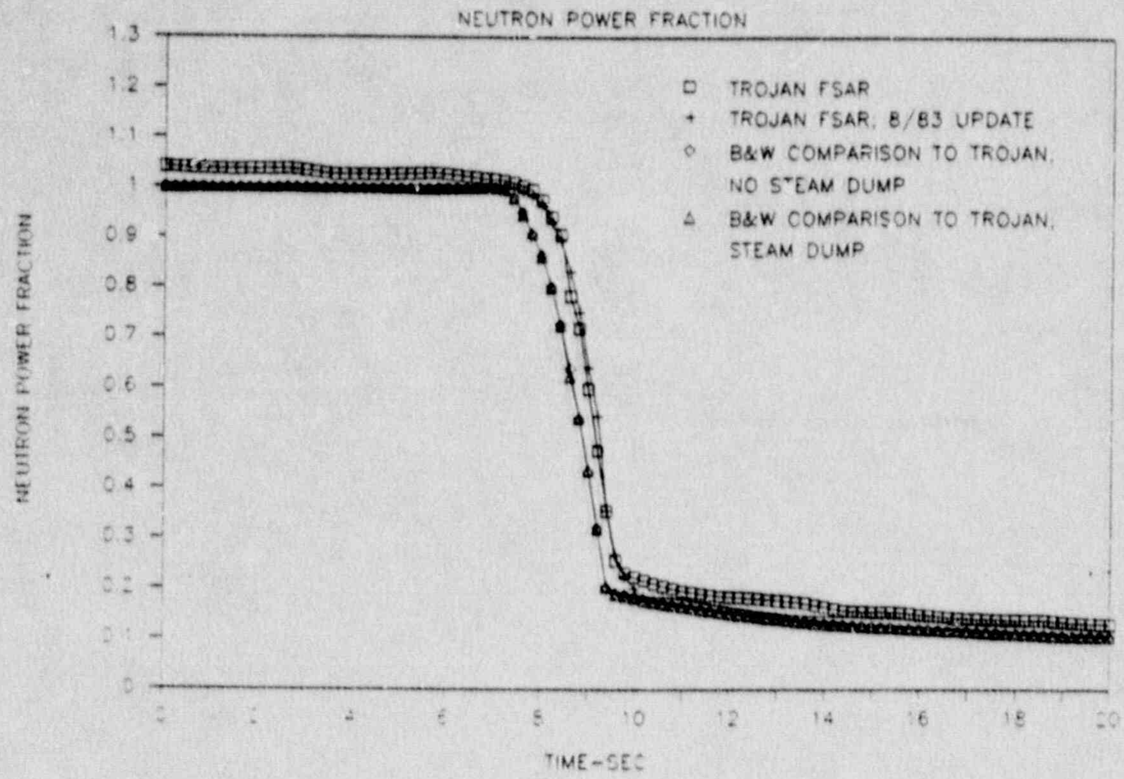


FIGURE 7.3.8. TURBINE TRIP

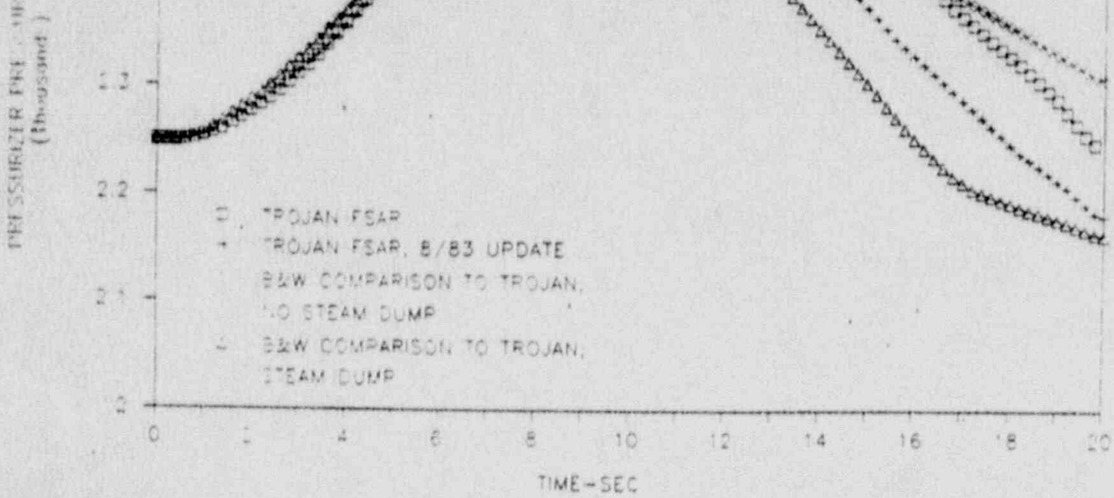


FIGURE 7.3.9. TURBINE TRIP

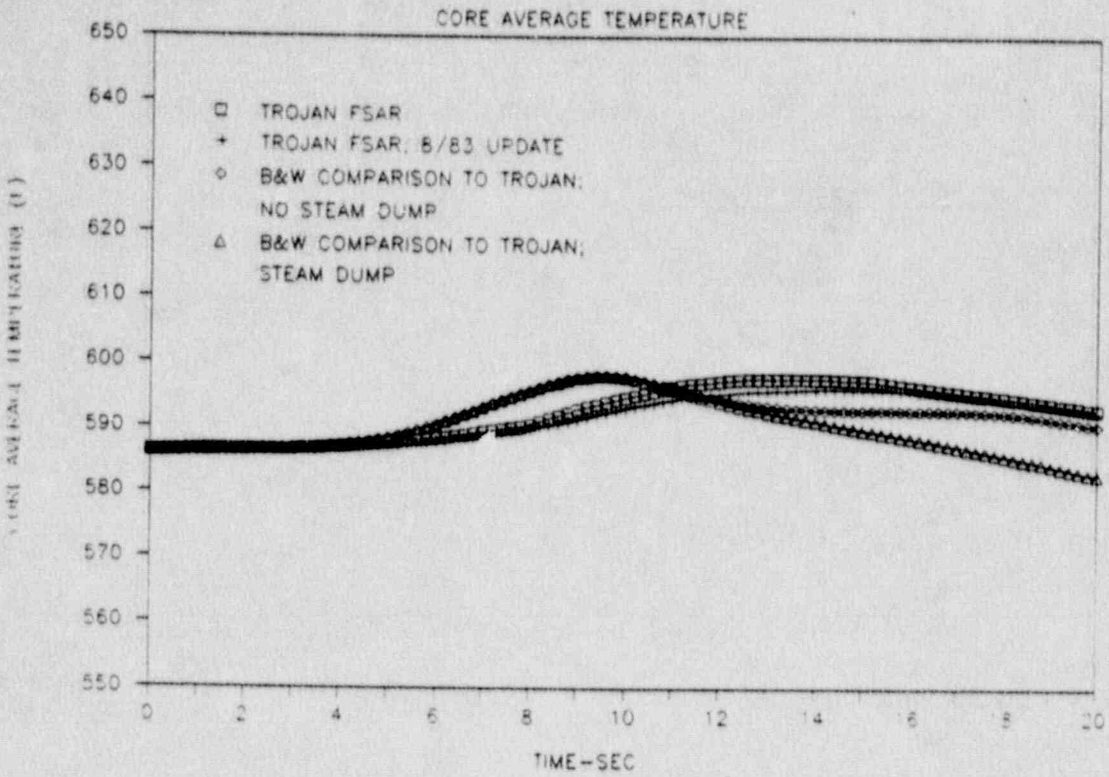


FIGURE 7.3.10. TURBINE TRIP

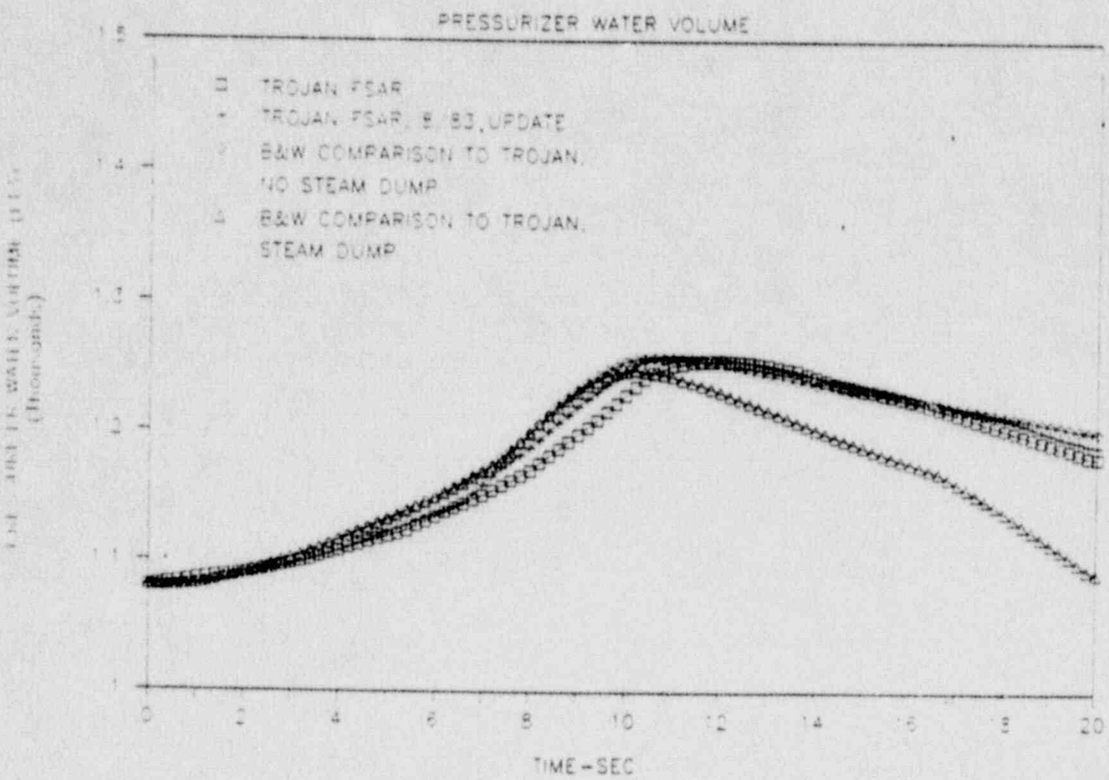


FIGURE 7.3.11. TURBINE TRIP

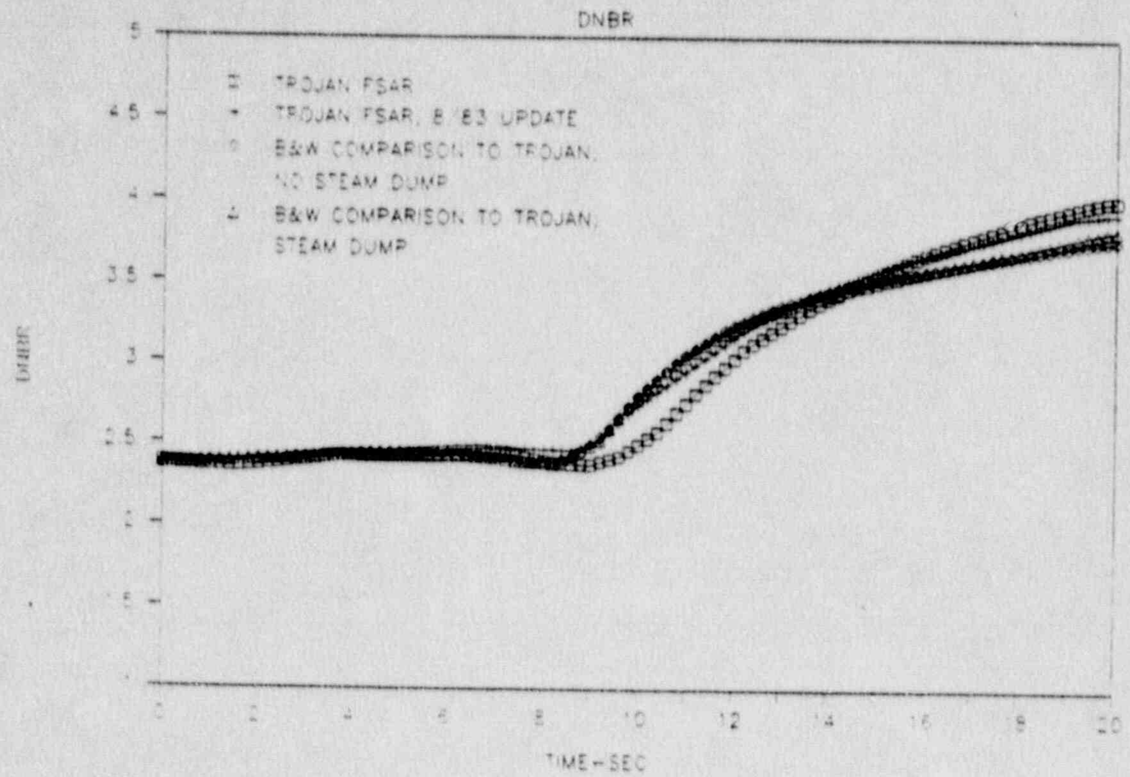


FIGURE 7.3.12. TURBINE TRIP

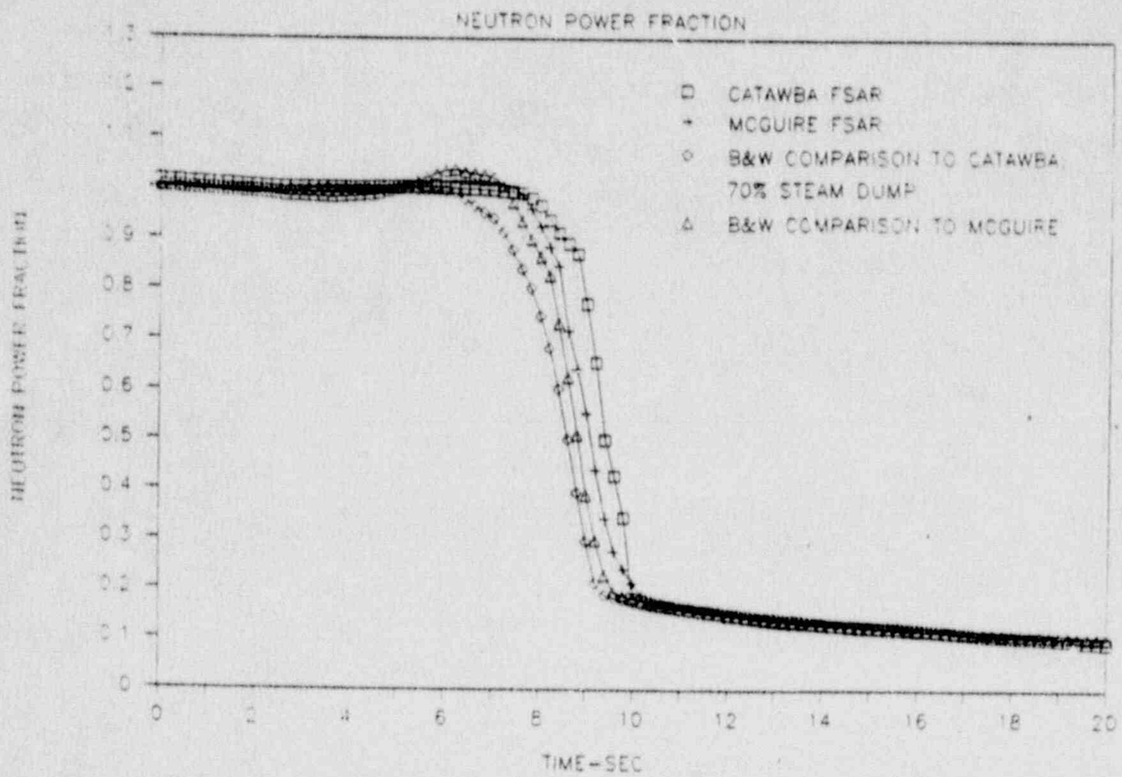


FIGURE 7.3.13. TURBINE TRIP

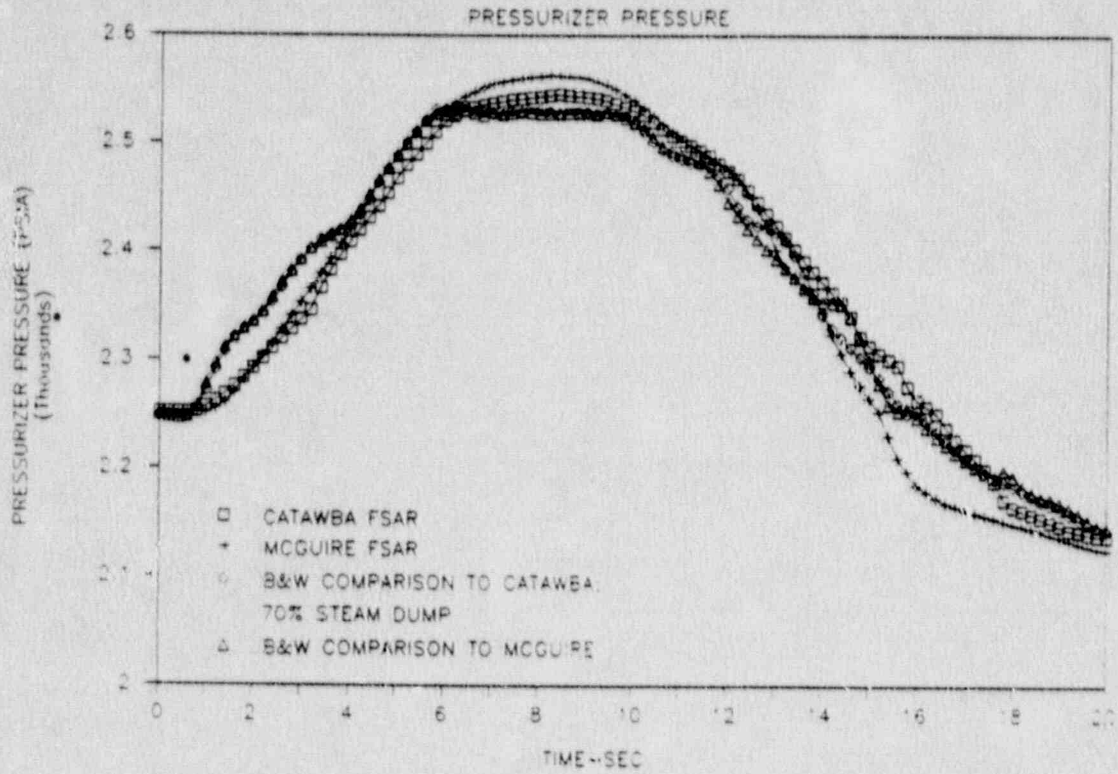


FIGURE 7.3.14. TURBINE TRIP

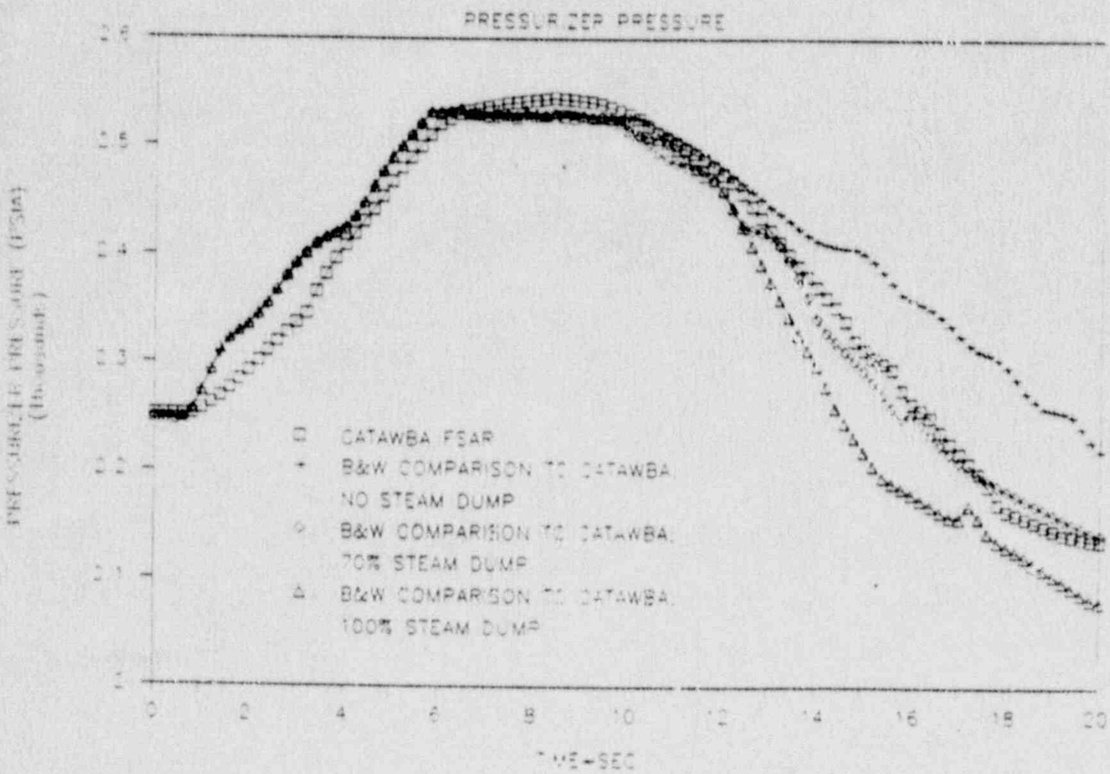


FIGURE 7.3.15. TURBINE TRIP

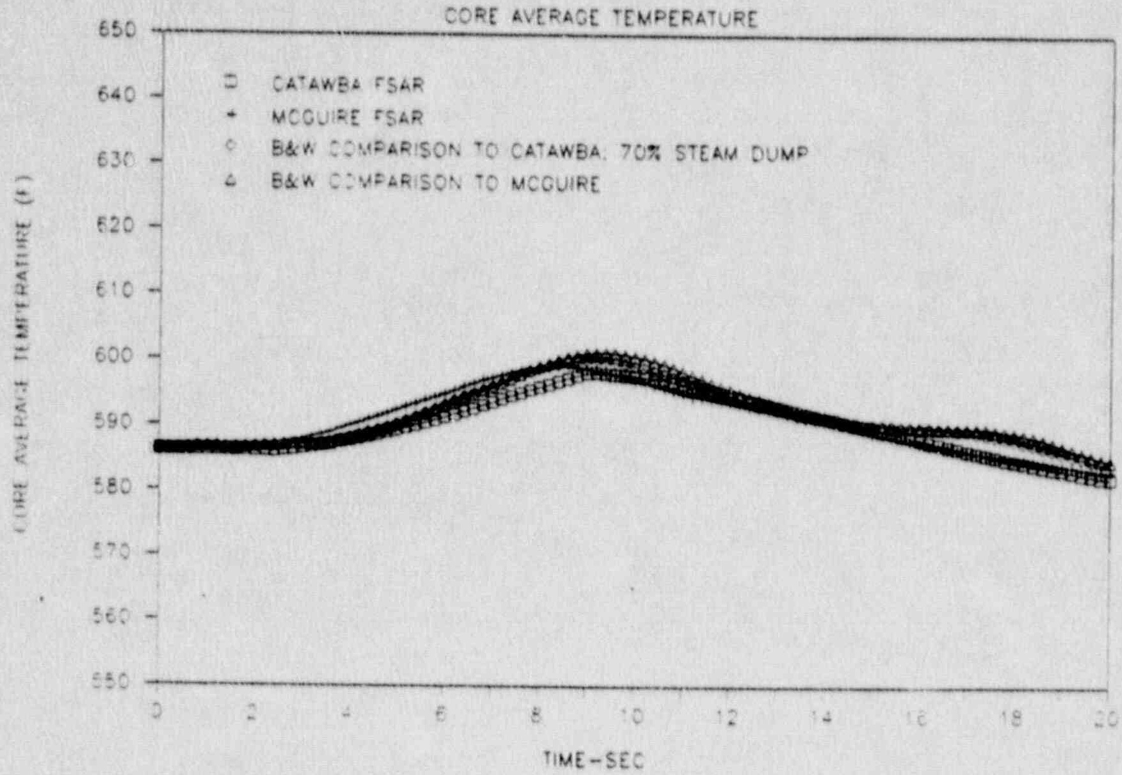


FIGURE 7.3.16. TURBINE TRIP

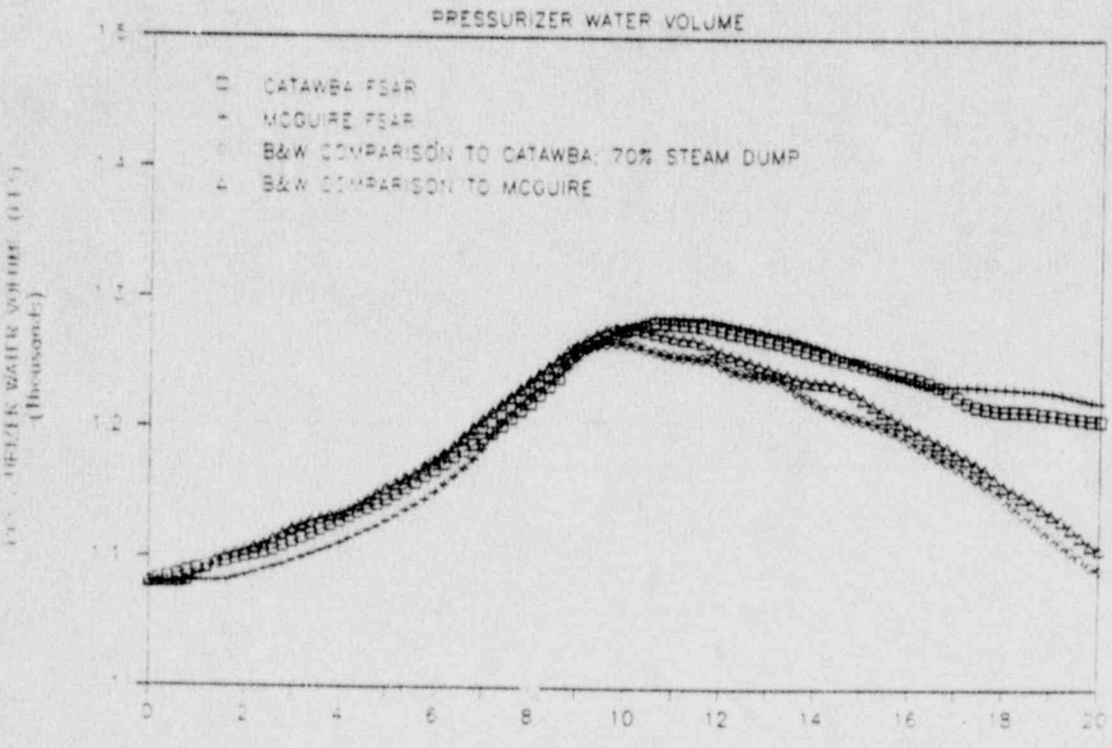


FIGURE 7.3.17. TURBINE TRIP

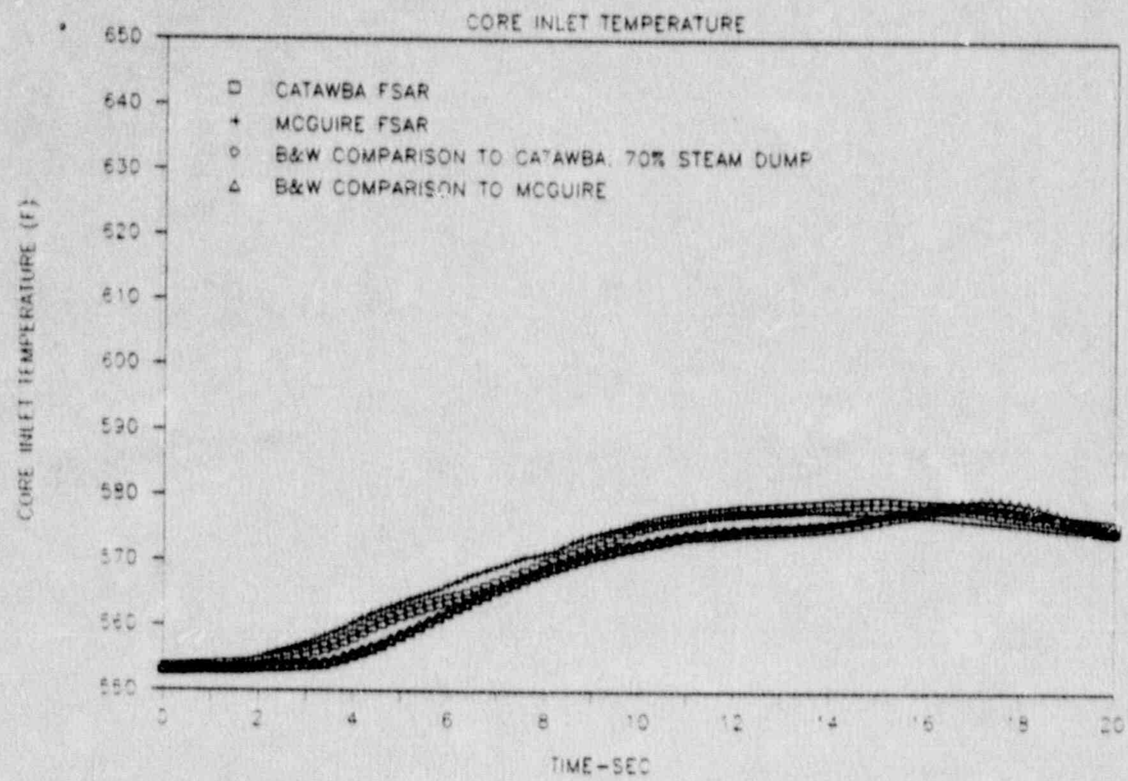


FIGURE 7.3.18. TURBINE TRIP

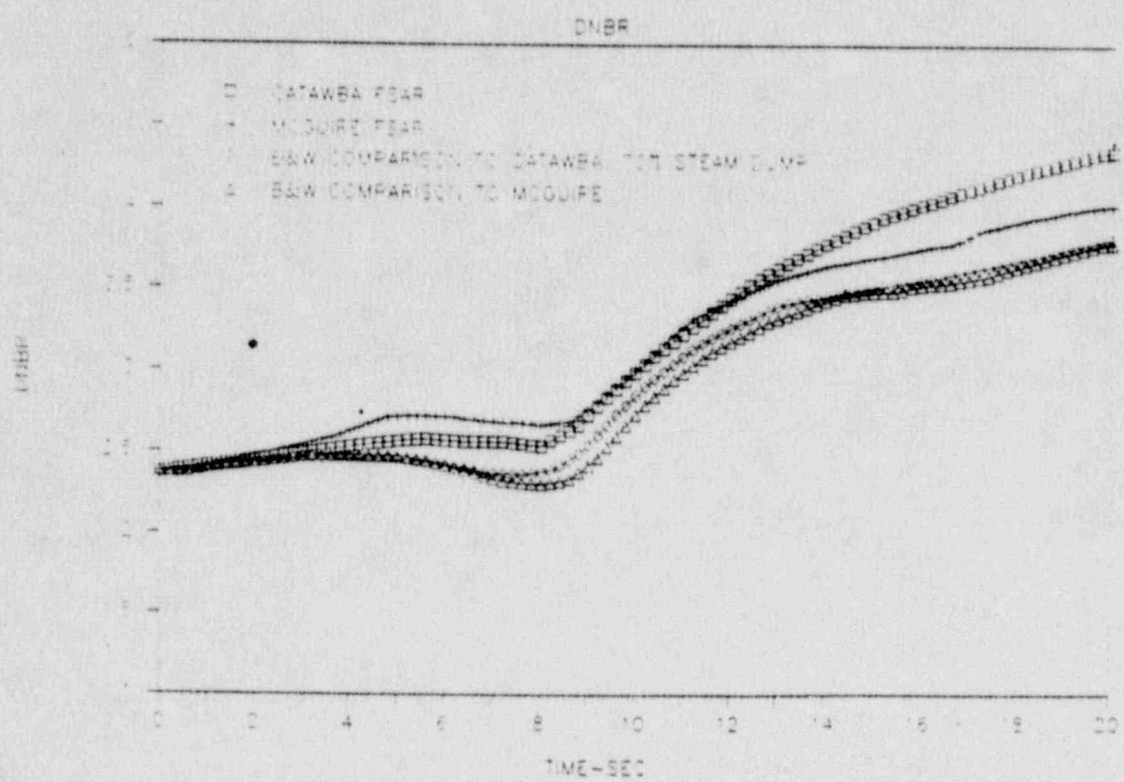


FIGURE 7.3.19. TURBINE TRIP

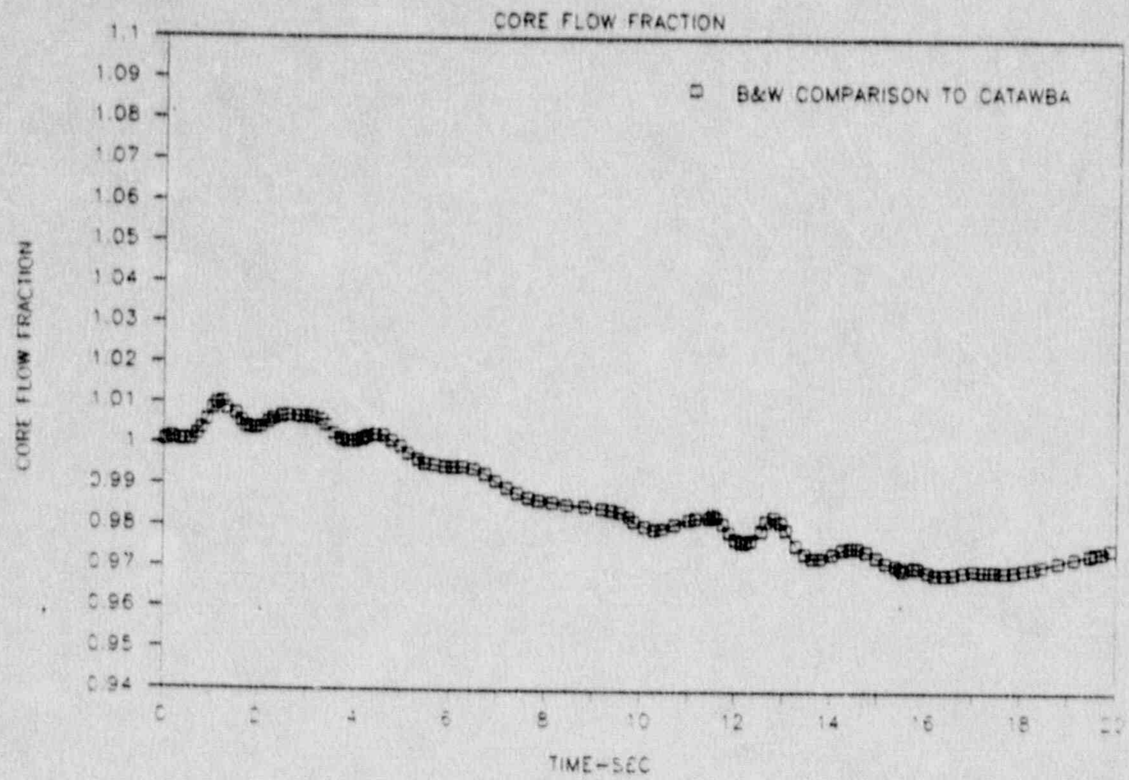


FIGURE 7.3.20. TURBINE TRIP

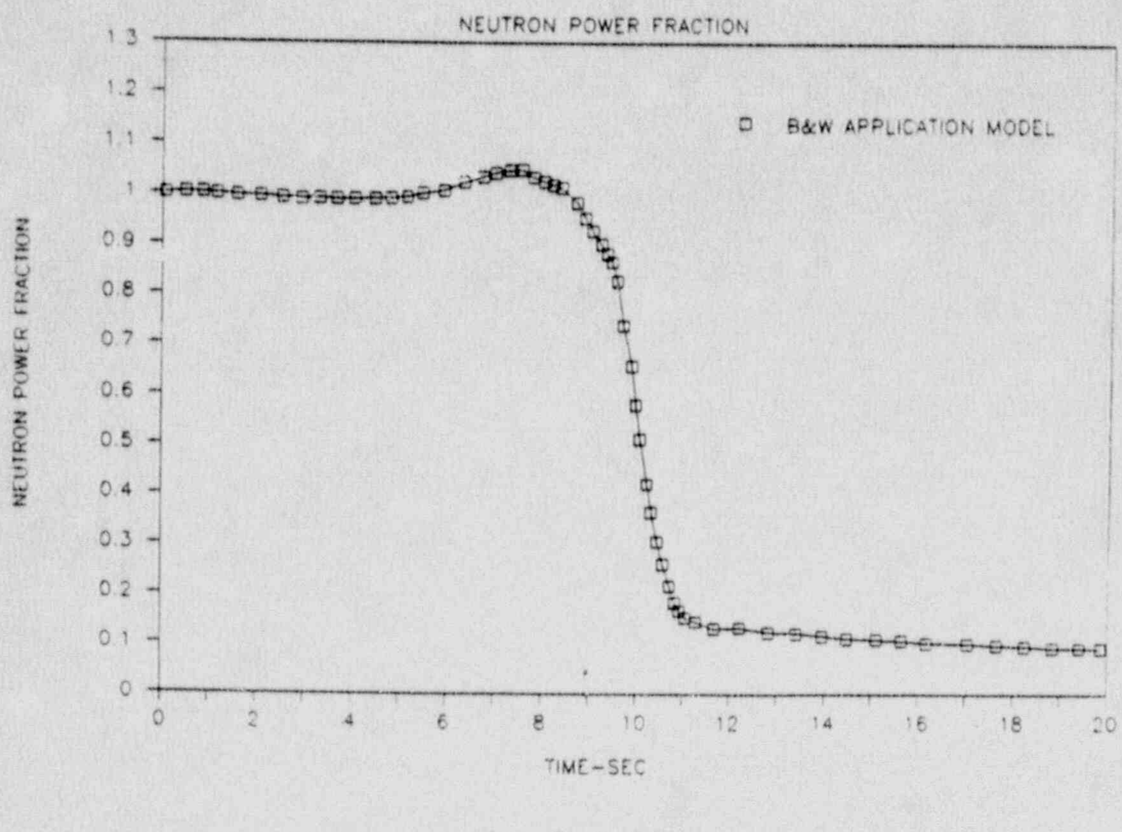


FIGURE 7.3.21. TURBINE TRIP

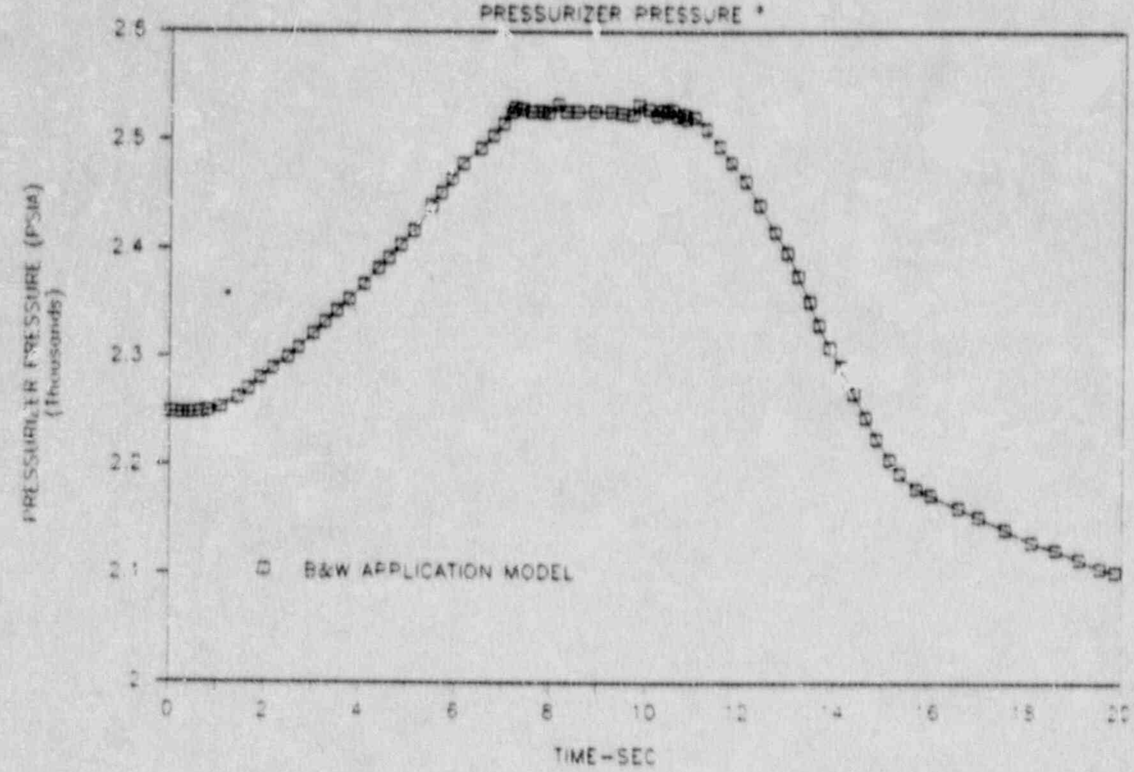


FIGURE 7.3.22. TURBINE TRIP

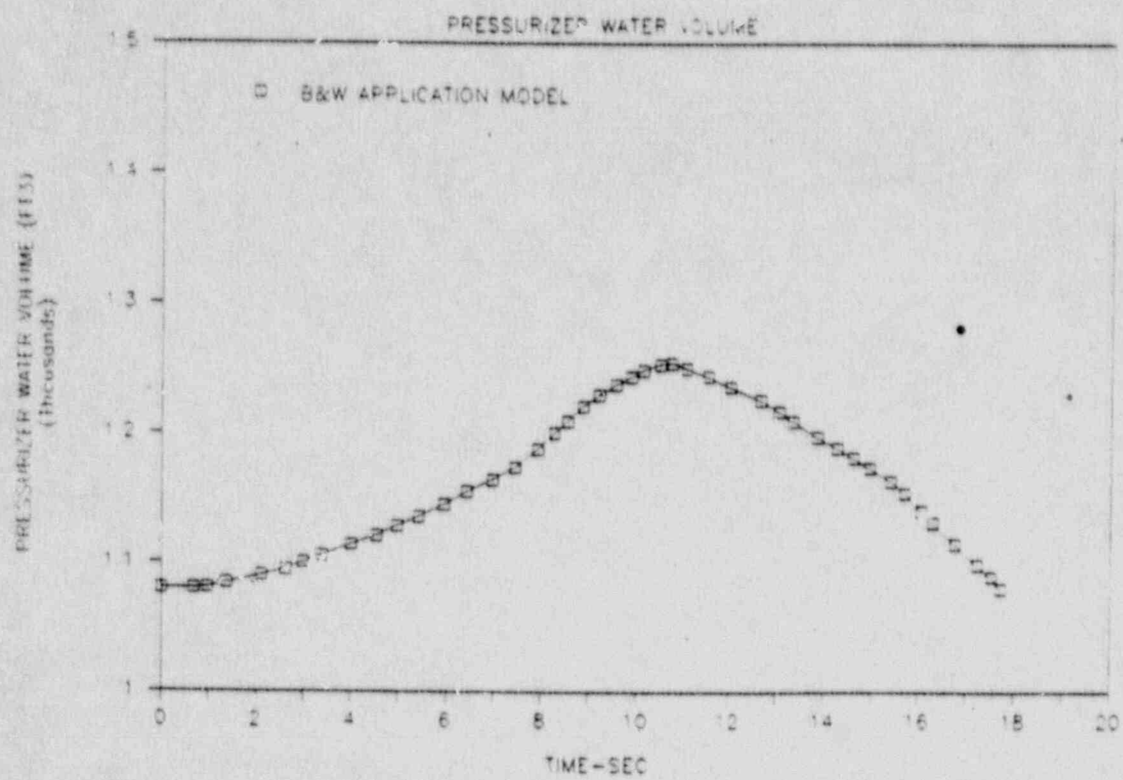


FIGURE 7.3.23. TURBINE TRIP

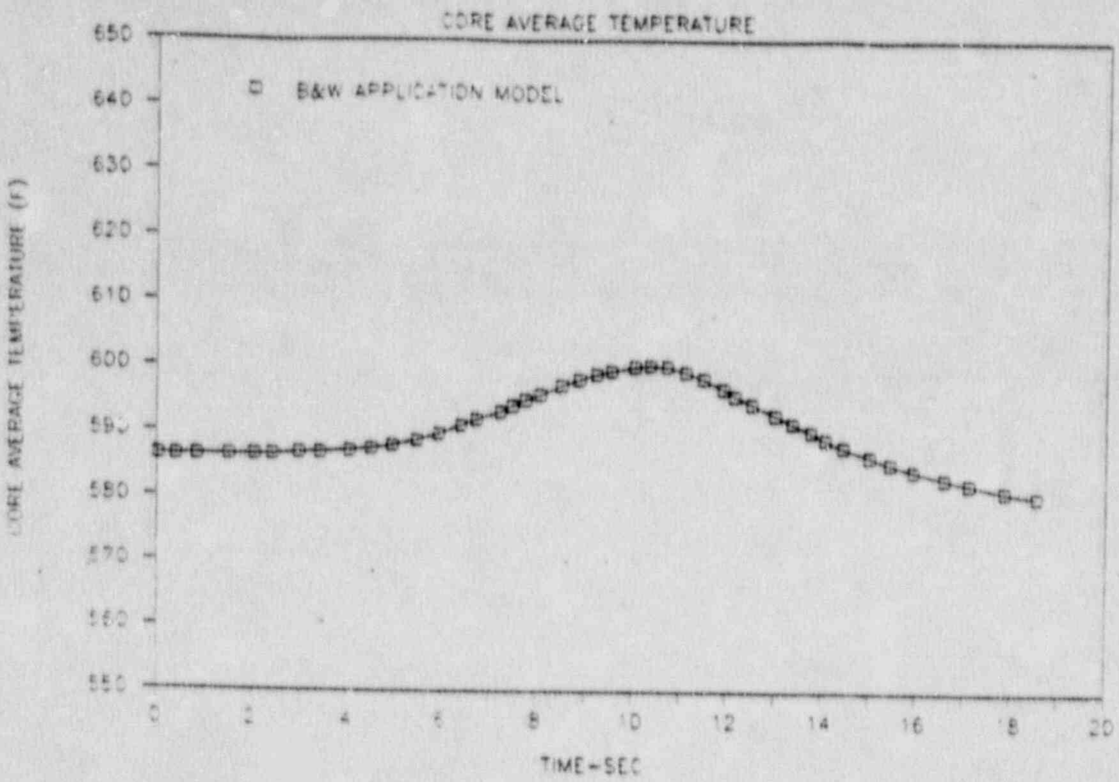


FIGURE 7.3.24. TURBINE TRIP

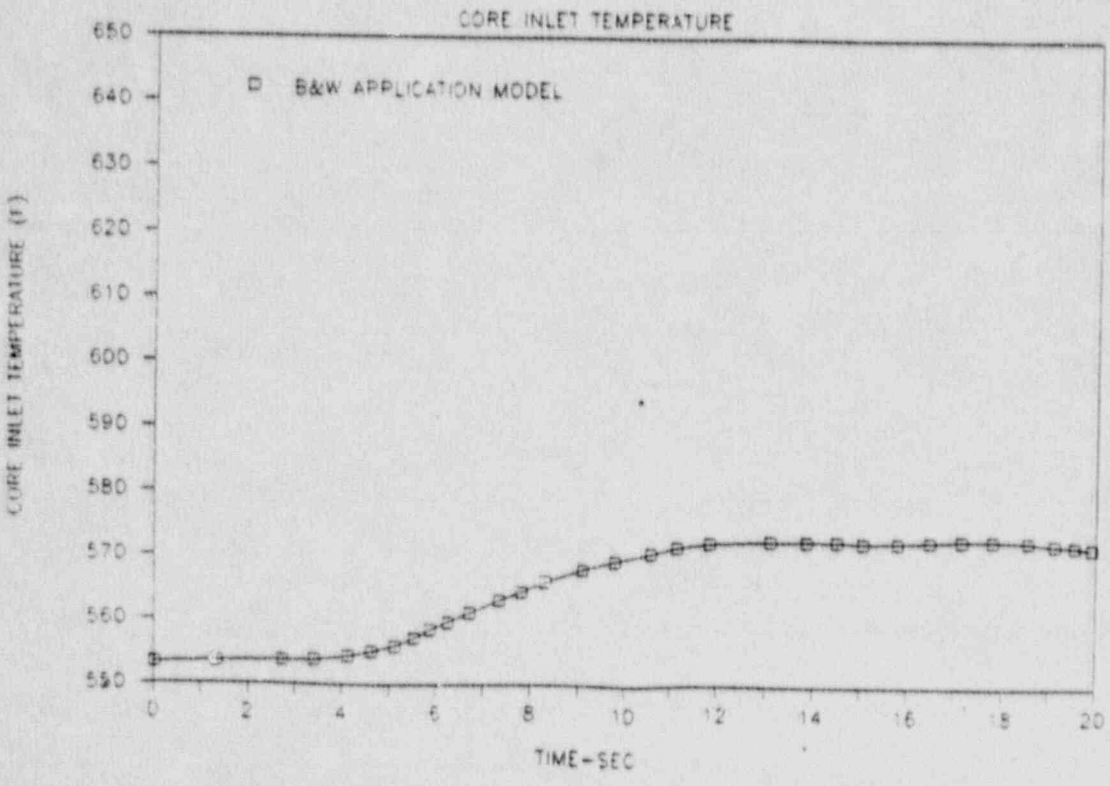


FIGURE 7.3.25. TURBINE TRIP

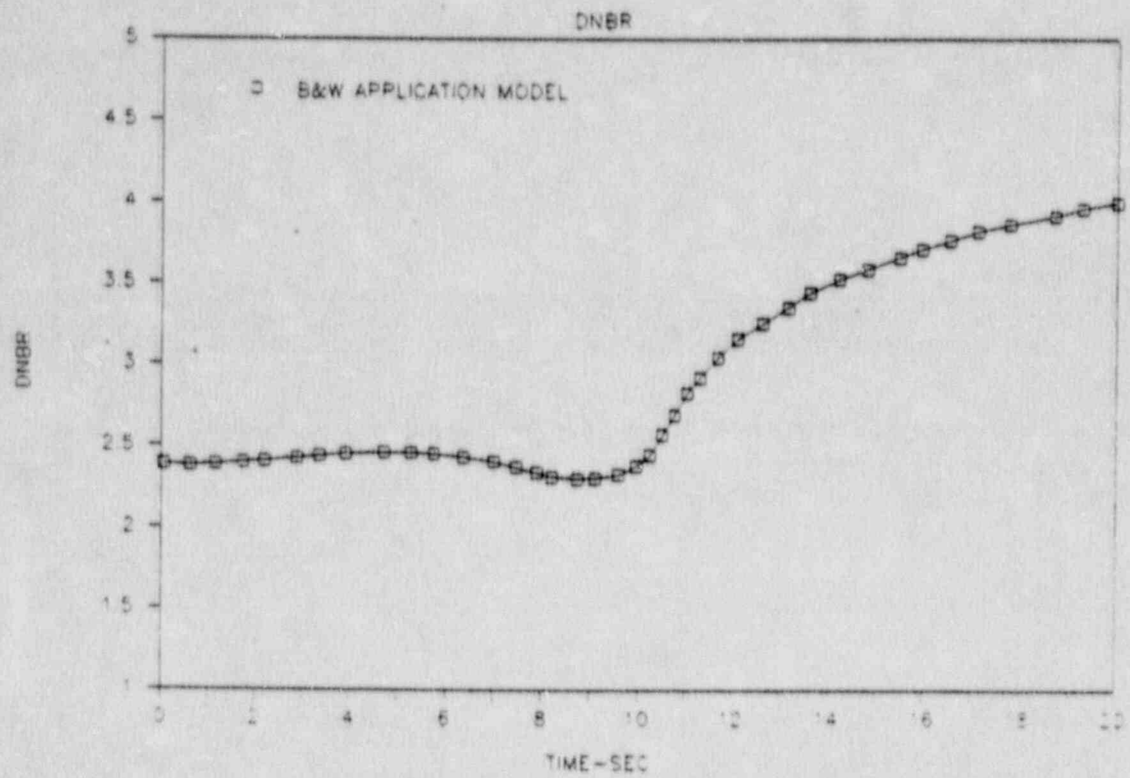
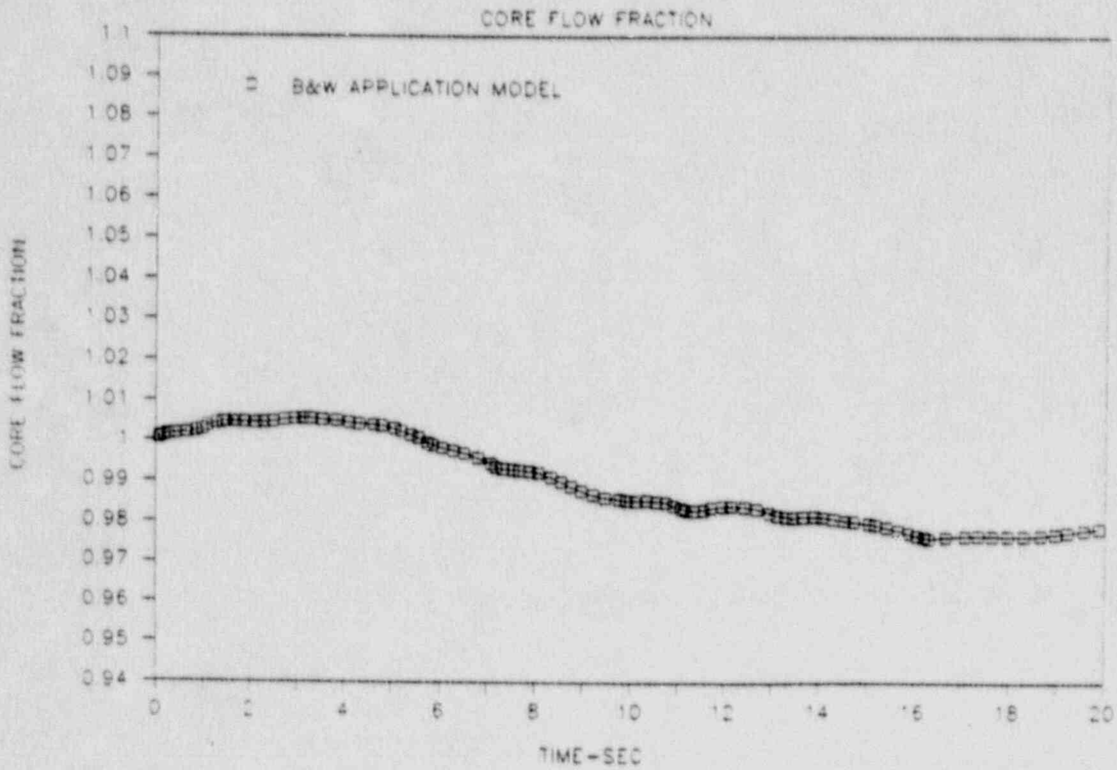


FIGURE 7.3.26. TURBINE TRIP



7.4. Reactor Coolant Pump Shaft Seizure (Locked Rotor)

7.4.1. Identification of Causes and Accident Description

The locked rotor event is postulated as an instantaneous seizure of a reactor coolant pump rotor. Flow through the affected reactor coolant loop is rapidly reduced, leading to reactor trip on a low flow signal.

Following the reactor trip, heat stored in the fuel rods continues to be transferred to the coolant causing the coolant to expand. At the same time, heat transfer to the shell side of the steam generators is reduced, first because the reduced flow results in a decreased tube side film coefficient and then because the reactor coolant in the tubes cools down while the shell side temperature increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, opens the power-operated relief valves, and opens the pressurizer safety valves, in that sequence.

A reactor coolant pump shaft seizure is classified as an ANS Condition IV event, a limiting fault. See Section 4.0 for a discussion of Condition IV events.

7.4.2. Analysis of Effects and Consequences

Method of Analysis

The reactor coolant pump shaft seizure transient has been analyzed for one seized with four loops in operation.

Two digital computer codes were used for the McGuire SAR to analyze this transient. The LOFTRAN code (Reference 7) was used to calculate the resulting loop and core flow transients following the pump seizure, the time of reactor trip based on the loop flow transients, the nuclear power following reactor trip, and the peak pressure. The thermal behavior of the fuel located at the core hot spot was investigated using the FACTRAN code (Reference 10), which uses the core flow and the nuclear power calculated by LOFTRAN. The FACTRAN code includes a film boiling heat transfer coefficient.

The B&W model analysis was done using the RELAP5/MOD2 code (Reference 6). The normal pump model in the affected loop was removed and the faulted loop flow fraction of the SAR was input as a flow versus time curve. For reload analyses, a similar flow versus time curve will be input, and the clad inner temperature will be determined using a detailed thermal-hydraulic code where the system parameters will be from RELAP5/MOD2. For this comparison calculation of clad inner temperatures, the clad film coefficient for the 12 fuel nodes was set to a value of 150 Btu/hr-ft2-F.

Initial and Boundary Conditions

Initial reactor power, pressure and RCS temperatures are assumed to be at their normal values as described in Section 7.0.

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion is assumed to begin one second after the flow in the affected loop reaches 87 percent of nominal flow. No credit is taken for the pressure reducing effect of the pressurizer spray or controlled feedwater flow after plant trip. Although these operations are expected to occur and would result in a lower peak pressure, an additional degree of conservatism is provided by ignoring their effect.

The pressurizer safety valves are fully open at 2575 psia and their capacity for steam relief is as described in Section 7.0. The value of 2575 psia includes allowance for a 30 psi pressure error and 15 psi for valve opening.

Evaluation of DNB in the Core During the Accident

For this accident, DNB is assumed to occur in the core, and therefore, an evaluation of the consequences with respect to fuel rod thermal transients is performed. Results obtained from analysis of this "hot spot" condition represent the upper limit with respect to clad temperature and zirconium water reaction. In the McGuire SAR evaluation, the rod power at the hot spot is assumed to be 2.5 times the average rod power (i.e., $F_Q = 2.5$) at the initial core power level.

Film Boiling Coefficient

The film boiling coefficient used for the McGuire SAR analysis was calculated in the FACTRAN code using the Bishop-Sandberg-Tong film boiling correlation. The fluid properties were evaluated at film temperature (average between wall and bulk temperatures). The program calculates the film coefficients at every time step based upon the actual heat transfer conditions at the time. The neutron flux, system pressure, bulk density and mass flow rate as a function of time were used as program input.

For the McGuire SAR analysis, the initial values of the pressure and the bulk density were used throughout the transient since they are the most conservative with respect to clad temperature response. For conservatism, DNB was assumed to start at the beginning of the accident.

Fuel Clad Gap Coefficient

The magnitude and time dependence of the heat transfer coefficient between fuel and clad (gap coefficient) have a pronounced influence on the thermal results.

The larger the value of the gap coefficient, the more heat is transferred between pellet and clad. Based on investigations on the effect of the gap coefficient upon the maximum clad temperature during the transient, the gap coefficient for the McGuire SAR analysis was assumed to increase from a steady state value consistent with initial fuel temperature to 10,000 BTU/hr-ft^2-F at the initiation of the transient. Thus, the large amount of energy stored in the fuel, because of the small initial value, is released to the clad at the initiation of the transient.

Reactivity Coefficients

The locked rotor transient is analyzed with minimum reactivity feedback. The minimum feedback cases assume a least negative moderator temperature coefficient and the least negative Doppler coefficients. For the B&W model analysis, a Doppler fuel temperature coefficient consistent with the time in life for this analysis was used.

A moderator coefficient of zero was used for the Trojan, RESAR and Catawba analysis. A positive moderator temperature coefficient of +5 pcm/F was used for the McGuire analysis. A positive moderator temperature coefficient of +7 pcm/F was used for the B&W application model analysis.

Reactor Trips

After pump seizure, the neutron flux is rapidly reduced by control rod insertion. Rod motion is assumed to begin one second after the flow in the affected loop reaches 87 percent of nominal flow.

7.4.3. Results

Safety Analysis Report Data

The SAR results for the locked rotor transient are plotted in Figures 7.4.1 through 7.4.6. The key outputs for these analyses are calculated system pressures and cladding temperatures. The locked rotor event has both reduced core flow and degraded steam generator conductance associated with the reduction in flow for both the affected loop and unaffected loops. Typically, the flow coastdown associated with pump shaft seizure in one loop is calculated separately and used as flow versus time input to the system response calculation. In addition to the two major outputs, parameters presented include both neutron and thermal power, core inlet and/or maximum system pressures, and the core and faulted loop flow fractions.

In examining the SAR analyses, it is noteworthy that the neutron and thermal power curves align directly according to the respective tripped rod insertion times for RESAR/Trojan and Catawba/McGuire. The times to trip are based upon the low flow setpoint, which occurs at effectively the same time in all four cases. There are no remarkable differences among the flow coastdown plots, yet the core inlet/peak system pressure curves differ significantly.

The pressure versus time SAR results, Figure 7.4.3, reflect significant differences among the analyses in both method and assumptions. The Trojan results, taken from a 1972 analysis, were generated using an early version of LOFTRAN. That version utilized a single loop simulation of the primary coolant system. The later analyses, RESAR and Catawba/McGuire, were done with a multiloop LOFTRAN code. The multiloop code allows the analyst to model the steam generators separately, according to the markedly different primary flows in the affected loop and unaffected loops. The single loop model forces the analyst to model all of the steam generators as a single unit using a bounding assumption on conductance or heat demand. Thus, the Trojan results reflect the most conservative analysis obtainable by zeroing the steam generator conductance at the start of the transient. The RESAR and Catawba/McGuire analyses were able to distinguish between the generators in the affected loop and unaffected loops, producing less severe pressure increases.

Closer examination of these later SAR results reveals the effect of treatment of a loss-of-offsite power (LOOP) coincident with the locked rotor event. The McGuire and RESAR pressure curves exhibit the LOOP impact upon the post-trip pressure increase, whereas the Catawba SAR analysis considers only the effect of the flow coastdown associated with the locked rotor and of the turbine trip (coincident with reactor trip in this case) in determining the primary pressure response. (This is confirmed in Chapter 13 of Reference 1 in response to NRC review questions.) The LOOP produces an earlier, somewhat higher peak pressure.

Trojan/RESAR Comparisons

Comparisons of RELAP5 results to those reported for Trojan and in the RESAR are presented in Figures 7.4.7 through 7.4.11. One of the RELAP5 runs was made with the steam generator conductances zeroed at the start of the transient to most closely emulate the Trojan analysis. The RESAR comparison case modeled flow dependent steam generator heat demand in the affected loop, the single steam generator. In the unaffected loops, the triple steam generator conductance was reduced (by 35 percent) to simulate coastdown of the pumps in those loops for loss of offsite power conditions (LOOP). The faulted loop flow data used as inputs to RELAP5 were available only for the RESAR, so both the Trojan and RESAR comparison runs were made with the same faulted loop flow versus time.

Correspondence between the RELAP5 analyses and the Trojan and RESAR curves is quite reasonable, given the more limited information at hand for duplicating these older SAR calculations. Use of the RESAR faulted loop flow for the Trojan comparison contributes to the mismatch in neutron power fraction in Figure 7.4.7 since the reactor trip signal is on low flow. Here again, the Trojan post-trip power drops off more steeply than the other SAR's and the RELAP5 cases evidence, suggesting a different method in feedback modeling.

Examining the pressurizer pressure responses in Figure 7.4.9 shows some delay in the RELAP5 curves, with peak pressures lower by about 40 psi. The active surge line model in the RELAP5 analyses and the multivolume pressurizer would be expected to produce this kind of difference in response. Nonetheless, the rates of pressure increase and decrease are well matched, and both RELAP5 cases line up with the SAR curves past the peak pressure.

Catawba and McGuire Comparisons

Figures 7.4.12 through 7.4.16 show the comparisons of the RELAP5 simulations of the Catawba and McGuire locked rotor cases to the SAR results. The significant differences between these cases are in the tripped rod insertion times, the moderator coefficients, and inclusion of the loss-of-offsite power (LOOP) effect during the event. The Catawba comparison case was done with 3.05 seconds to 85 percent tripped rod insertion time. A zero moderator coefficient was used, and no adjustment for LOOP effect was modeled. The SAR faulted loop flow was input, and steam generator heat transfer was allowed to vary according to the flow reduction in all loops. For the McGuire case, a 2.7 seconds to

85 percent insertion time was used for the tripped rod. A positive moderator coefficient, +5.0 pcm/F, was used along with simulation of the LOOP via reduction of the triple RSG conductance. A reduction of 35 percent in that parameter is representative of the additional pressure rise effect of coastdown of the three RC pumps in the unaffected loops. The steam dump system was assumed active for the McGuire analysis.

The effects of the different tripped rod insertion times and moderator coefficients are readily discernible in the neutron power fractions plotted in Figure 7.4.12. The RELAP5 results line up well with both of the SAR curves. There is also good agreement in the thermal power fractions compared in the following figure. The relatively minor differences between the RELAP5 and SAR curves in the neutron and thermal power plots are probably due to methods used in feedback modeling. The RELAP5 analyses used a constant, minimum Doppler coefficient with no temperature feedback.

Pressurizer pressures, shown in Figure 7.4.14, indicate close agreement between the SAR and RELAP5 calculations. As in the Trojan/RESAR comparisons, the active surge line modeled in RELAP5 simulates the slight delay in pressurizer response during the first second. Otherwise the rates of increase and decrease and the peak pressures produced by RELAP5 virtually reproduce the plant SAR plots.

The faulted loop flow fractions taken from the Catawba and McGuire SAR's (Figure 7.4.16) were, as stated, used as inputs to the RELAP5 calculations. The flow fraction curves are quite similar for the two plant SAR's, but there is a difference in the final flow fraction that is sustained. Examination of the core flow fraction curves of Figure 7.4.15 shows that the core flow fractions actively calculated in RELAP5 show the same effect of the difference in faulted loop flows as do the Catawba and McGuire curves.

Clad Inner Temperature Comparison

Figure 7.4.17 shows a comparison of RELAP5 results with Catawba/McGuire/Trojan SAR results for clad inner temperature. The RELAP5 run was done with a 3.05 seconds to 85% tripped rod insertion (like Catawba) and the clad heat transfer coefficient for a 12 fuel nodes set to 150 Btu/hr-ft2-F at the start of the transient. This is equivalent to an analysis assumption of DNB at the start of the transient. For design analysis the clad inner temperature will be calculated using a detailed thermal-hydraulic code with system parameters from RELAP5. The RELAP5 results show good agreement to McGuire results during the early part of the transient and consistent agreement with all the data for the rest of the transient.

B&W Application Model

Figures 7.4.18 through 7.4.22 show RELAP5 analysis results for the B&W application model. The difference between the Catawba/McGuire comparison RELAP5 runs and the B&W model run are:

1) The B&W application model has a five region fuel pin, a 1 region gap and a 2 region clad model.

2) A moderator temperature coefficient of +7.0 pcm/F was used and a fuel temperature Doppler reactivity was used.

3) The Catawba SAR affected loop flow versus time curve was used.

4) A 70% steam dump is used but no RSG tube heat transfer coefficient reduction is used because the RELAP5 model has an explicit RC pump model.

The peak reactor coolant system pressure reached during the transient is less than that which would cause stresses to exceed the faulted condition stress limits, thus the integrity of the primary coolant system is not endangered. Also, the peak clad surface temperature is expected to be considerably less than 2700 F because the RELAP5 inner clad temperature results of Figure 7.4.17 are consistent with the results from several plant SARs. It should be noted that the clad temperature was conservatively calculated assuming that DNB occurs at the initiation of the transient. For peak clad surface temperatures in this range, the core will remain in place and intact with no loss of core cooling capability.

Figure 7.4.21 shows that the core flow rapidly reaches a new equilibrium value. With the reactor tripped, a stable plant condition will eventually be attained. Normal plant shutdown may then proceed.

7.4.4. Conclusions

The RELAP5 model has shown agreement with SAR locked rotor analysis results for several different plants and is therefore a valid code for the analysis of any specific plant when the appropriate plant geometry and plant parameters are modeled.

FIGURE 7.4.1. LOCKED ROTOR

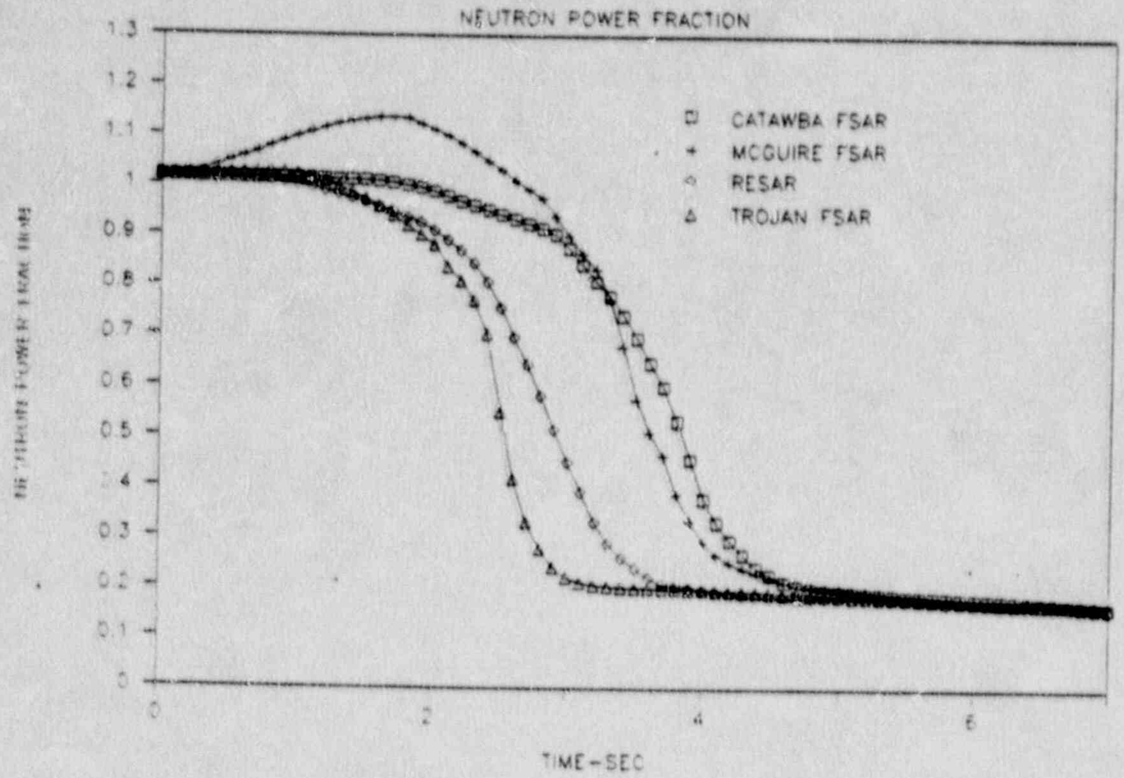


FIGURE 7.4.2. LOCKED ROTOR

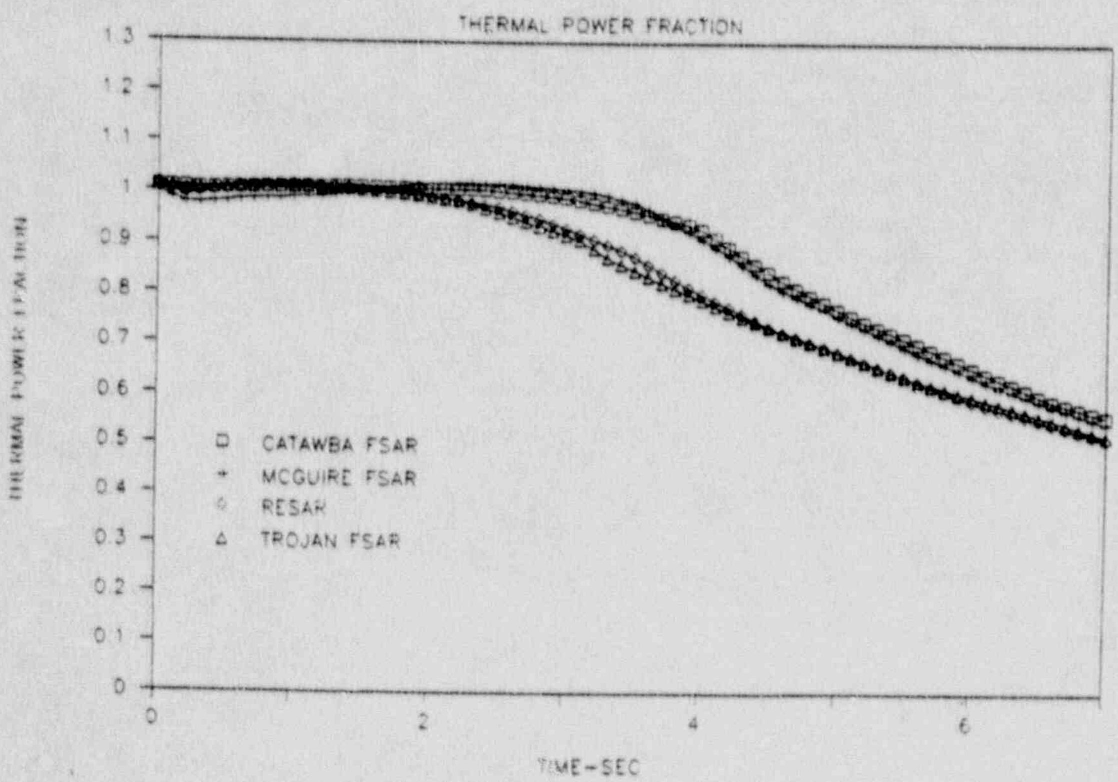


FIGURE 7.4.3. LOCKED ROTOR

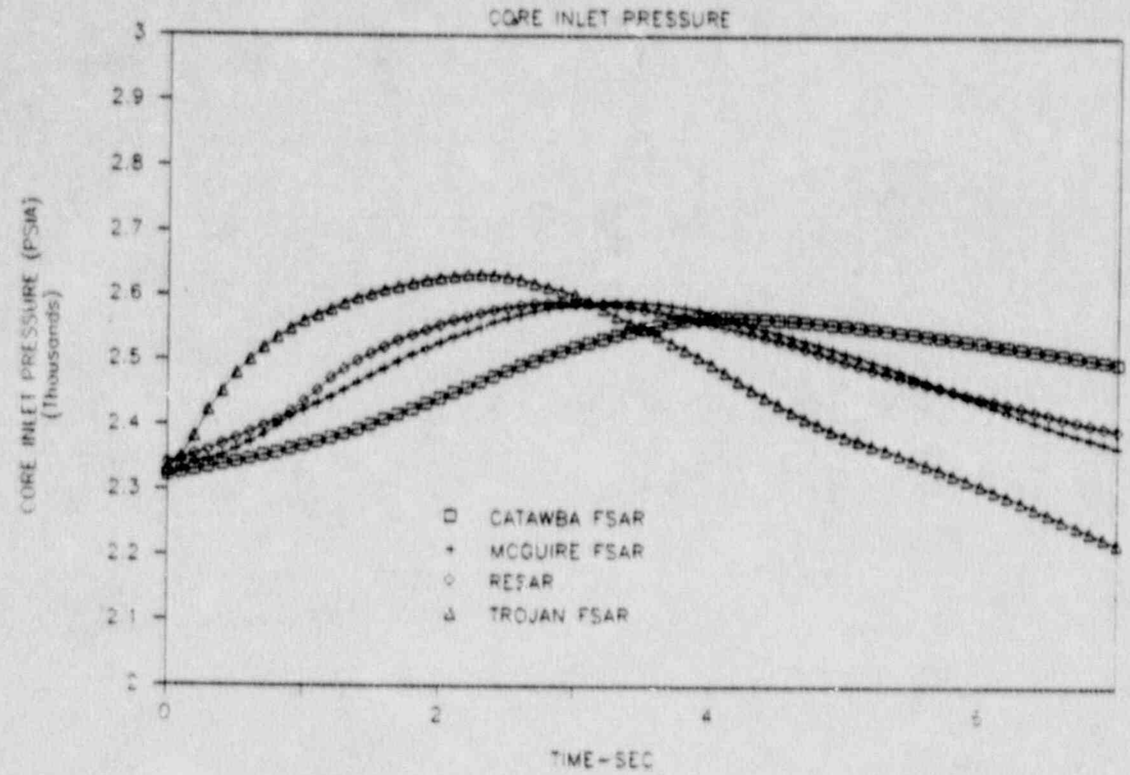


FIGURE 7.4.4. LOCKED ROTOR

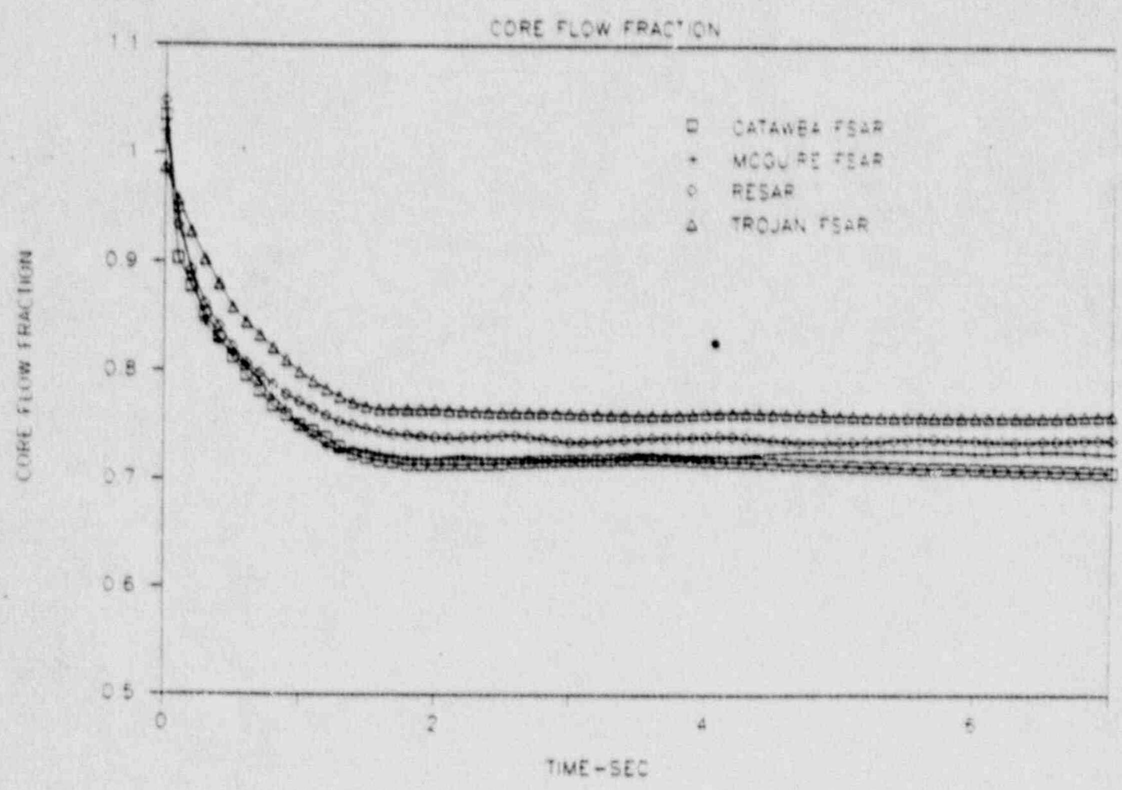


FIGURE 7.4.5. LOCKED ROTOR

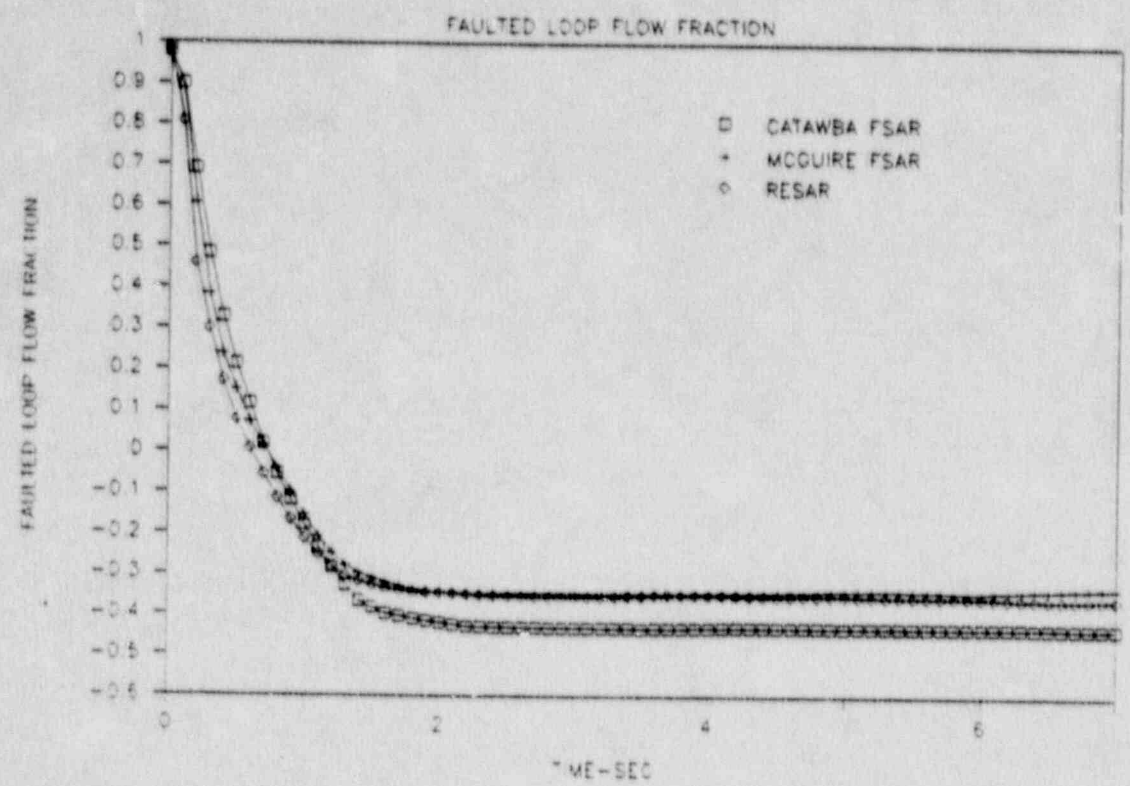


FIGURE 7.4.6. LOCKED ROTOR

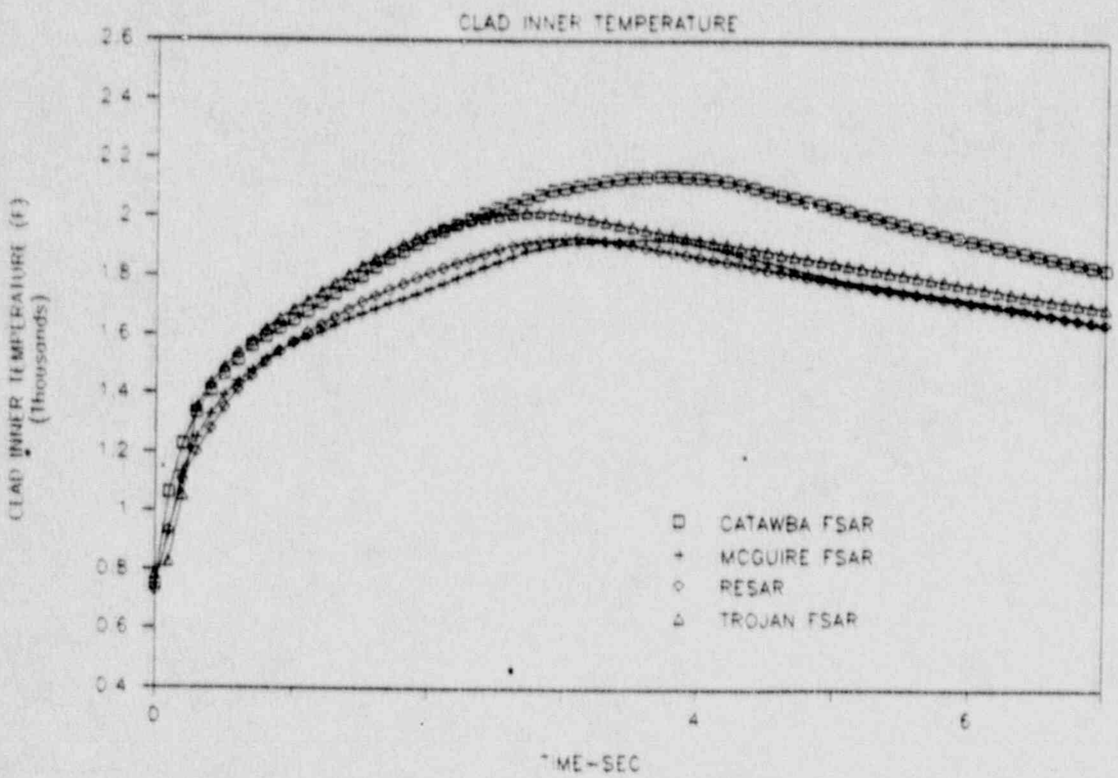


FIGURE 7.4.7. LOCKED ROTOR

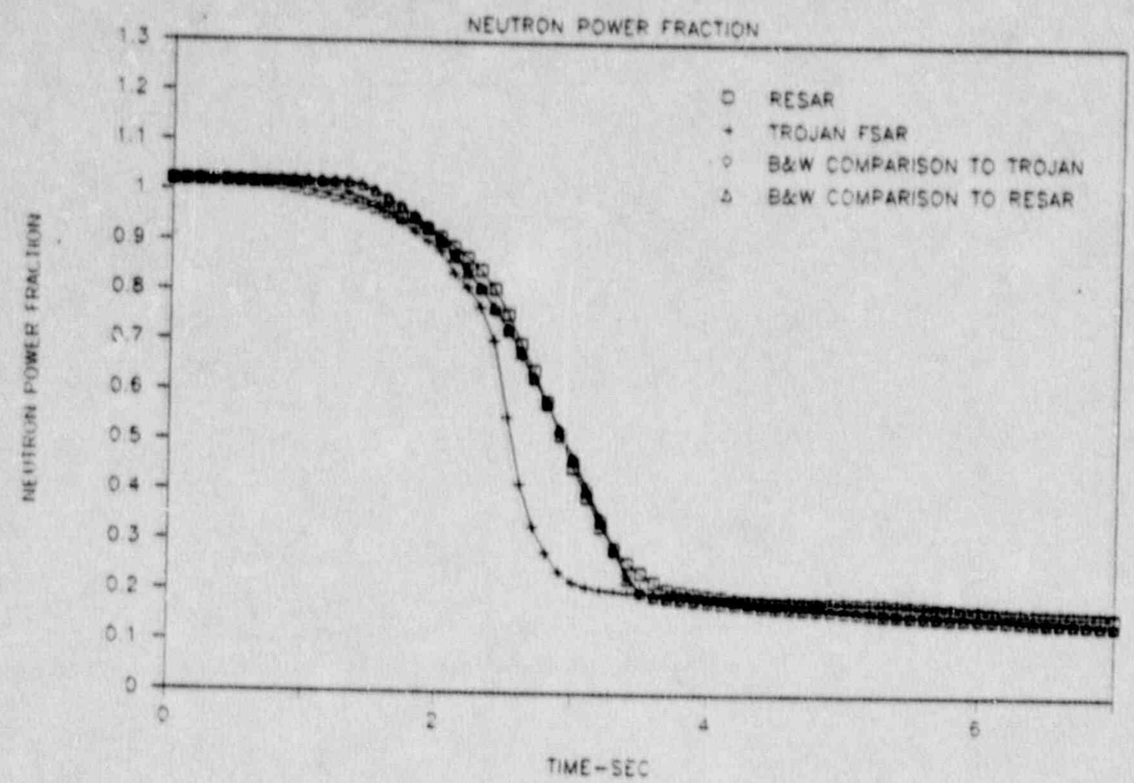


FIGURE 7.4.8. LOCKED ROTOR

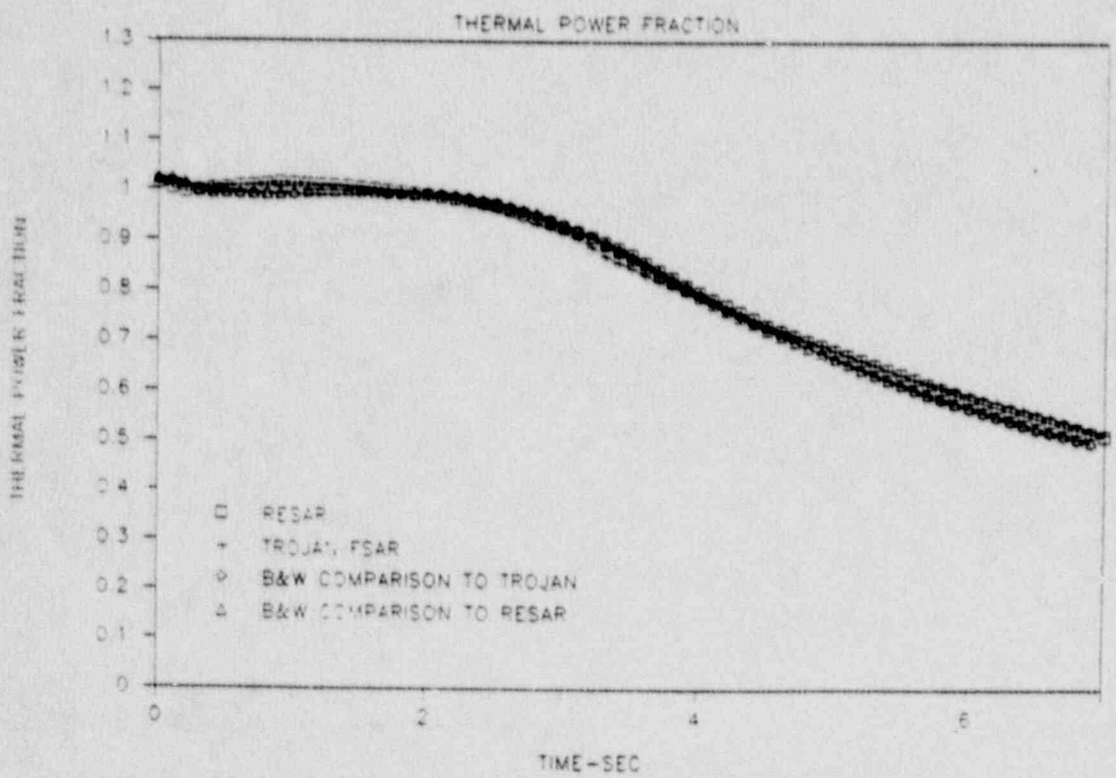


FIGURE 7.4.9. LOCKED ROTOR

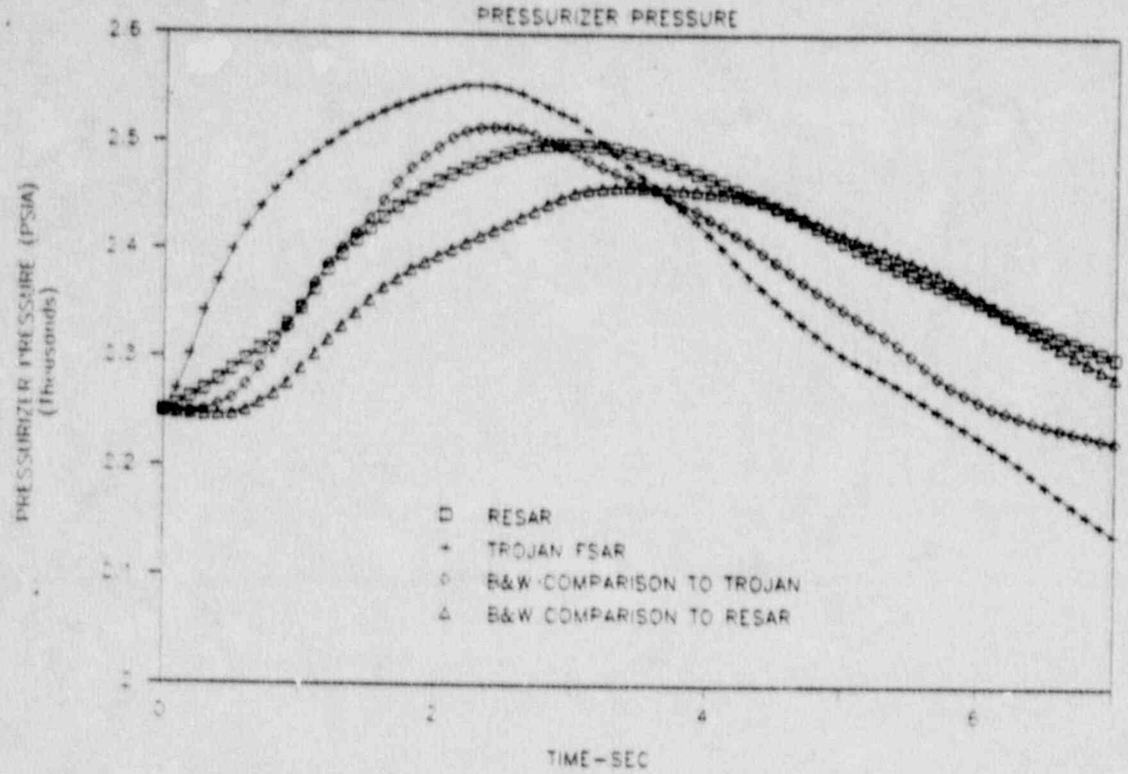


FIGURE 7.4.10. LOCKED ROTOR

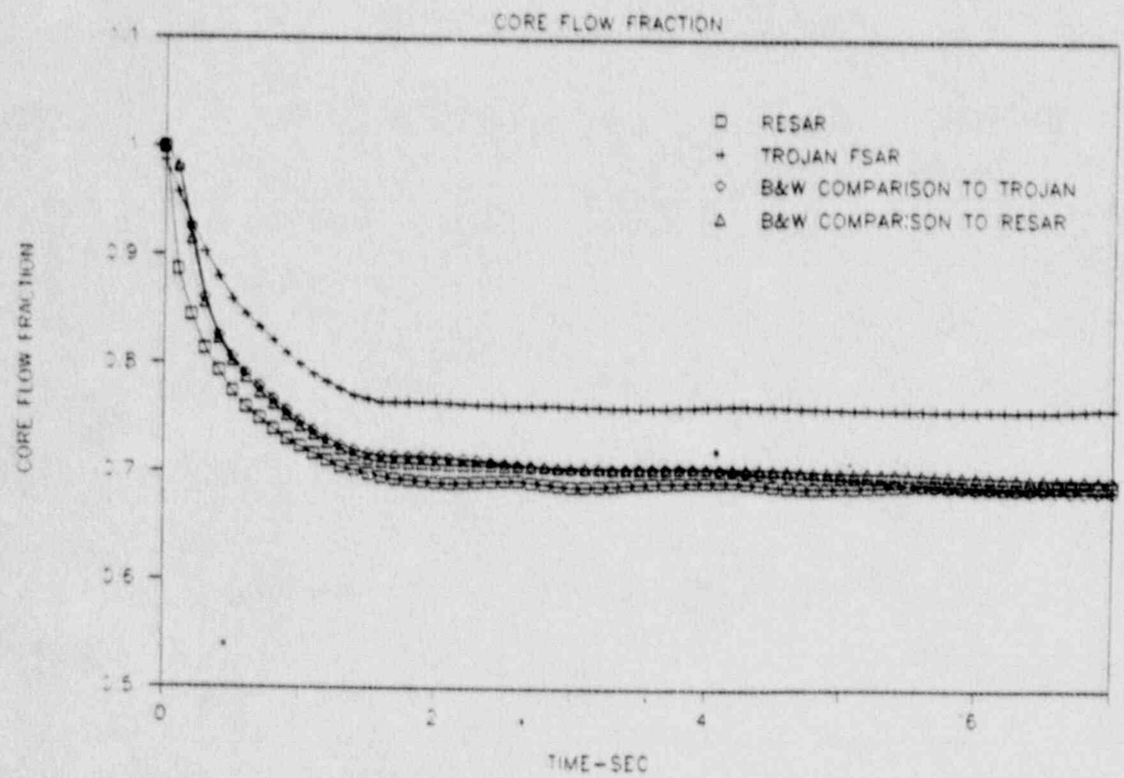


FIGURE 7.4.11. LOCKED ROTOR

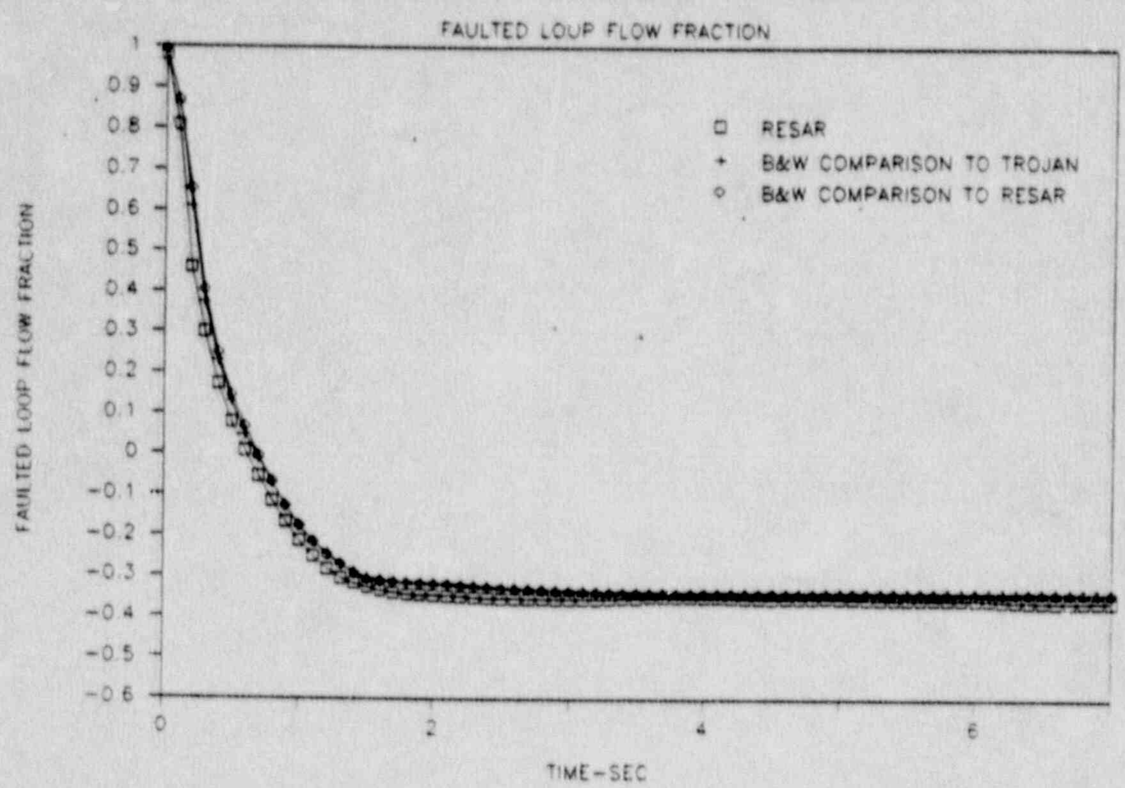


FIGURE 7.4.12. LOCKED ROTOR

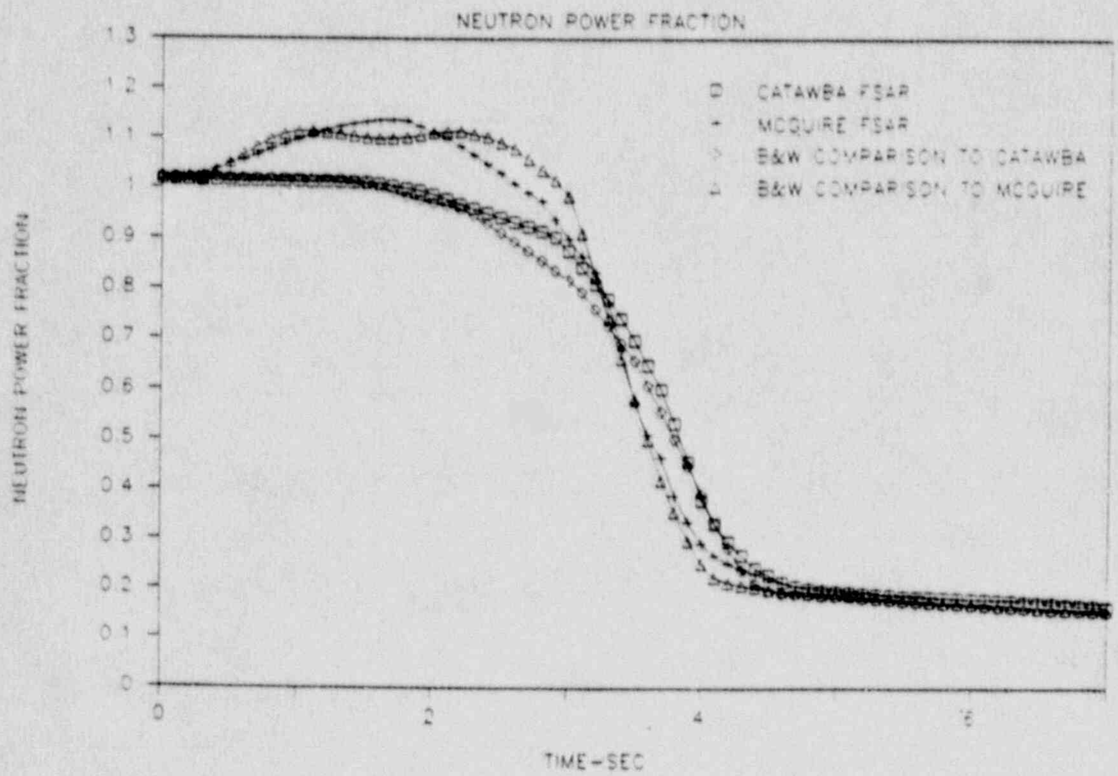


FIGURE 7.4.13. LOCKED ROTOR

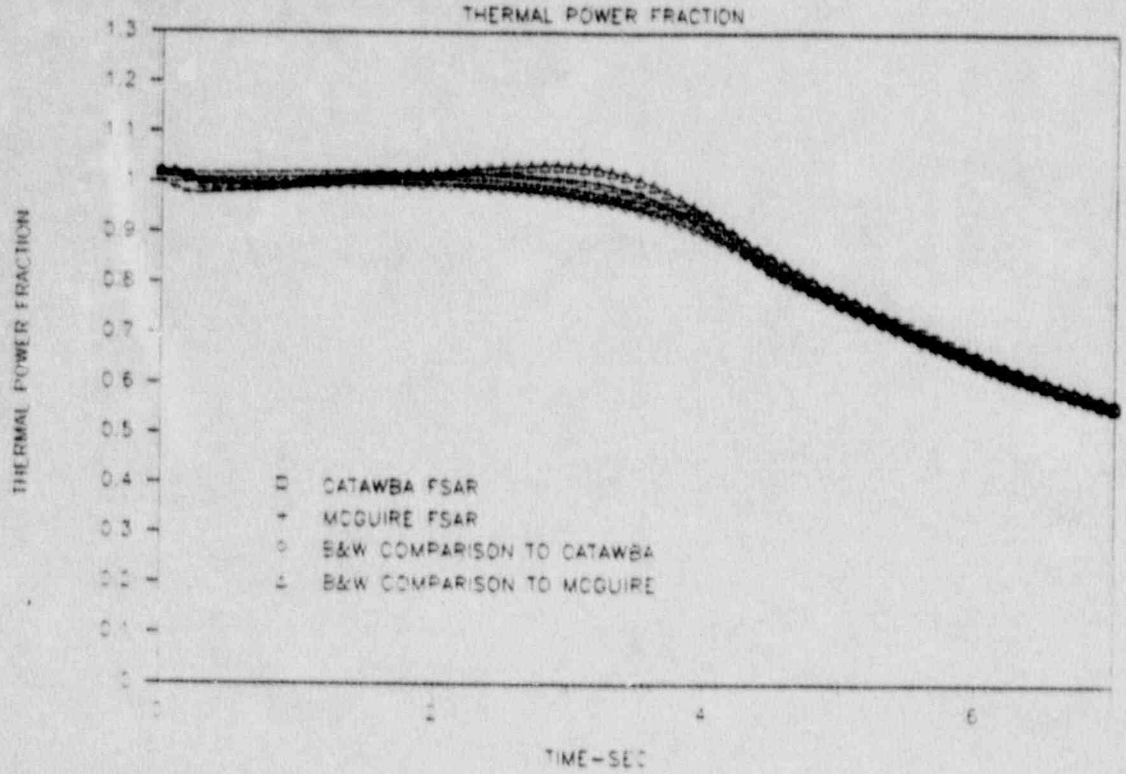


FIGURE 7.4.14. LOCKED ROTOR

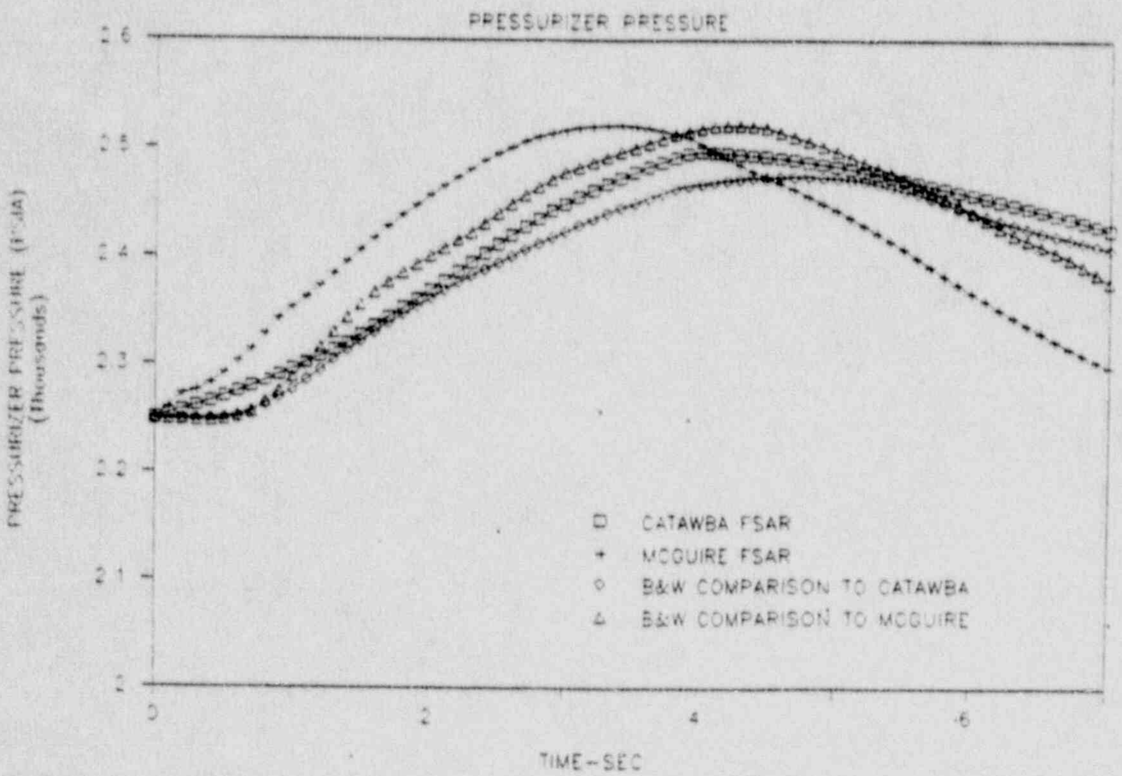


FIGURE 7.4.15. LOCKED ROTOR

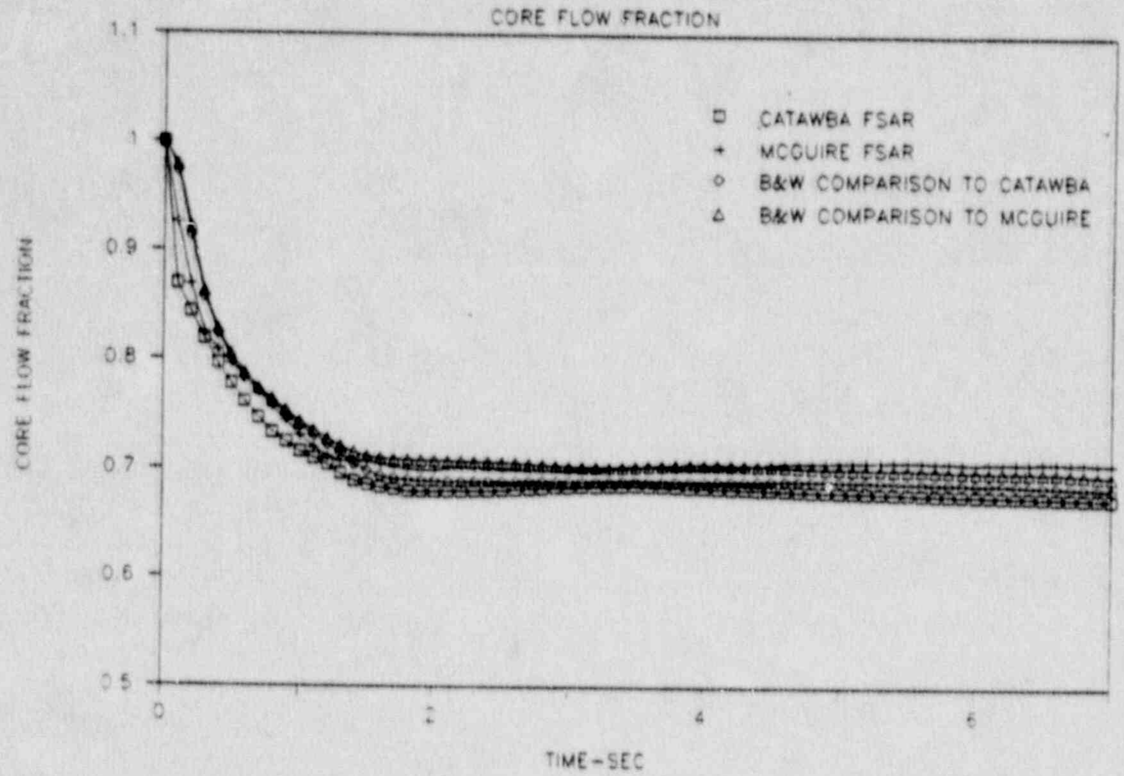


FIGURE 7.4.16. LOCKED ROTOR

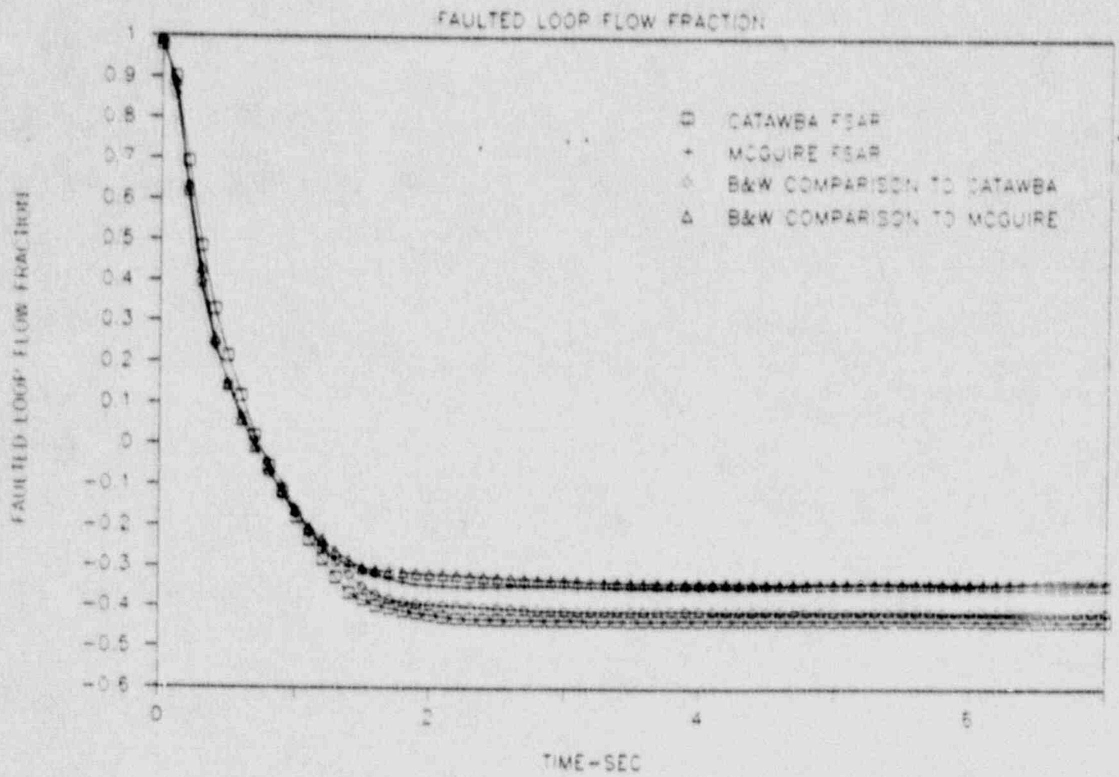


FIGURE 7.4.17. LOCKED ROTOR

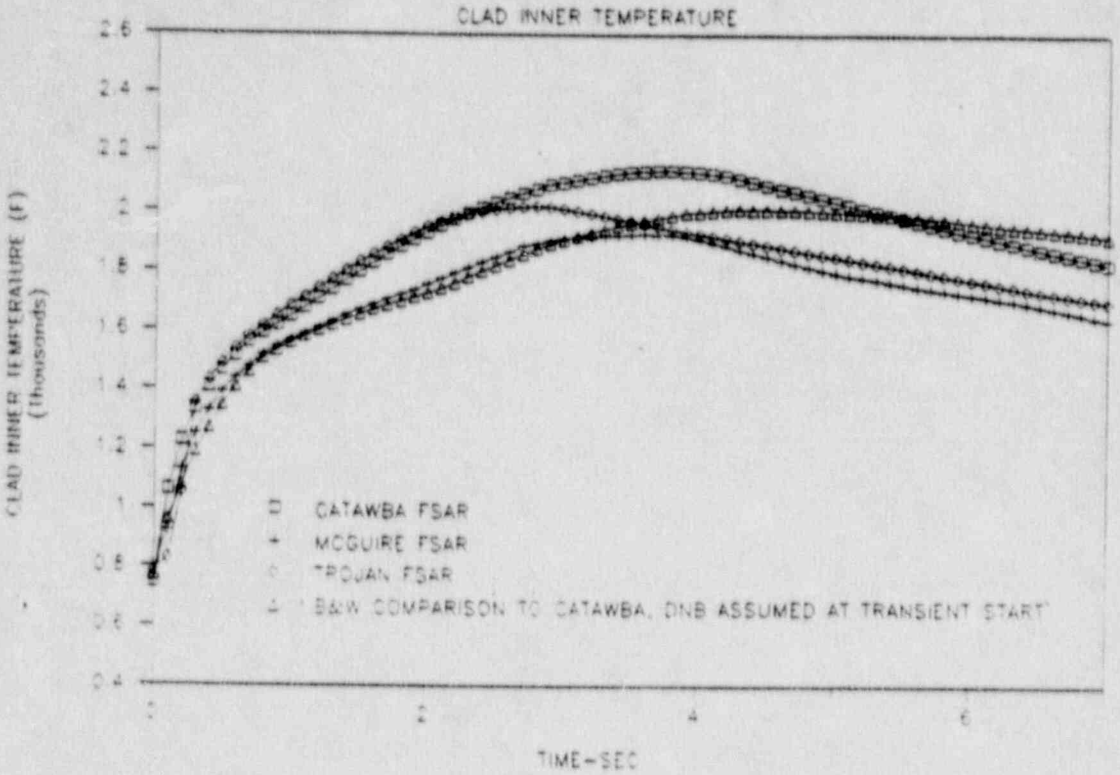


FIGURE 7.4.18. LOCKED ROTOR

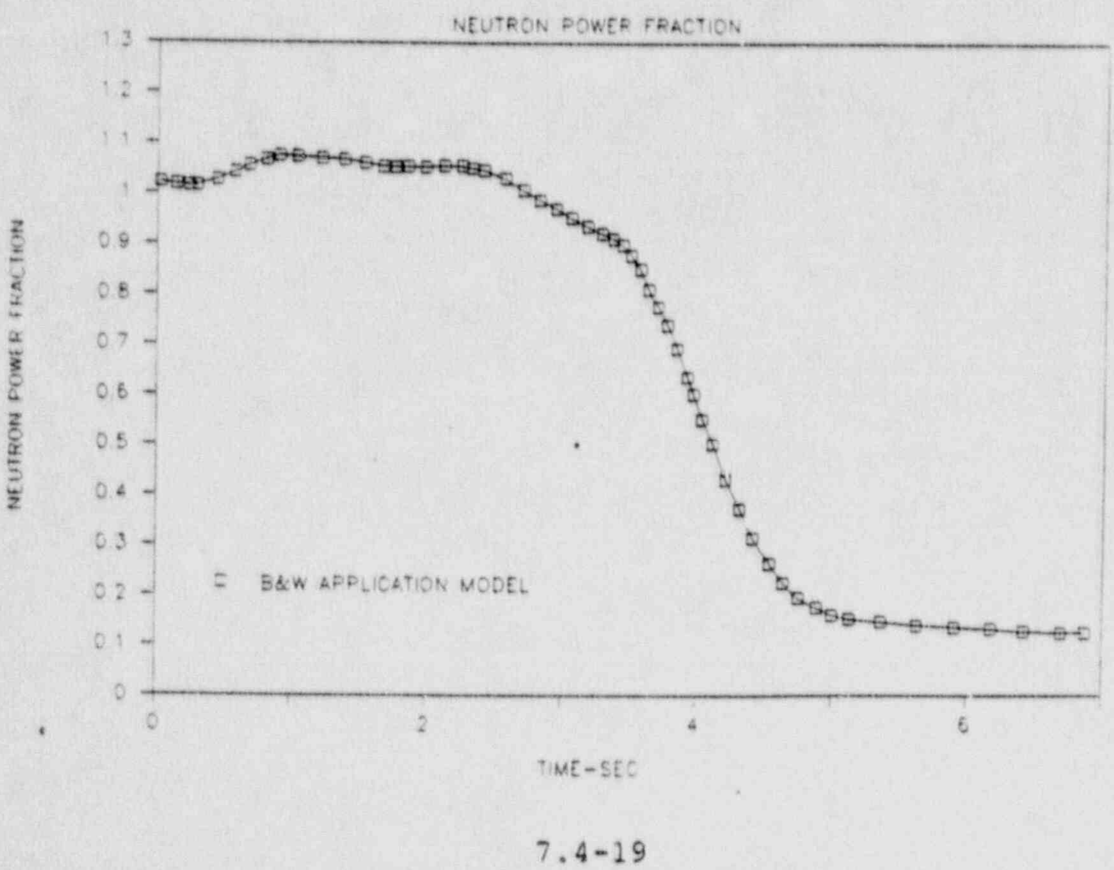


FIGURE 7.4.19. LOCKED ROTOR

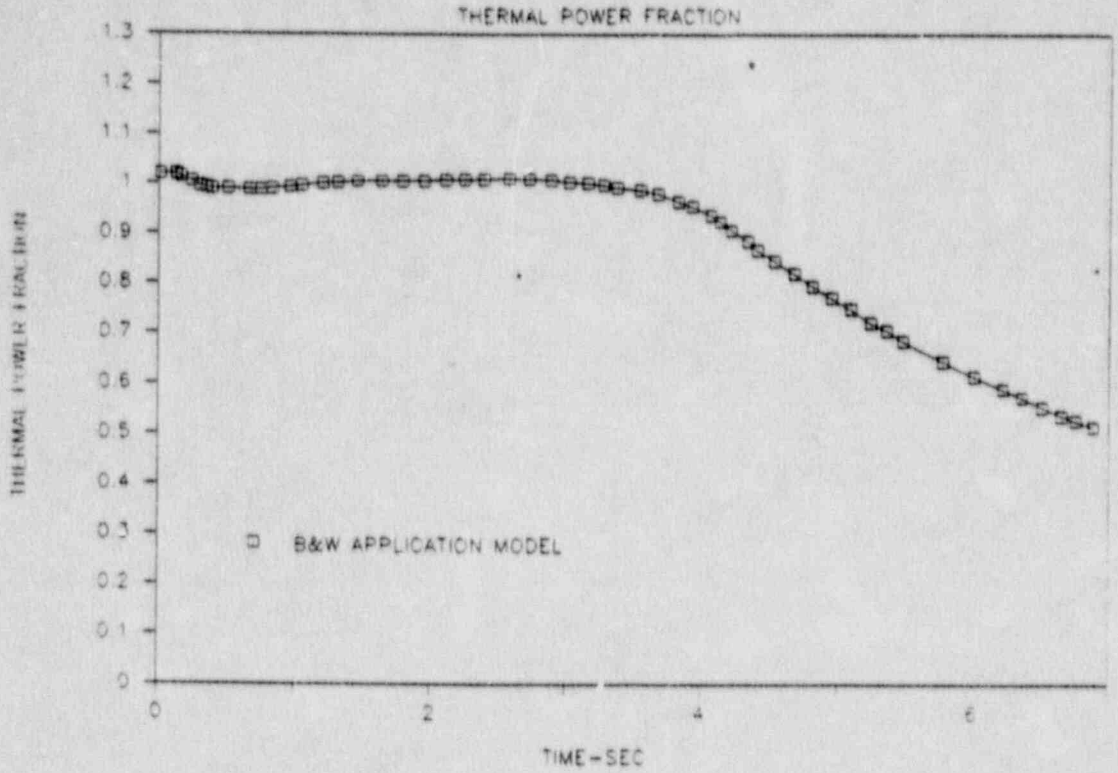


FIGURE 7.4.20. LOCKED ROTOR

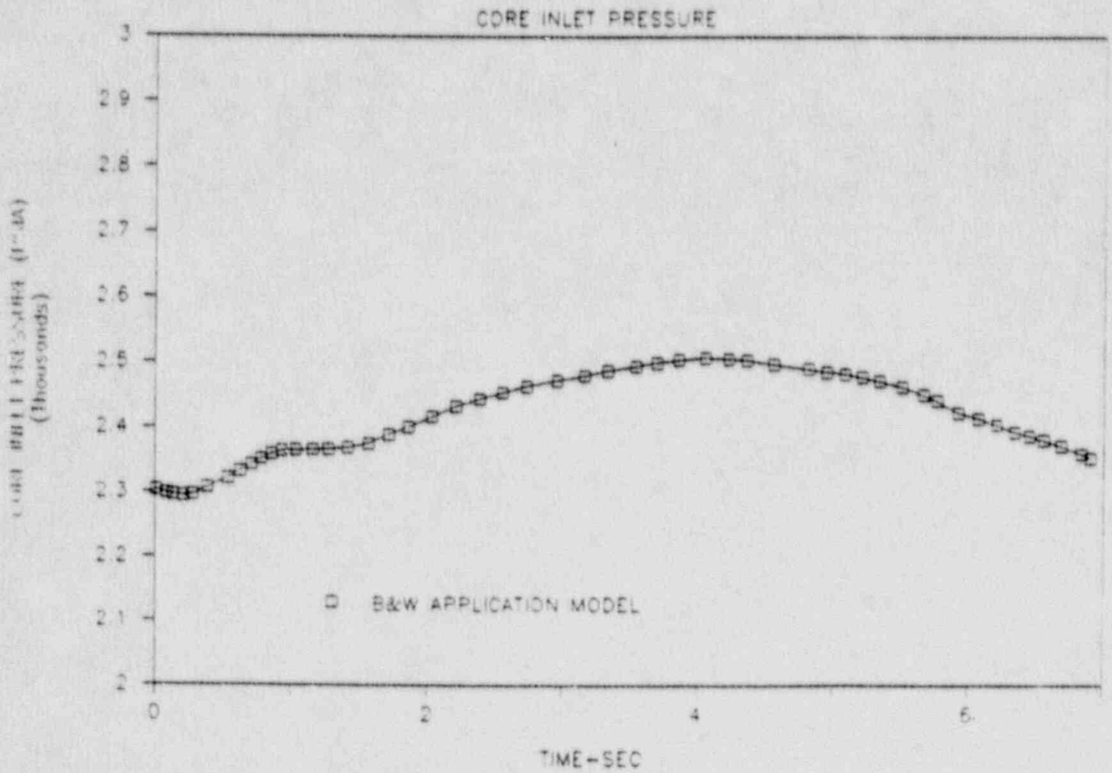


FIGURE 7.4.21. LOCKED ROTOR

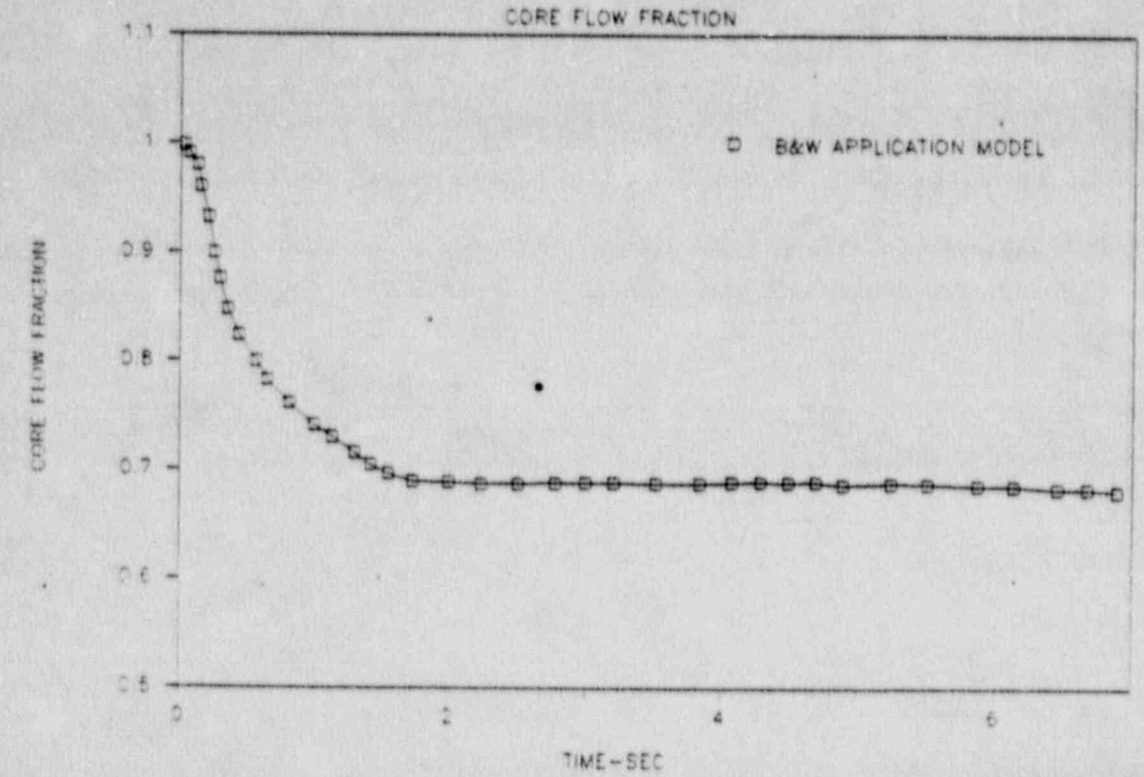
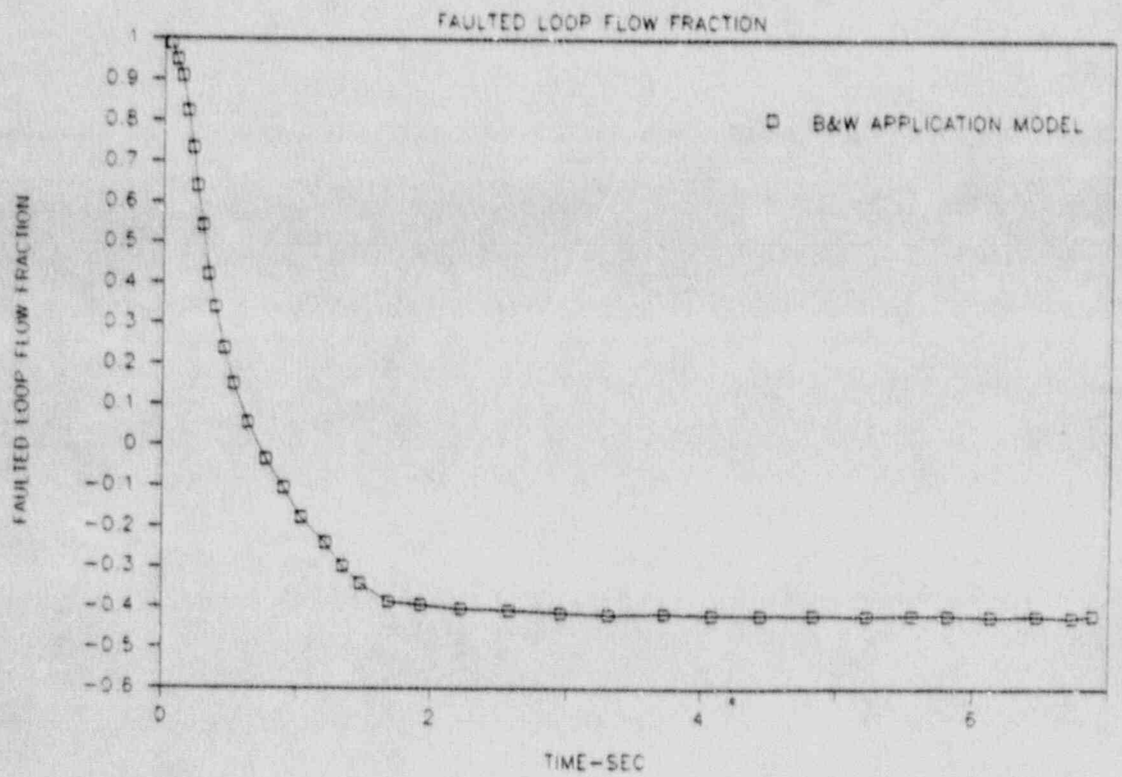


FIGURE 7.4.22. LOCKED ROTOR



7.5. Steam System Piping Failure

7.5.1. Identification of Causes and Accident Description

The steam release arising from a rupture of a main steam line would result in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the RCS causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity. If the most reactive rod cluster control assembly (RCCA) is assumed stuck in its fully withdrawn position after reactor trip, there is an increased possibility that the core will become critical and return to power. A return to power following a steam line rupture is a potential problem mainly because of the high power peaking factors which exist assuming the most reactive RCCA to be stuck in its fully withdrawn position. The core is ultimately shut down by the boric acid injection delivered by the safety injection system.

The analysis of a main steam line rupture is performed to demonstrate that the following criteria are satisfied:

> Assuming a stuck RCCA with or without offsite power, and assuming a single failure in the engineered safety features, the core remains in place and intact. Radiation doses do not exceed the guidelines of 10CFR100.
>
> Although DNB and possible clad perforation following a steam pipe rupture are not necessarily unacceptable, the analysis shown in the Catawba SAR, in fact, shows that no DNB occurs for any rupture assuming the most reactive assembly stuck in its fully withdrawn position.

A major steam line rupture is classified as an ANS Condition IV event. See Section 4.0 for a discussion of Condition IV events.

The major rupture of a steam line is the most limiting cooldown transient and is analyzed at zero power with no decay heat. Decay heat would retard the cooldown thereby reducing the return to power. A detailed analysis of this transient with the most limiting break size, a double ended rupture, is presented here.

The following functions provide the protection for a steam line rupture:

1. Safety injection system actuation from any of the following:

   a. Two-out-of-three low steam line pressure signals in any one loop.

   b. Two-out-of-four low pressurizer pressure signals.

   c. Two-out-of-four high containment pressure signals.

2. The overpower reactor trips (neutron flux and ΔT) and the reactor trip occurring in conjunction with receipt of the safety injection signal.

3. Redundant isolation of the main feedwater lines.

   Sustained high feedwater flow would cause additional cooldown. Therefore, in addition to the normal control action which will close the main feedwater valves a feedwater isolation signal will rapidly close all feedwater control valves and back up feedwater isolation valves, trip the main feedwater pumps, and close the feedwater pump discharge valves for the Catawba plant.

4. Trip of the fast acting steam line stop valves (designed to close in less than 5 seconds) on:

a. Two-out-of-three low steam line pressure signals in any one loop.

b. Two-out-of-three high-high containment pressure signals.

c. Two-out-of-three high negative steam line pressure rate signals in any one loop (used only during cooldown and heatup operations).

Fast-acting isolation valves are provided in each steam line; these valves will fully close within 10 seconds of a large break in the steam line. For breaks downsteam of the isolation valves, closure of all valves would completely terminate the blowdown. For any break, in any location, no more than one steam generator would experience an uncontrolled blowdown even if one of the isolation valves fails to close.

Steam flow is measured by monitoring dynamic head in nozzles located in the throat of the steam generator. The effective throat area of the nozzles is 1.4 square feet, which is considerably less than the main steam pipe area; thus, the nozzles also serve to limit the maximum steam flow for a break at any location.

7.5.2. Analysis of Effects and Consequences

7.5.2.1. Method of Analysis

The analysis of the steam pipe rupture has been performed to determine:

1. The core heat flux and RCS temperature and pressure resulting from the cooldown following the steam line break. The LOFTRAN CODE (Reference 7) has been used for the Catawba SAR results.

2. The thermal and hydraulic behavior of the core following a steam line break. A detailed thermal and hydraulic digital-computer code, THINC, was used for the Catawba SAR to determine if DNB occurs for the core conditions computed in item 1 above.

The B&W model analysis was done using the RELAP5/MOD2 computer code (Reference 6) for the system parameters. A combination of codes will be used for the thermal/hydraulic analysis. These codes will consider both the detailed nuclear feedback associated with the stuck rod channel and the thermal/hydraulic effects in the stuck rod channel.

7.5.2.2. Initial Conditions

Studies have been performed to determine the sensitivity of steam line break results to various assumptions. Based upon these studies, the following conditions were assumed to exist at the time of the main steam line break accident:

1. End-of-life shutdown margin at no-load, equilibrium xenon conditions, and the most reactive RCCA stuck in its fully withdrawn position. Operation of the control rod banks during core burnup is restricted in such a way that addition of positive reactivity in a steam line break accident will not lead to a more adverse condition than the case analyzed.

2. A negative moderator coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in the fully withdrawn position. The variation of the coefficient with temperature and pressure has been included. The K_{eff} versus temperature at 1000 psi corresponding to the negative moderator temperature coefficient used for the Catawba SAR results is shown in Figure 7.5.1. The effect of power generation in the core on overall reactivity is shown in Figure 7.5.2 for the Catawba SAR results.

The core properties associated with the sector nearest the affected steam generator and those associated with the remaining sector were conservatively combined to obtain average core properties for reactivity feedback calculations for the Catawba SAR. Further, it was conservatively assumed that the core power distribution was uniform. These two conditions cause underpredictions of the reactivity feedback in the high power region near the stuck rod. To verify the conservatism of this method, the reactivity as well as the power distribution were checked for the limiting statepoints for the cases analyzed.

This core analysis considered the Doppler reactivity from the high fuel temperature near the stuck RCCA, moderator feedback from the high water enthalpy near the stuck RCCA, power redistribution and nonuniform core inlet temperature effects. For cases in which steam generation occurs in the high flux regions of the core, the effect of void formation was also included. It was determined that the reactivity employed in the kinetics analysis was always larger than the reactivity calculated including the above local effects for the statepoints. These results verify conservatism (i.e., underprediction of negative reactivity feedback from power generation).

3. Minimum capability for injection of boric acid (2,000 ppm from the RWST) solution corresponding to the most restrictive single failure in the safety injection system was used in the Catawba SAR analysis. The emergency core cooling system consists of three systems: (1) the passive accumulators, (2) the residual heat removal system, and (3) the safety injection system. Only the safety injection system is modeled for the steam line break accident analysis. The flow corresponds to that delivered by one charging pump delivering its full flow to the cold leg header. No credit has been taken for the low concentration

borated water, which must be swept from the lines downsteam of the RWST prior to the delivery of high concentration boric acid to the reactor coolant loops.

For the cases where offsite power is assumed, the sequence of events in the safety injection system is the following. After the generation of the safety injection signal (appropriate delays for instrumentation, logic, and signal transport included), the appropriate valves begin to operate and the high head safety injection pump starts. In 10 seconds, the valves are assumed to be in their final position and the pump is assumed to be at full speed. The volume containing the low concentration borated water is swept before the 2,000 ppm reaches the core. This delay, described above, is inherently included in the modeling.

In cases where offsite power is not available, an additional 10 second delay is assumed to start the diesels and to load the necessary safety injection equipment onto them.

4. Design value of the steam generator heat transfer coefficient including allowance for fouling factor was used for the Catawba SAR analysis.

5. Since the steam generators are provided with integral flow restrictors with a 1.4 square foot throat area, any rupture with a break area greater than 1.4 square feet, regardless of location, would have the same effect on the NSSS as the 1.4 square foot break. The case considered in determining the core power and RCS transients was the complete severance of a pipe, with the plant initially at no-load conditions, full reactor coolant flow with offsite power available.

6. Power peaking factors corresponding to one stuck RCCA and nonuniform core inlet coolant temperatures determined at end of core life were used for the Catawba SAR analysis. The

coldest core inlet temperatures are assumed to occur in the sector with the stuck rod. The power peaking factors account for the effect of the local void region of the struck control assembly during the return to power phase following the steam line break. This void in conjunction with the large negative moderator coefficient partially offsets the effect of the stuck assembly. The power peaking factors depend upon the core power, temperature, pressure, and flow.

Initial hot standby conditions were assumed at time zero since this represents the most pessimistic initial condition. Should the reactor be just critical or operating at power at the time of a steam line break, the reactor will be tripped by the normal overpower protection system when power level reaches a trip point. Following a trip at power, the RCS contains more stored energy than at no-load, the average coolant temperature is higher than at no-load and there is appreciable energy stored in the fuel. Thus, the additional stored energy is removed via the cooldown caused by the steam line break before the no-load conditions of RCS temperature and shutdown margin assumed in the analyses are reached. After the additional stored energy has been removed, the cooldown and reactivity insertions proceed in the same manner as in the analysis which assumed no-load condition at time zero.

7. In computing the steam flow during a steam line break, the Moody Curve for $f(L/D) = 0$ is used for the Catawba SAR results.

8. The upper head injection (UHI) is simulated for the Catawba SAR results. The actuation pressure for the UHI is near the saturation pressure for the inactive coolant in the upper head. The insurge of cold UHI water keeps this inactive coolant from flashing and from retarding the pressure

decrease. The effect of UHI is a faster pressure decrease which in turn permits more safety injection flow into the core. These effects are relatively small and results are not significantly affected.

7.5.3. Results

7.5.3.1. Safety Analysis Report Data

Figures 7.5.3 through 7.5.11 show the following SAR steam line break data:

1. Trojan (old) - Figure 7.5.3 - These results are for 20,000 ppm boron injection.

2. Trojan (new) - Figure 7.5.4 - These data are presented because the curves show better detail and represent 2000 ppm boron injection.

3. RESAR - Figure 7.5.5 - These data are presented for consistency of data presentation, however the results are for 20,000 ppm boron injection.

4. Catawba - Figures 7.5.6,7,8 - These data are presented because more parameters are shown for Catawba than for RESAR or Trojan.

5. McGuire - Figures 7.5.9,10,11 - These data are presented to allow comparison between Catawba and McGuire results. The key steam line break events are indicated on these figures.

Digitized replots of the SAR results are not presented because of the compressed time scaling of the SAR curves. A single RELAP5 analysis is presented because the steam line break analysis is heavily plant specific in auxiliary feedwater flow and boron injection assumptions. This will be discussed in detail.

7.5.3.2. Analysis Comparison

The results presented are a conservative indication of the events that would occur assuming a steam line rupture since it is postulated that all of the conditions described previously occur simultaneously.

Core Power and Reactor Coolant System Transient

Figures 7.5.12 through 7.5.27 show the RCS transient and core heat flux following a main steam line rupture (complete severance of a pipe) at initial no-load condition.

Offsite power is assumed available so that full reactor coolant flow exists. The transient shown assumes an uncontrolled steam release from only one steam generator. Should the core be critical at near zero power when the rupture occurs the initiation of safety injection by low steam line pressure will trip the reactor. Steam release from more than one steam generator will be prevented by automatic trip of the fast acting isolation valves in the steam lines by low steam line pressure signals, high-high containment pressure signals, or high negative steam line pressure rate signals. Even with the failure of one valve, release is limited to no more than 10 seconds for the other steam generators while the one generator blows down. The steam line stop valves are designed to be fully closed in less than 5 seconds from receipt of a closure signal.

As shown in Figure 7.5.19 the core attains criticality with the RCCAs inserted (with the design shutdown assuming one stuck RCCA) shortly after boron solution at 2,000 ppm enters the RCS. The continued addition of boron results in a peak core power significantly lower than the nominal full power value. The calculation assumes the boric acid is mixed with, and diluted by, the water flowing in the RCS prior to entering the reactor core. The concentration after mixing depends upon the relative flow rates in the RCS and in the safety injection system. The variation of mass flow rate in the RCS due to water density changes is included in the calculation as is the variation of flow rate in the safety injection system due to changes in the RCS pressure. The safety injection system flow calculation includes the line losses in the system as well as the pump head curve.

It should be noted that following a steam line break only one steam generator blows down completely. Thus, the remaining steam generators are still available for dissipation of decay heat after the initial transient is over. In case of loss of offsite power this heat is removed to the atmosphere via the steam line safety valves.

<u>Margin to Critical Heat Flux</u>

A DNB analysis was performed for the Catawba SAR results and it was found a minimum DNBR greater than 1.3 exists for both the with and without offsite power cases.

Figures 7.5.12 through 7.5.27 show the RELAP5 steam line break results. The following is a detailed discussion of these results:

1. Figure 7.5.12 - Neutron Power - The 25 second time of return to critical predicted by RELAP5 is in agreement with all five sets of SAR data. The maximum return to critical power level of 18 percent is consistent with the 16 percent value shown for Trojan (new), Catawba and McGuire. Trojan (old) and RESAR show a slightly lower return to critical power level, but they are based on 20,000 ppm boron injection instead of 2000 ppm boron injection used in RELAP5 and later SAR analysis. The RELAP5 single RSG inventory depletes at 170 seconds because the auxiliary feedwater was terminated at 170 seconds in this run. The 170-second

inventory depletion is consistent with the Catawba SAR results. As will be discussed later, the time of dryout for the single RSG is dependent on the magnitude of the additional fluid added to the RSG. The boron addition controls the slope of the neutron power curve after the initial turnover due to the Doppler power feedback. The results shown are for 75 seconds of boron addition (approximately the volume of the boron injection tank-BIT). Also shown is the neutron power decrease that would occur if the boron addition had been continued as shown in Figure 7.5.20.

2. Figure 7.5.13 - Thermal Power - The important sequence of events are the same as for neutron power, except for the initial rise at 5 seconds caused by lower water temperatures and not heat in the fuel rod. This initial thermal power rise is in agreement with Catawba and McGuire.

3. Figure 7.5.14 - Pressurizer Pressure - The pressurizer pressure is consistent with SAR data because they all drop to about 1000 psi and hold that value until single RSG depletion occurs. The pressurizer is on the triple loop in this model. The location of the pressurizer does not affect the analysis because pressure equalization occurs due to the crossflow modeling at the core inlet and outlet plenum. Very little fluid flow exists at the crossflow junctions but it is sufficient to assure the pressurizer pressure is representative of the system pressure and not an individual loop.

4. Figure 7.5.15 - Pressure Water Volume - The pressurizer water volume is in agreement with SAR data because RELAP5 predicts the pressurizer emptying in about 18 seconds and the SAR data is 14 to 20 seconds. The RELAP5 pressurizer refill occurs when the single RSG inventory is depleted.

5. Figure 7.5.16 - Single Loop RV Inlet Temperature - This curve shows the single loop temperature decreases until single RSG dryout occurs, then the temperature rises to the total systems temperature value. These results are consistent with SAR results.

6. Figure 7.5.17 - Triple Loop RV Inlet Temperature - This curve shows the triple loop temperature decreases at the same rate as the single loop until the triple RSG is isolated at 10 seconds. The temperature continues to decrease, at a slower rate, until the dryout occurs in the single RSG, then the temperature levels at the total system temperature value. These results are consistent with SAR results.

7. Figure 7.5.18 - Core Average Temperature - This curve shows an average of the triple loop and single loop results of Figures 7.5.16 and 7.5.17.

8. Figure 7.5.19 - Reactivity (%Δk/k) - This curve shows the initial -1.6% Δk/k shutdown value and then a steady rise as the EOL negative moderator coefficient produces a positive reactivity addition as a result of the decreasing fluid temperatures. The return to critical at 25 seconds is in agreement with all the SAR values. The reactivity turns over at approximately 70 seconds due to negative reactivity addition produced by the negative Doppler power feedback and the rising neutron power (fuel temperature). Also, negative reactivity is being added by the boron that is being added to the core (see Figure 7.5.20) by the safety injection system. The reactor returns to subcritical when the system fluid temperatures rise due to the single RSG dryout.

9. Figure 7.5.20 - Boron (ppm) - The boron appears in the reactor core at 29 seconds due to the low pressurizer pressure trip and delays associated with valve openings.

The boron (ppm) increase is consistent with the SAR results up to 100 seconds; however, the results past 100 seconds are different because the boron addition was terminated 75 seconds after the start. Also shown is the linear increase in boron concentration that would be associated with continued safety injection system operation. These results are indicative of the need for very accurate plant specific safety injection system modeling.

10. Figure 7.5.21 - Single RSG Feedwater Flow Fraction - The 110% feedwater flow that is terminated at 10 seconds is the same as shown by the Catawba and McGuire SAR. For the single RSG, 21% auxiliary feedwater was added until 170 seconds, thus the single RSG dryout occurred at 170 seconds. If no auxiliary feedwater is added the single RSG will dry out at approximately 100 seconds.

11. Figure 7.5.22 - Triple RSG Feedwater Flow Fraction - The 110% feedwater flow that is terminated at 10 seconds is the same as shown by the Catawba and McGuire SAR. For the triple RSG 7% auxiliary feedwater was added indefinitely.

12. Figure 7.5.23 - Single RSG Steam Flow Fraction - This curve shows the steam flow through the 1.4 ft^2 flow restrictor located at the top of the RSG. SAR results are also presented. The reason the RELAP5 results are higher is that RELAP5 predicts moisture in the steam dome early in the blowdown due to the sudden pressure drop. The choked flowrate calculated by RELAP5 is higher than the SAR flowrate because the SAR flowrate is based on saturated (no moisture) conditions. This subject will be discussed further later in this report.

13. Figure 7.5.24 - Triple RSG Steam Flow Fraction - These results are the same as the single RSG steam flow fraction until the triple RSG is isolated at 10 seconds, then the steam flow goes to zero.

14. Figure 7.5.25 - Single RSG Steam Pressure - The steam pressure results are consistent with SAR results because the pressure drops to approximately 250 psi and holds there until single RSG dryout occurs.

15. Figure 7.5.26 - Triple RSG Steam Pressure - The steam pressure results are the same as for the single RSG until triple RSG isolation at 10 seconds. The pressure then rises and holds at a constant value, as shown by the SAR results.

16. Figure 7.5.27 - Reactor Vessel Flow Fraction - The reactor vessel flow is increasing due to the colder fluid going through the pumps. The SAR results are different because they are all based on constant 100% flow.

All of the curves shown are consistent with the Catawba/McGuire results. This implies the difference between the Type 51 RSG of the Trojan plant modeled in the present RELAP5 model and the Catawba/McGuire preheat RSG are not significant relative to safety analysis transients.

Figures 7.5.28 through 7.5.35 show single RSG liquid fraction and pressure results. The following is a discussion of these results:

1. Figure 7.5.28 - Flow Restrictor Mass Flow Rate - This is the same curve as shown in Figure 7.5.23.

2. Figure 7.5.29 - Flow Restrictor Liquid Fraction - This curve shows the liquid fraction that causes RELAP5 to predict a higher steam flowrate through the flow restrictor than the SAR results that are based on saturated steam conditions (no moisture). The maximum value is 13% and drops to zero at 40 seconds.

3. Figure 7.5.30 - Steam Line Liquid Fraction - This curve shows the liquid fraction in the steam line is the same shape as in the flow restrictor but the maximum value is 2.2%.

4. Figure 7.5.31 - Steam Line Pressure - This curve shows the steam line pressure drops to 100 psi at 50 seconds where it flattens due to choked flow at the steam line isolation valve (3.77 ft2). The flow choking in the steam line has no effect on flow through the flow restrictor because the steam linc is a downstream condition.

5. Figure 7.5.32 - Steam Dome Liquid Fraction - The steam dome liquid fraction is identical to the flow restrictor liquid fraction (Figure 7.5.29).

6. Figure 7.5.33 - Steam Dome Pressure - The steam dome pressure decreases to 250 psi at 50 seconds as expected and is unaffected by the liquid fraction changes.

7. Figure 7.5.34 - Liquid Below Separator Liquid Fraction- This curve shows the changes going on in the steam dome have little or no effect on the area of the RSG below the separator where the heat transfer from the primary system is taking place.

8. Figure 7.5.35 - Tube Nest Liquid Fraction - This curve confirms that the changes in the steam dome have little or no effect on the heat transfer from the primary system.

Figures 7.5.28 through 7.5.35 indicate the liquid fraction in the steam dome is produced by the rapid pressure drop and not flashing from lower in the RSG. Due to this, the heat removal from the primary system has the same conservatism in RELAP5 as does the SAR results, even though the RELAP5 steam flow is based on nonsaturated conditions.

7.5.3.4. B&W Application Model

A B&W application model analysis was not performed because of the plant specific nature of the safety injection system boron injection and the feedwater flow addition. Any analysis would be expected to be similar to the analysis of Figures 7.5.12 through 7.5.27 up to the time of boron injection.

7.5.4. Conclusions

The RELAP5 model has shown agreement with SAR steam line break results for several different plants and is therefore a valid method for the analysis of any specific plant when the appropriate geometry and plant parameters are modeled.

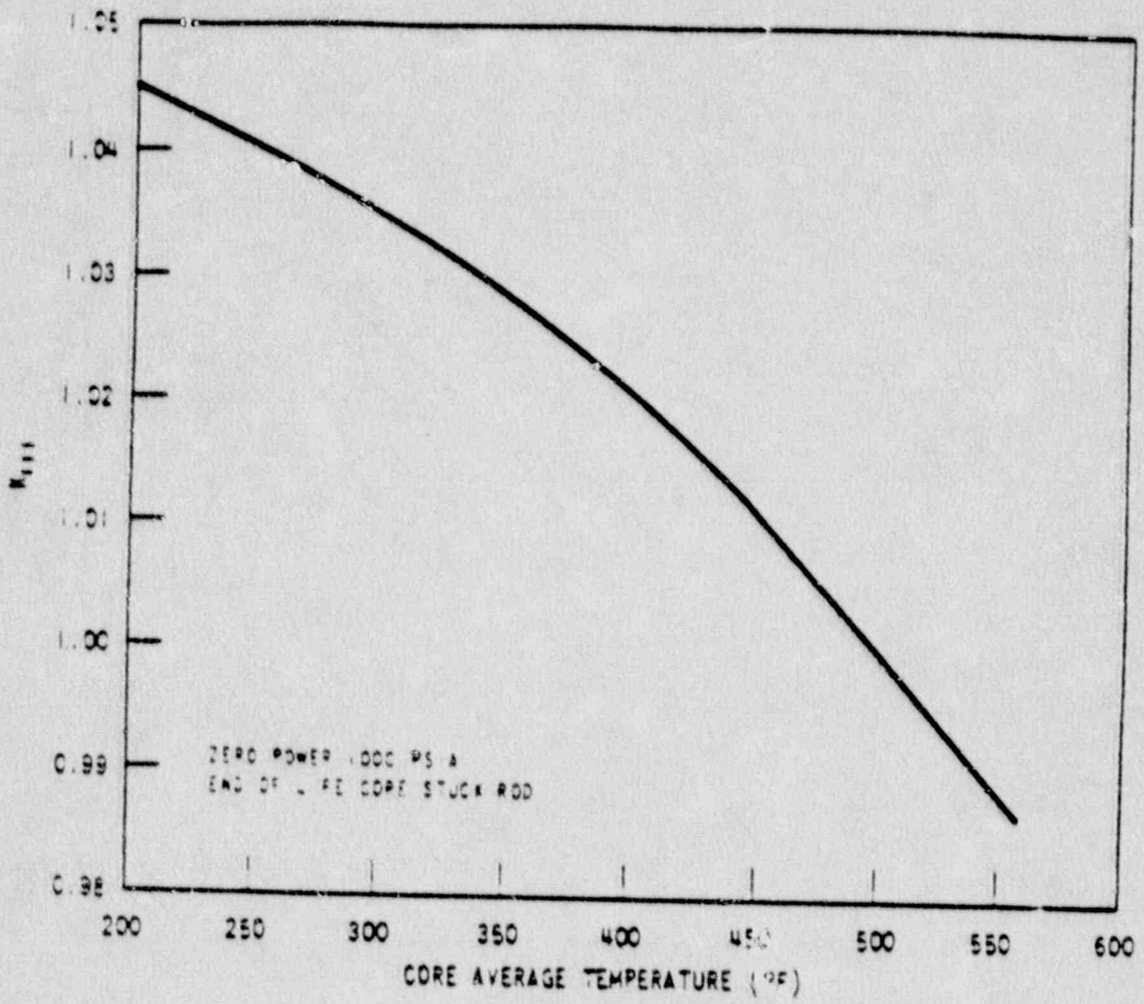


Figure 7.5.1. Catawba SAR Figure 15.1.4-1 Revision 3
Keff versus Temperature

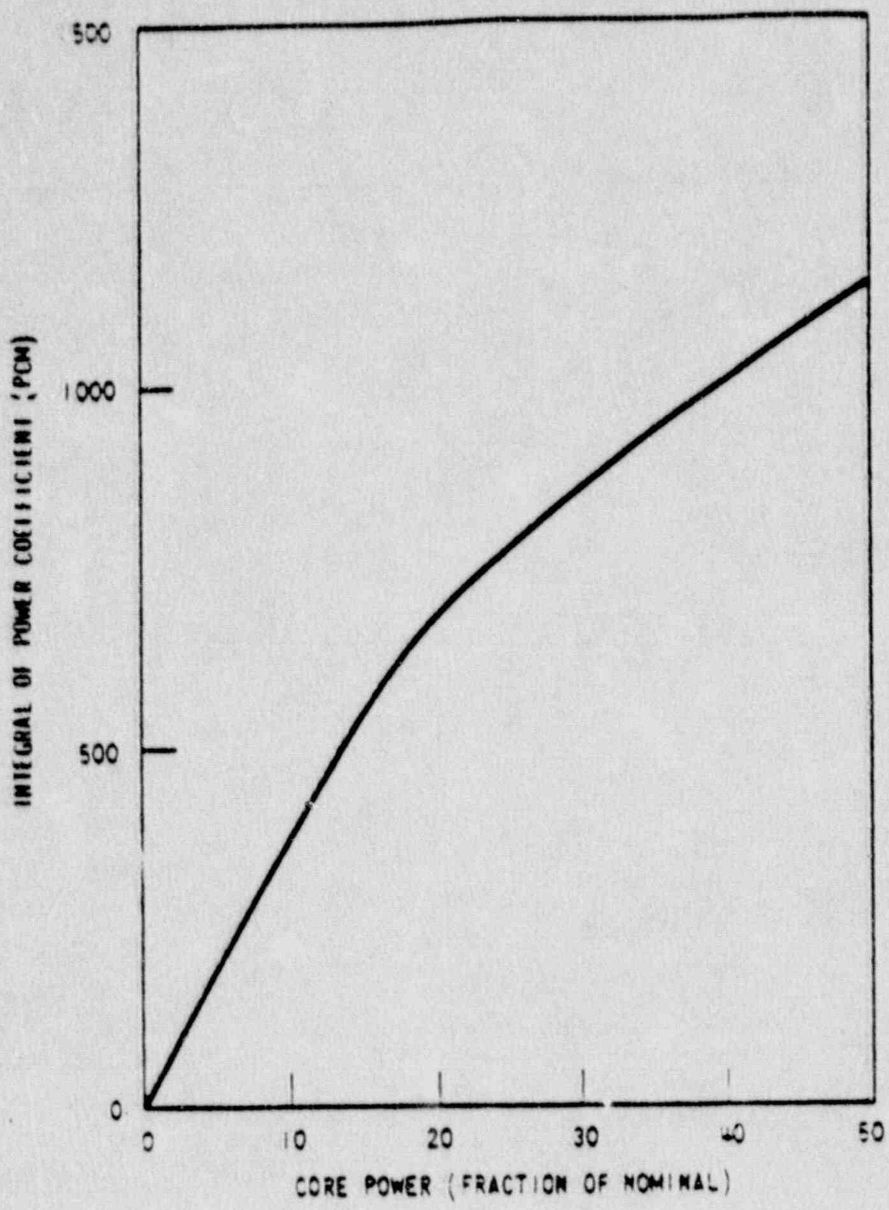


Figure 7.5.2. Catawba SAR Figure 15.1.5-1 Revision 3
Doppler Power Feedback

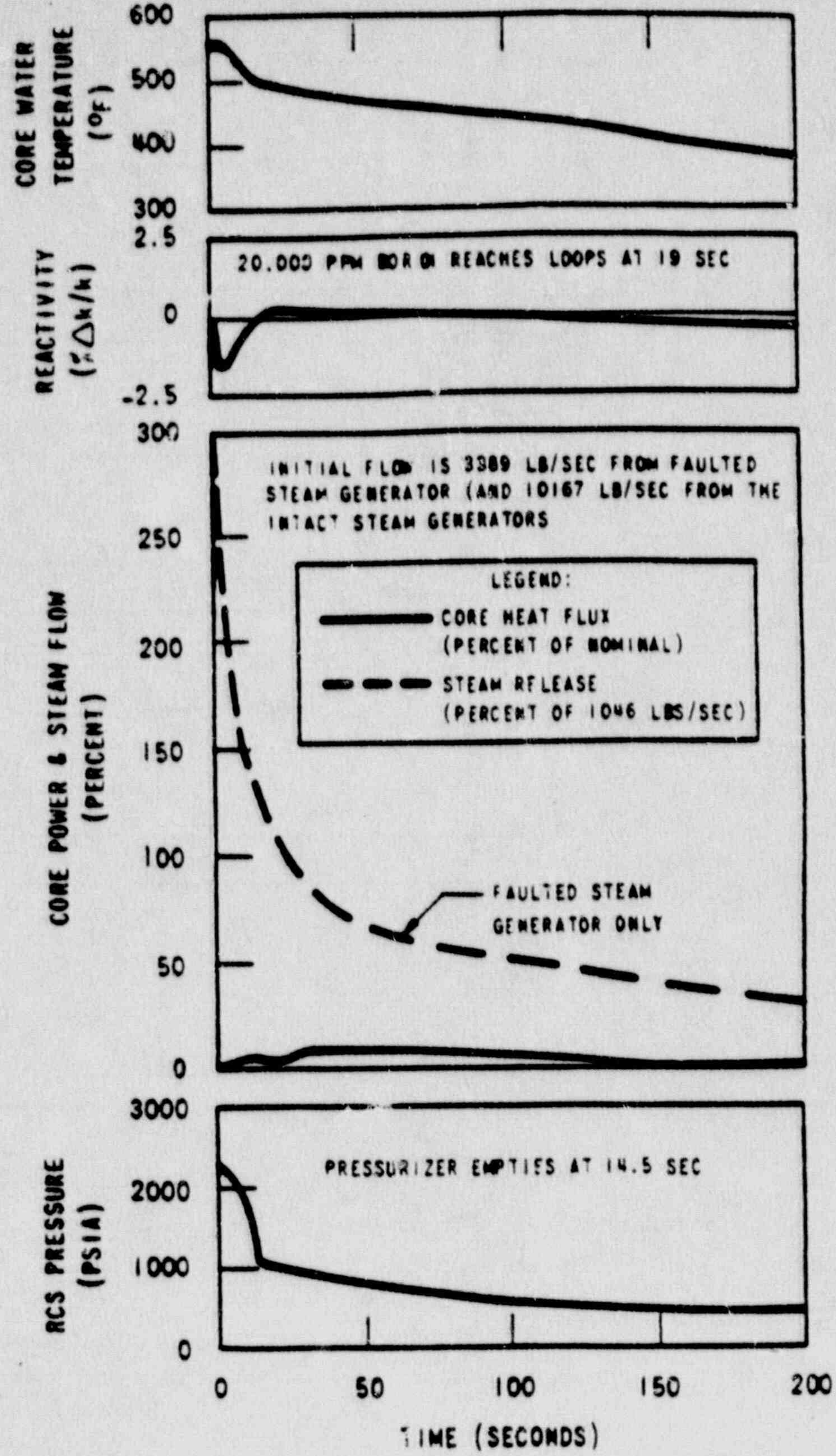


Figure 7.5.3. Trojan SAR Figure 15.1-14 Transient Response to Steam Line Break Downstream of Flow Measuring Nozzle with Safety Injection and Off-Site Power (case a)

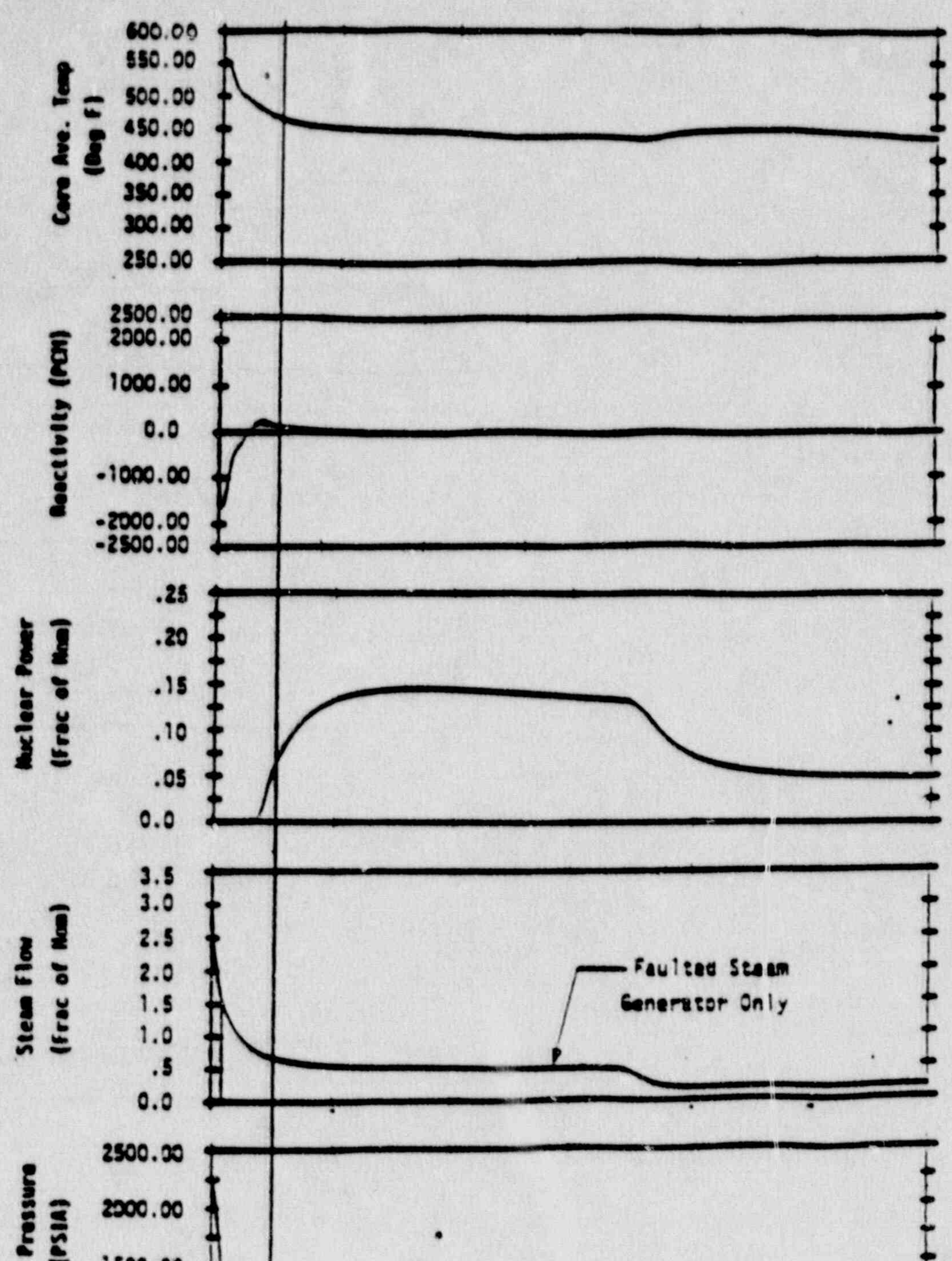


Figure 7.5.4 Trojan(new) SAR Figure 15.1-14 Amendment 3 (July 1985) Transient Response to Steam Line Break Downstream of Flow Measuring Nozzle with Safety Injection and Offsite Power (case a)

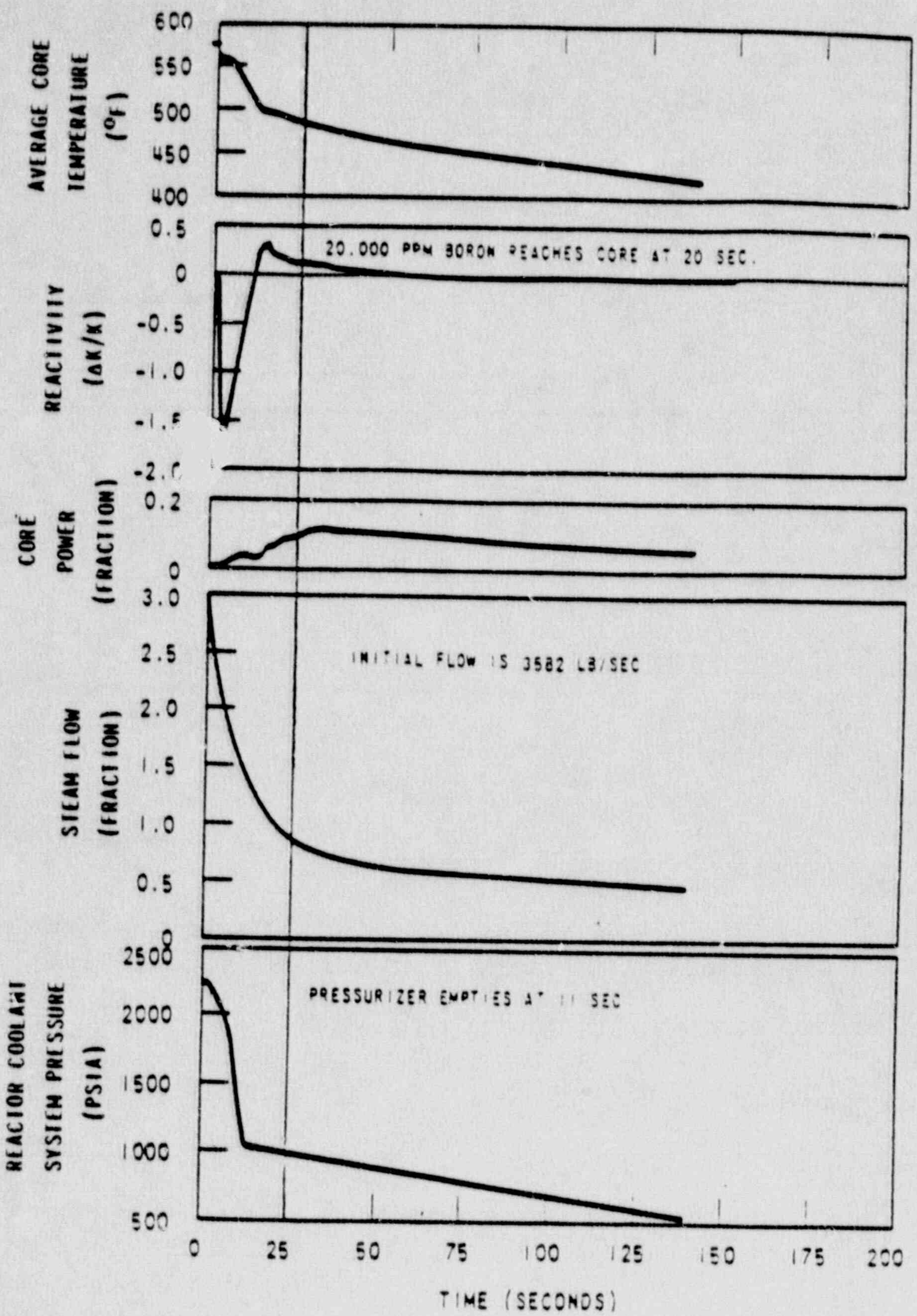


Figure 7.5.5. RESAR Figure 15.4-66 Transient Response to Steam Line Break with Safety Injection and Offsite Power (case a)

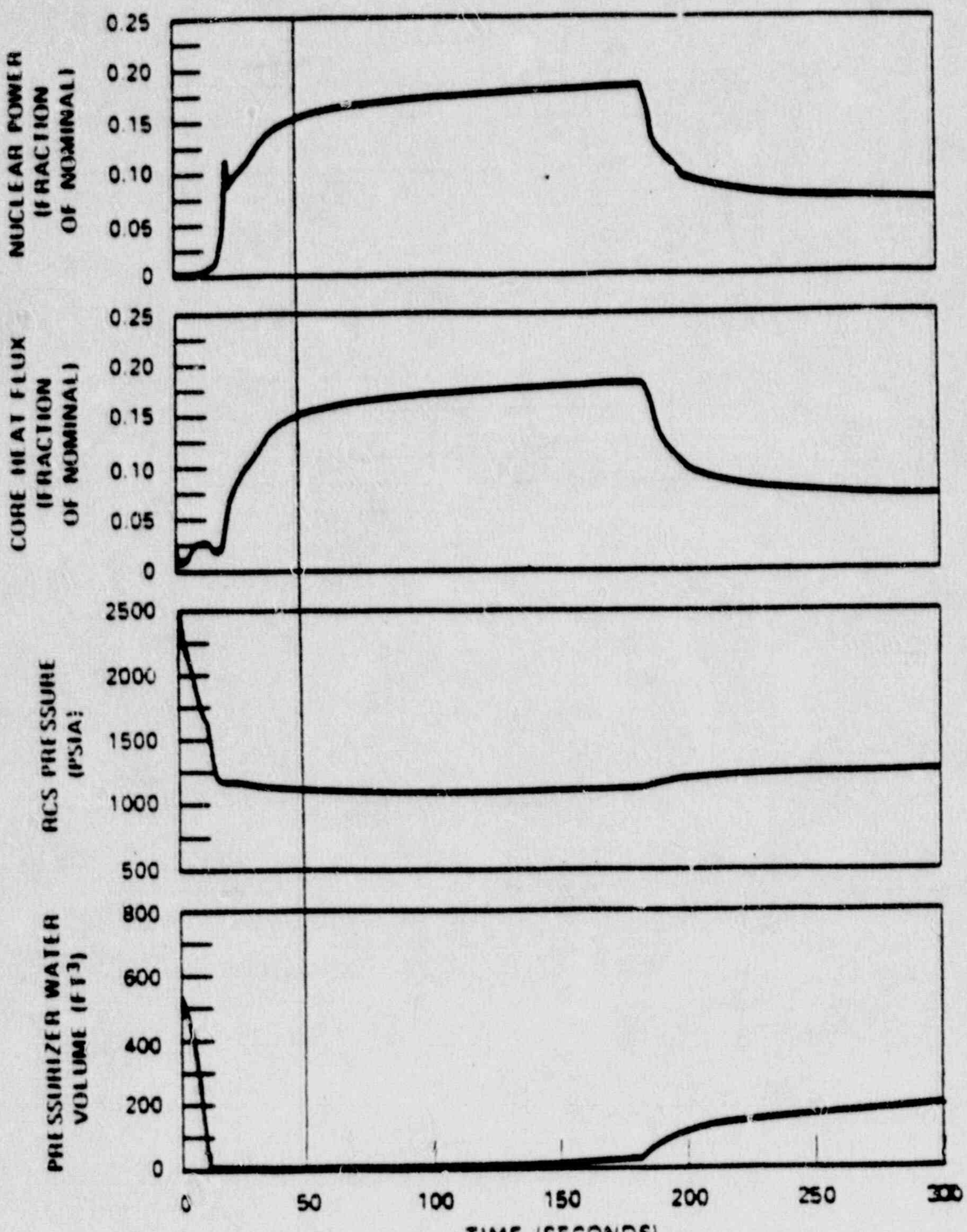


Figure 7.5.6. Catawba SAR F'gure 15.1.5-2 Rev. 8

1.4 Ft2 Steamline Rupture Offsite Power Available

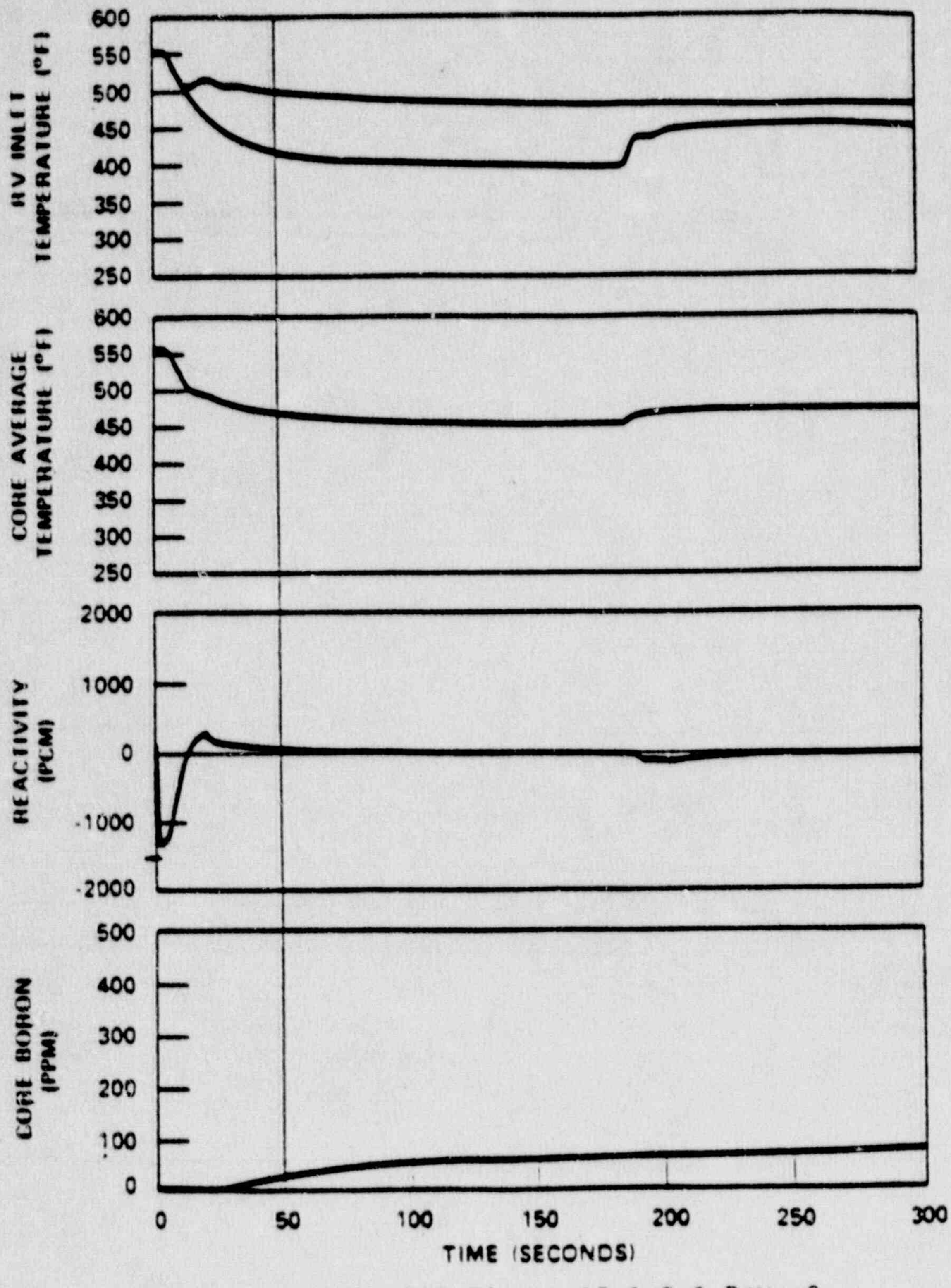


Figure 7.5.7. Catawba SAR Figure 15.1.5-3 Rev. 8
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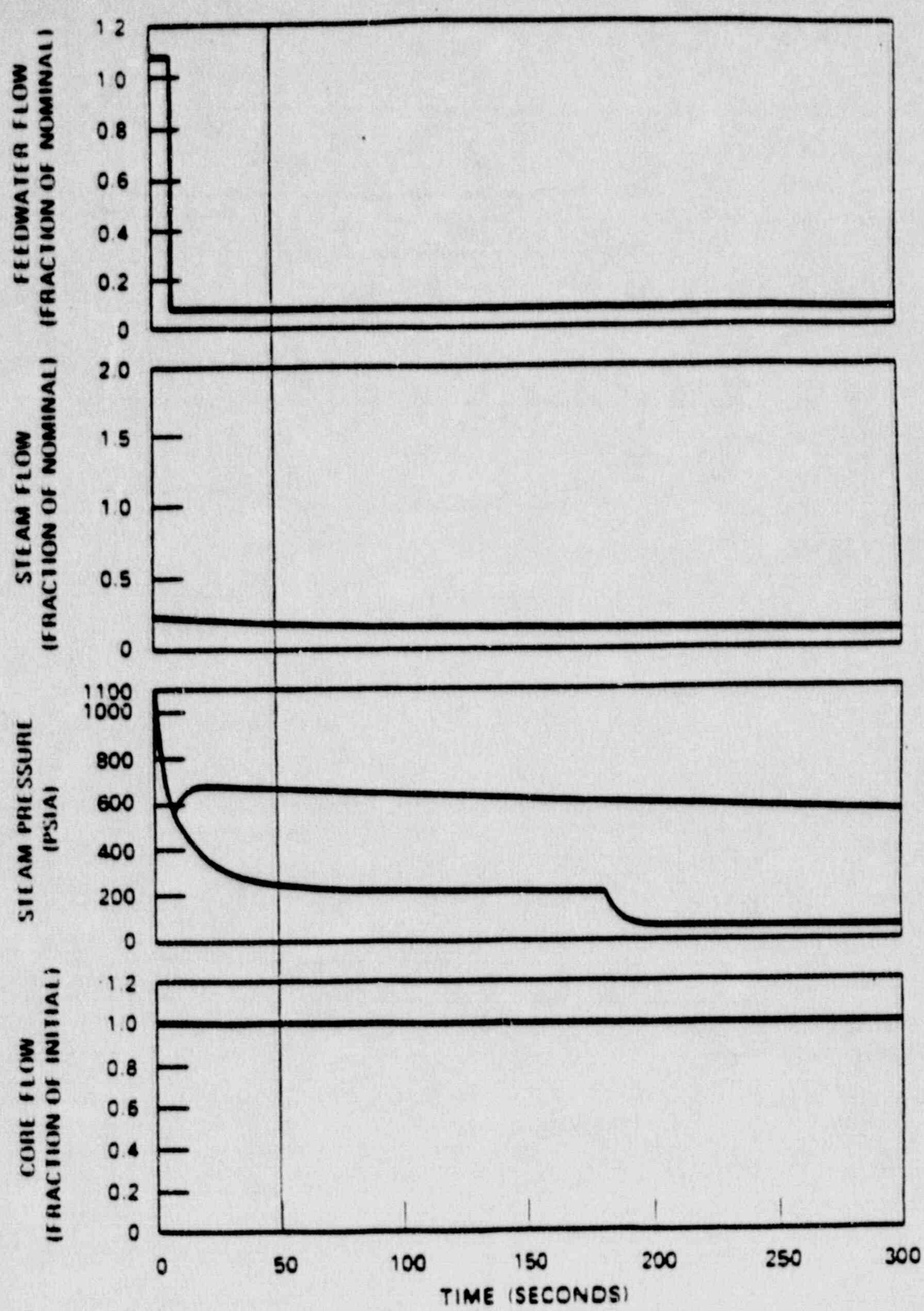

Figure 7.5.8. Catawba SAR Figure 15.1.5-4 Rev. 8

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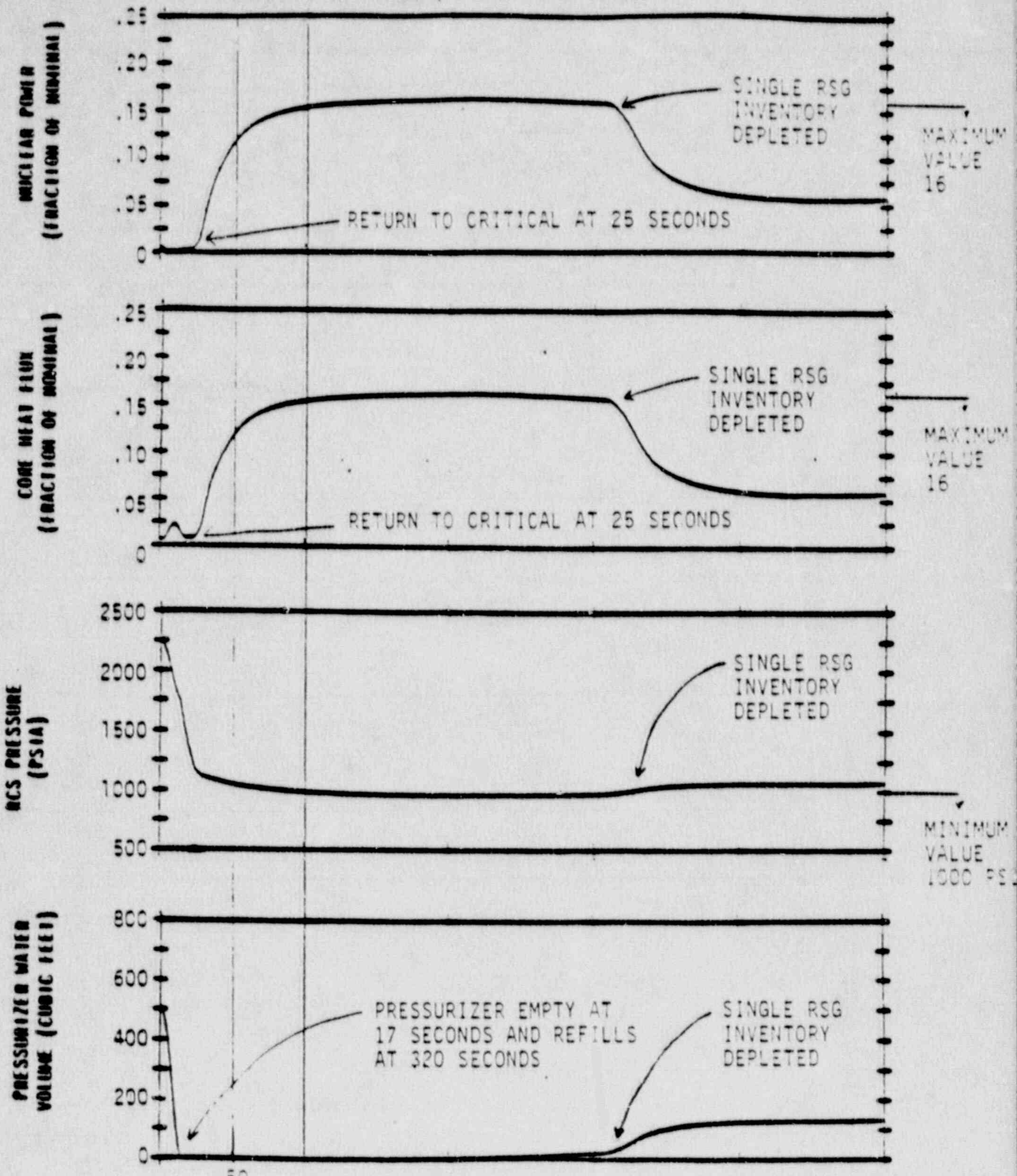

Figure 7.5.9. McGuire SAR Figure 15.1.5-2 1984 Update
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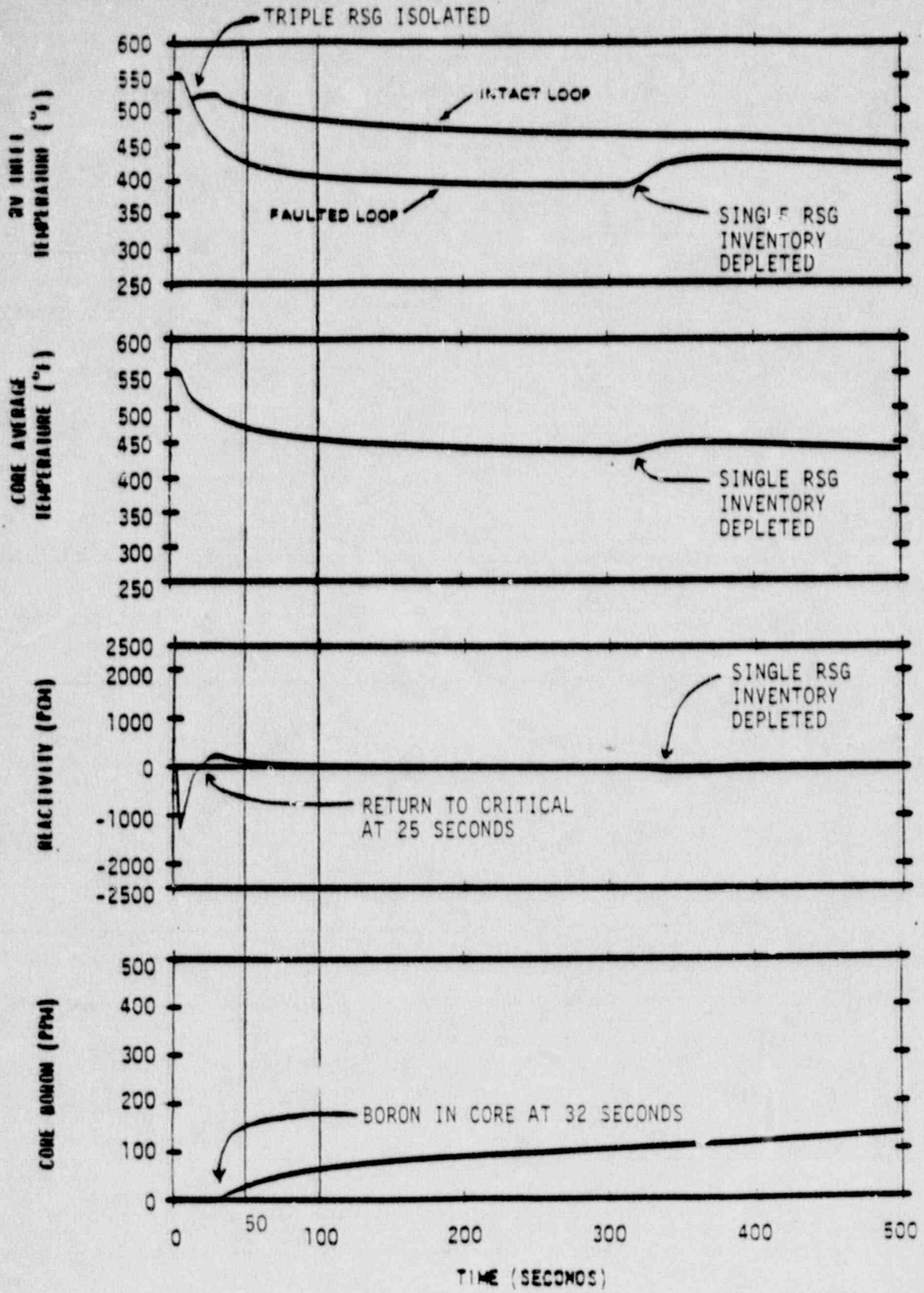

Figure 7.5.10 McGuire SAR Figure 15.1.5-3 1984 Update
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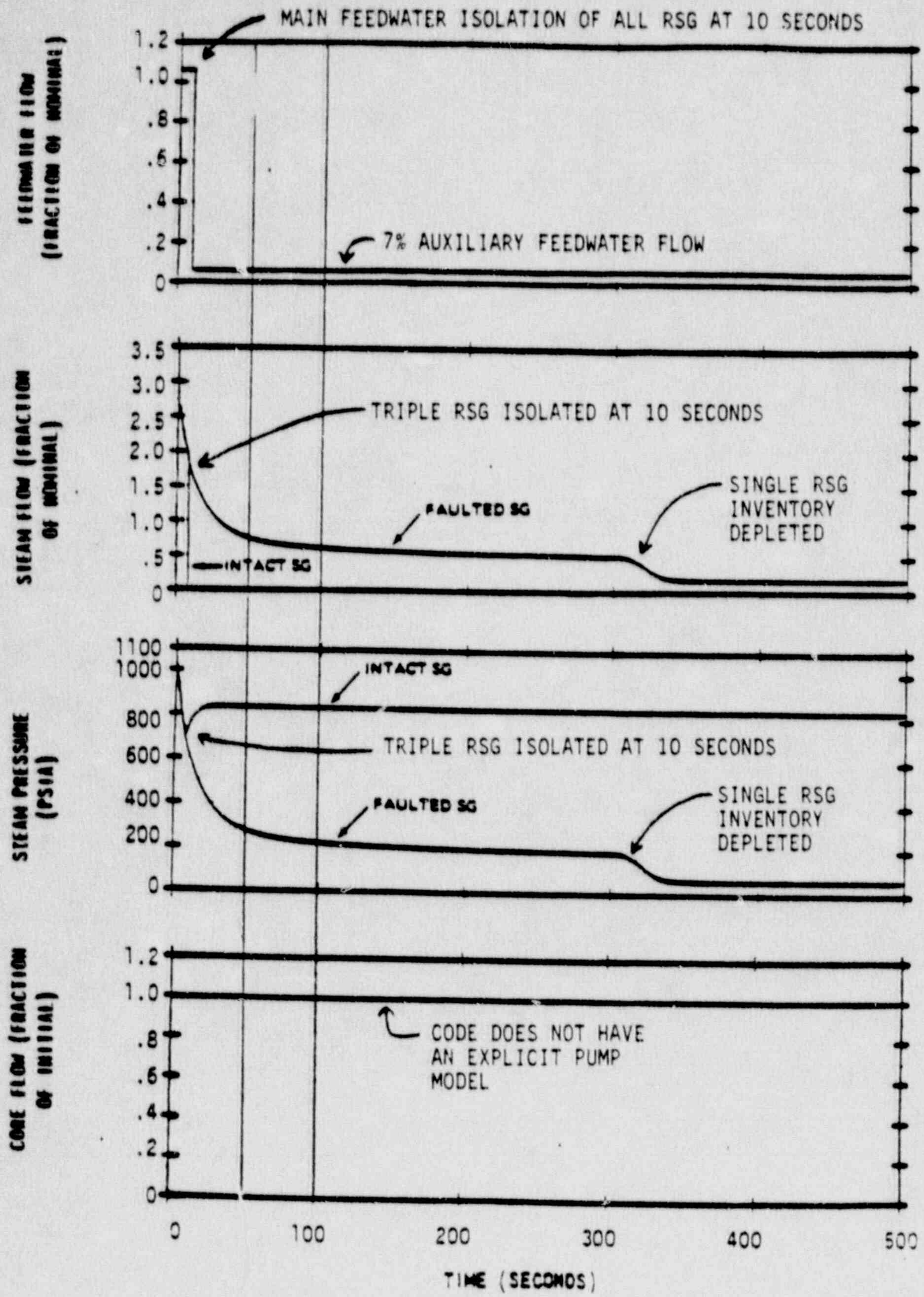

Figure 7.5.11 McGuire SAR Figure 15.1.5-4 1984 Update
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FIGURE 7.5.12. STEAM LINE BREAK

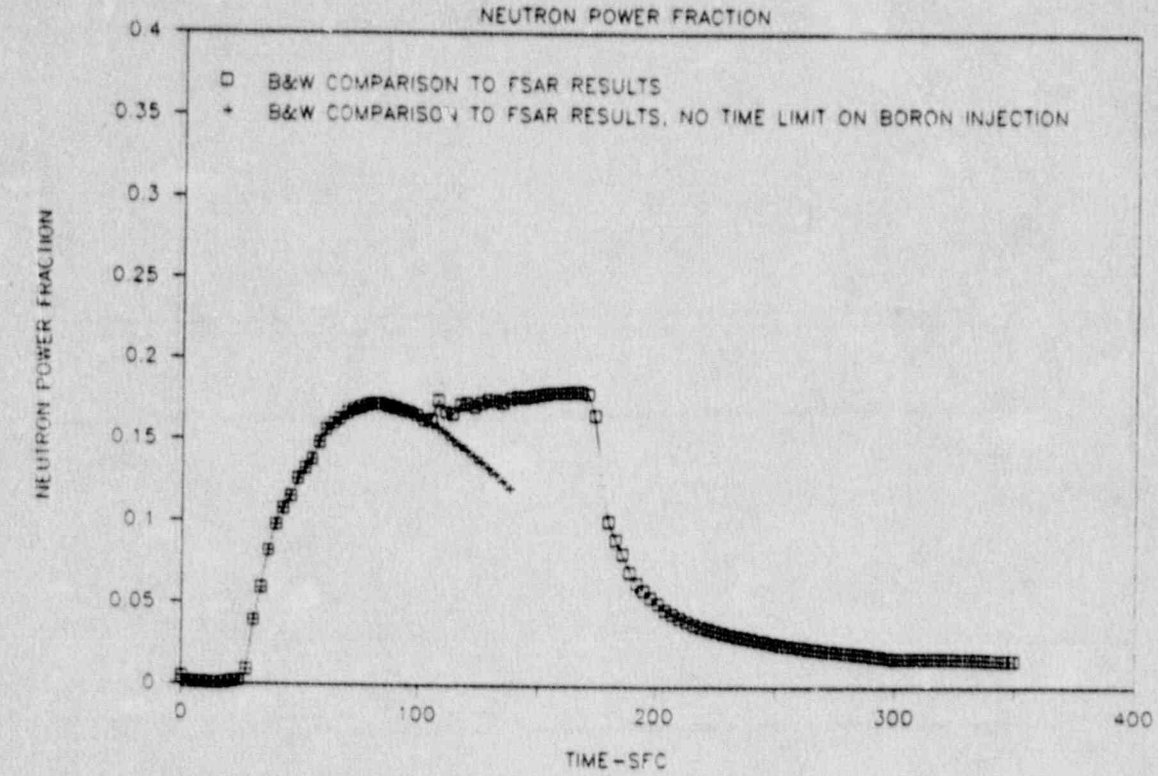


FIGURE 7.5.13. STEAM LINE BREAK

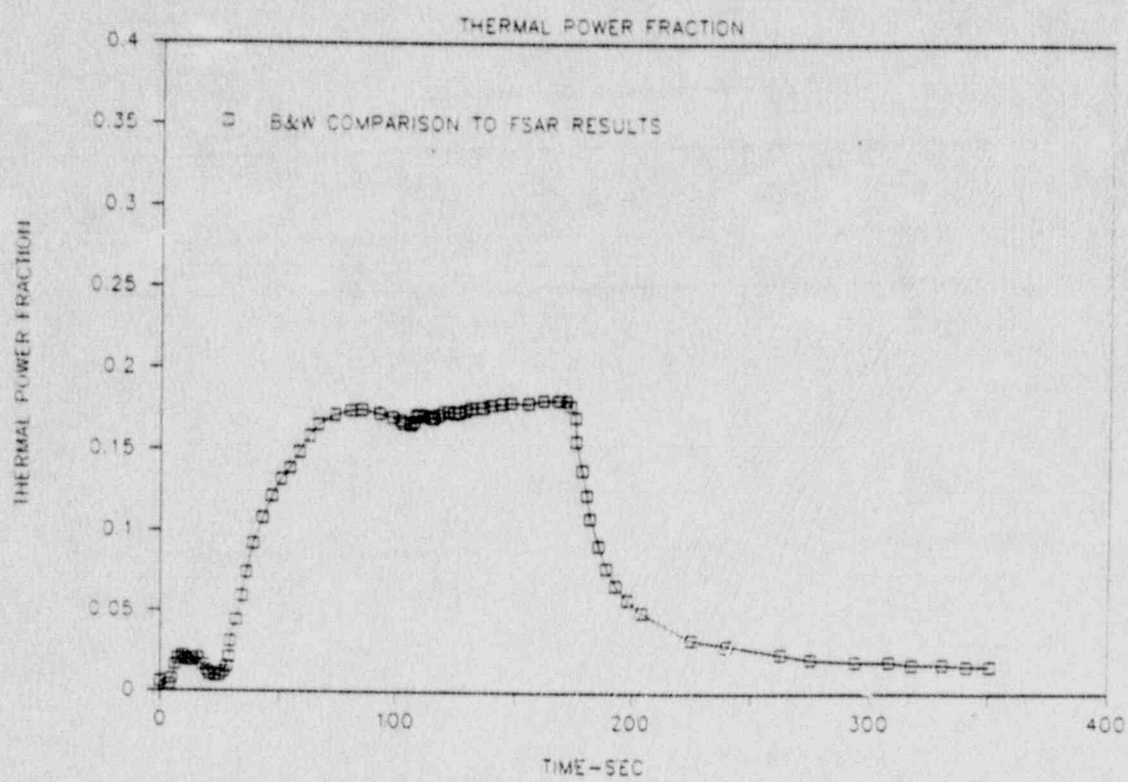


FIGURE 7.5.14. STEAM LINE BREAK

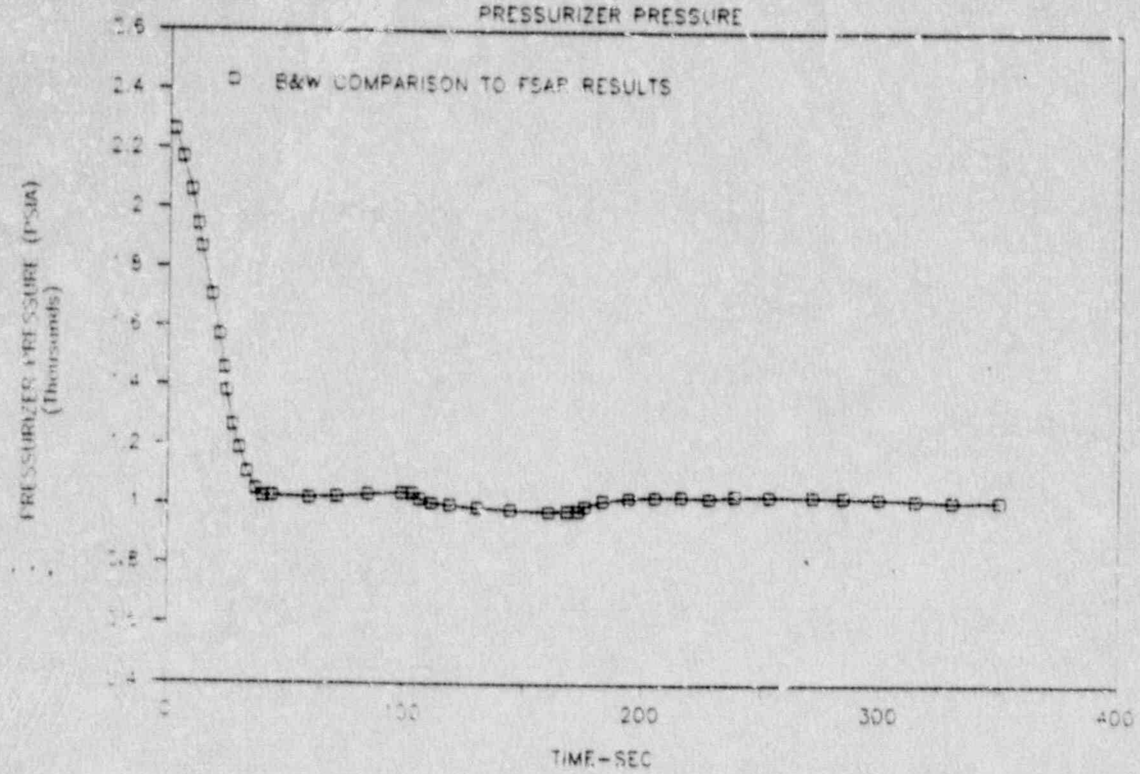
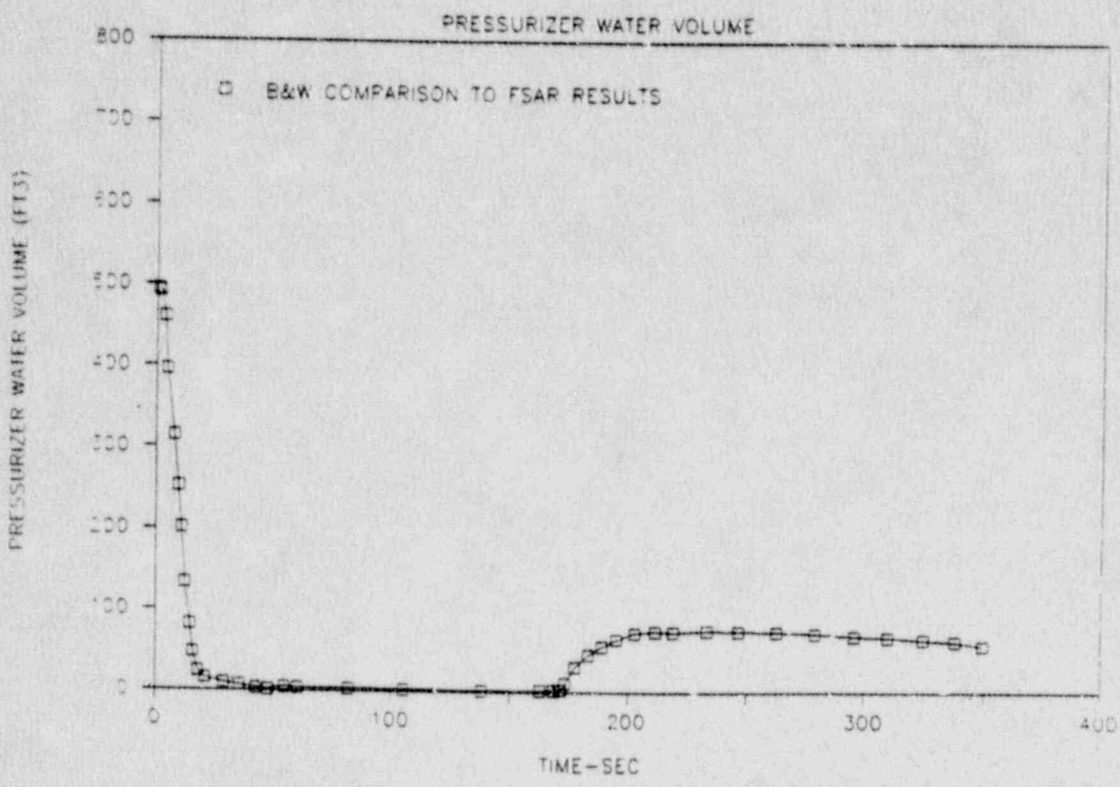


FIGURE 7.5.15. STEAM LINE BREAK



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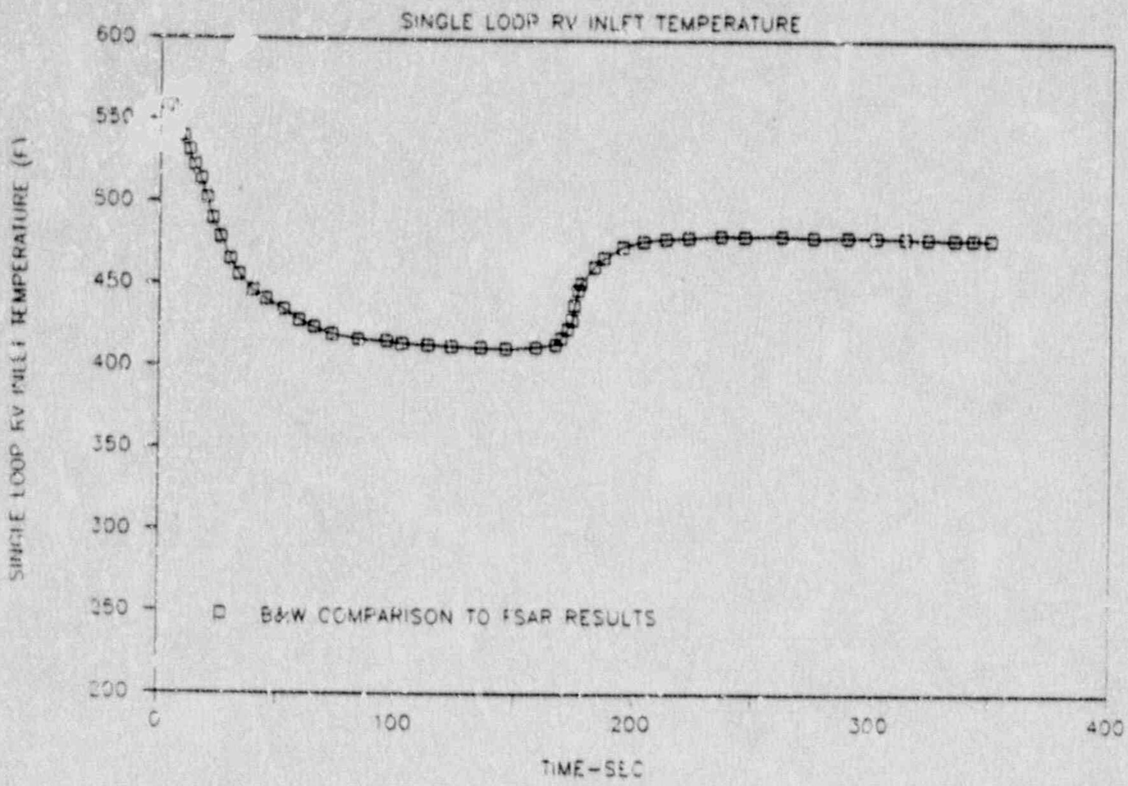

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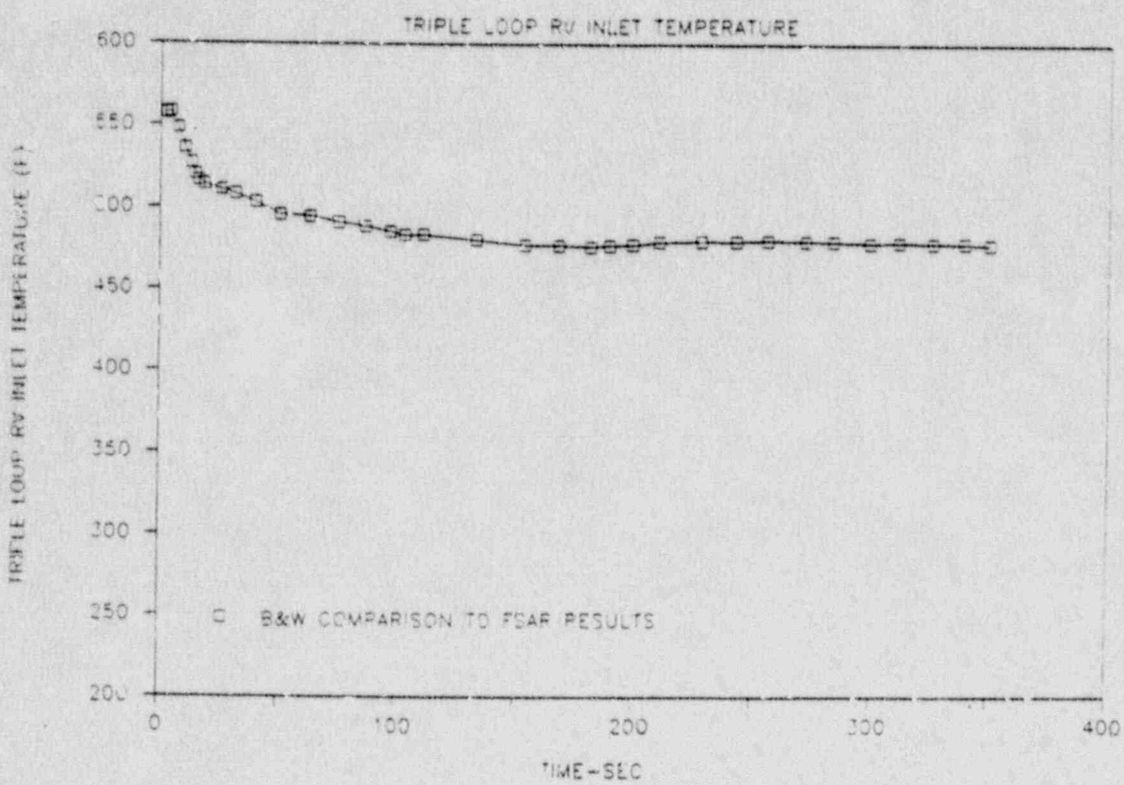
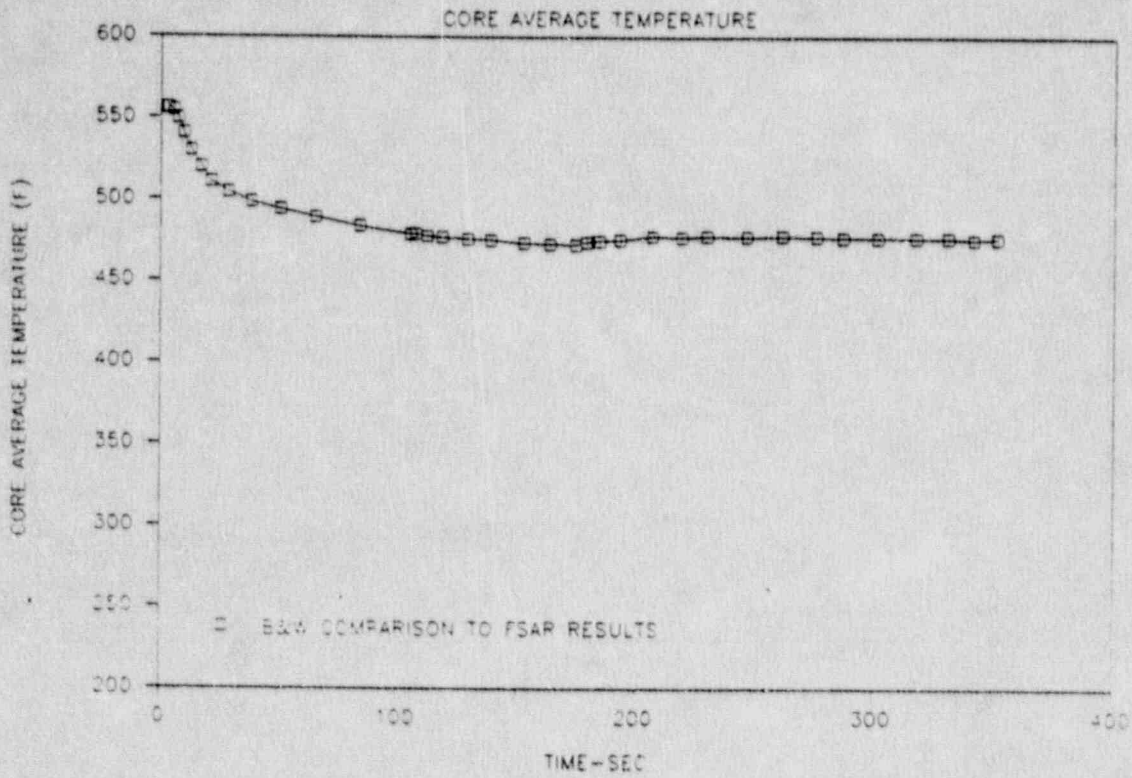


FIGURE 7.5.18. STEAM LINE BREAK



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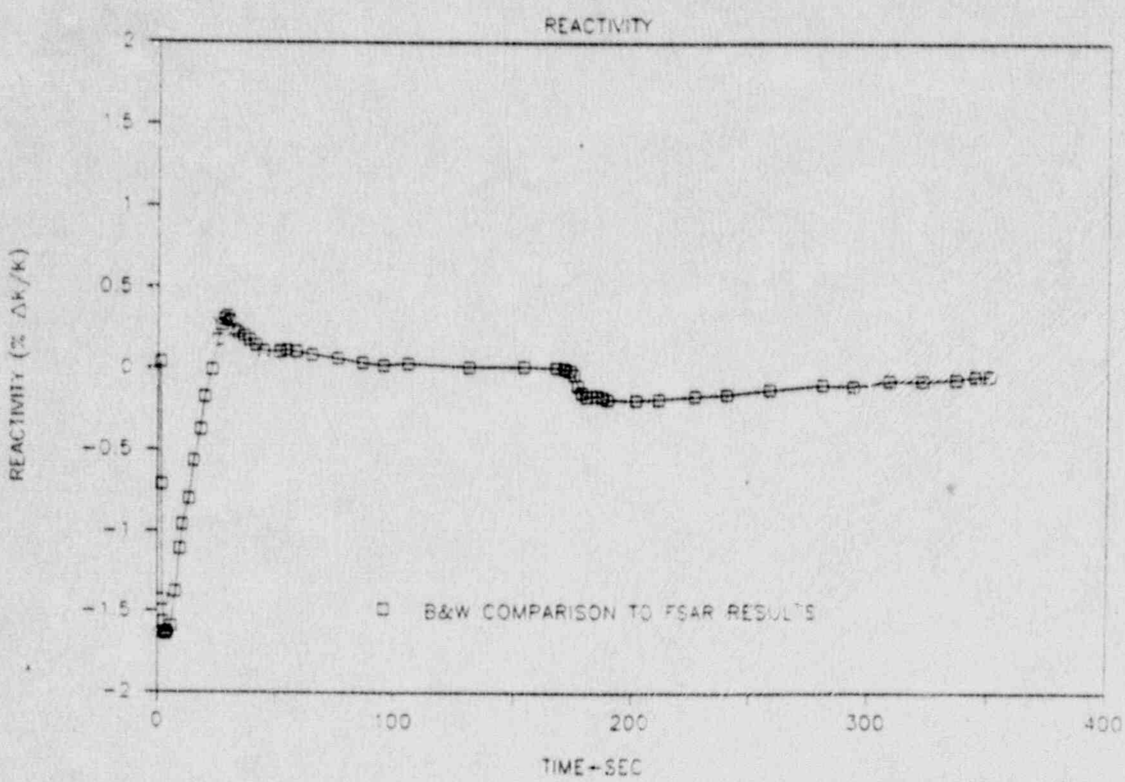


FIGURE 7.5.20. STEAM L NE BREAK

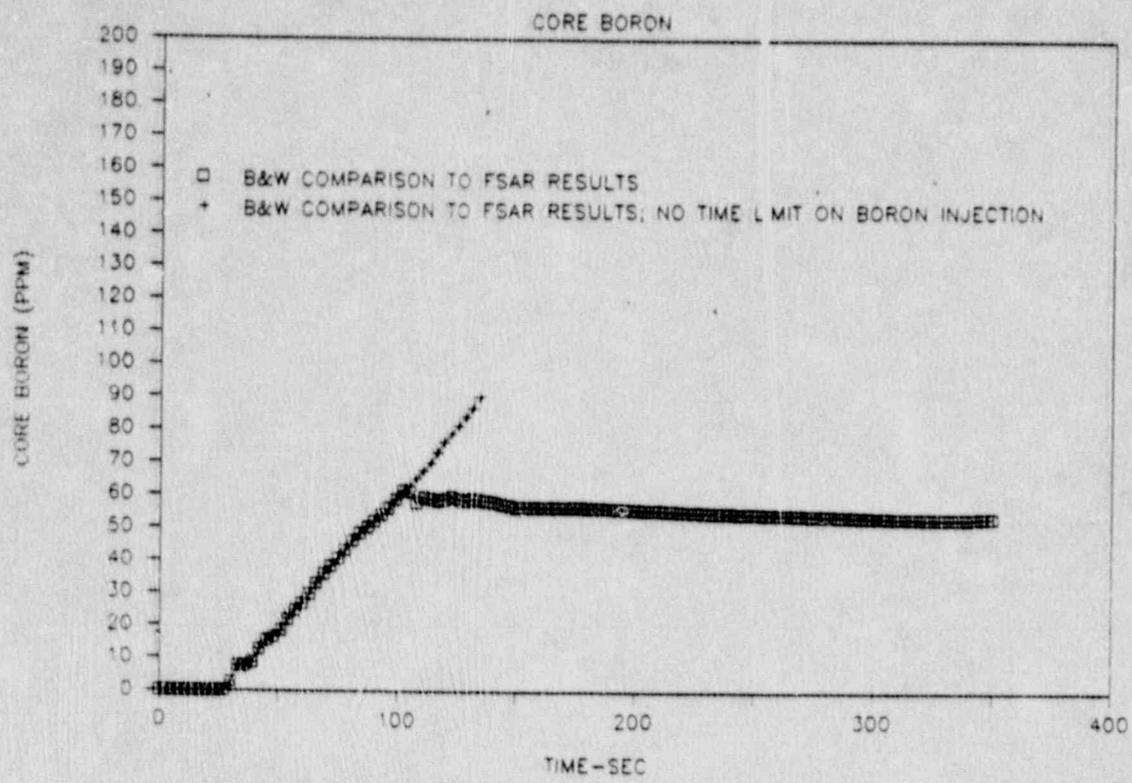


FIGURE 7.5.21. STEAM LINE BREA

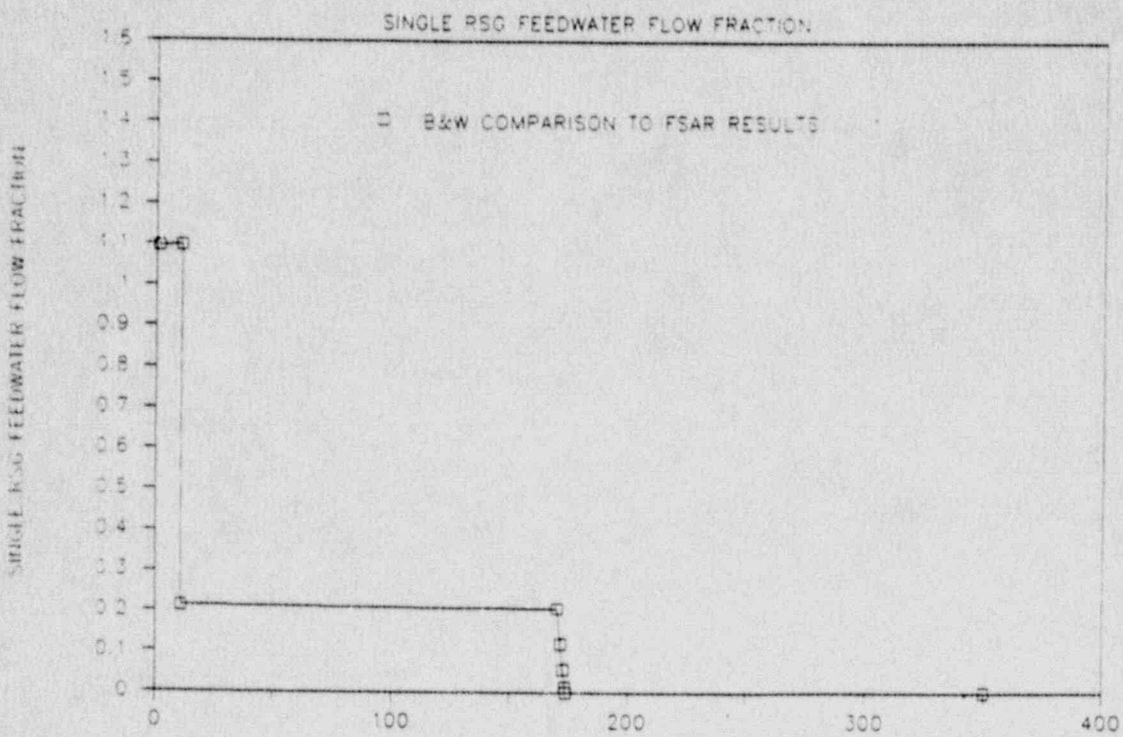


FIGURE 7.5.22. STEAM LINE BREAK

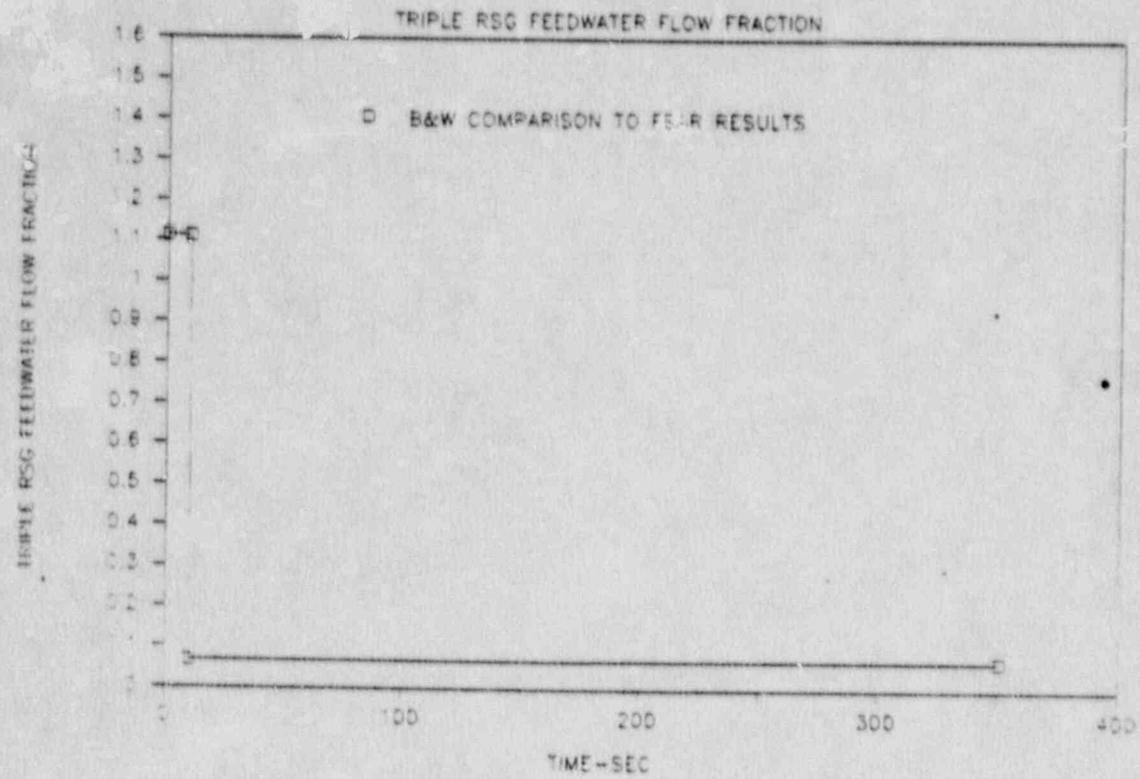
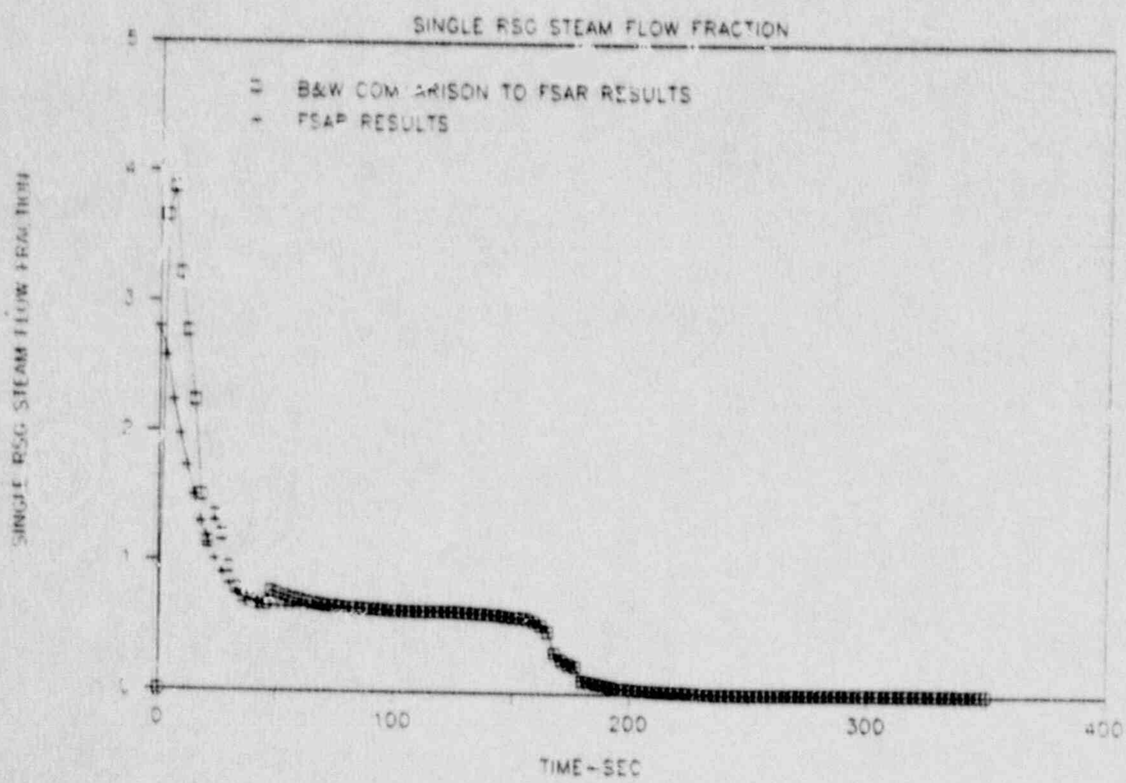


FIGURE 7.5.23. STEAM LINE BREAK



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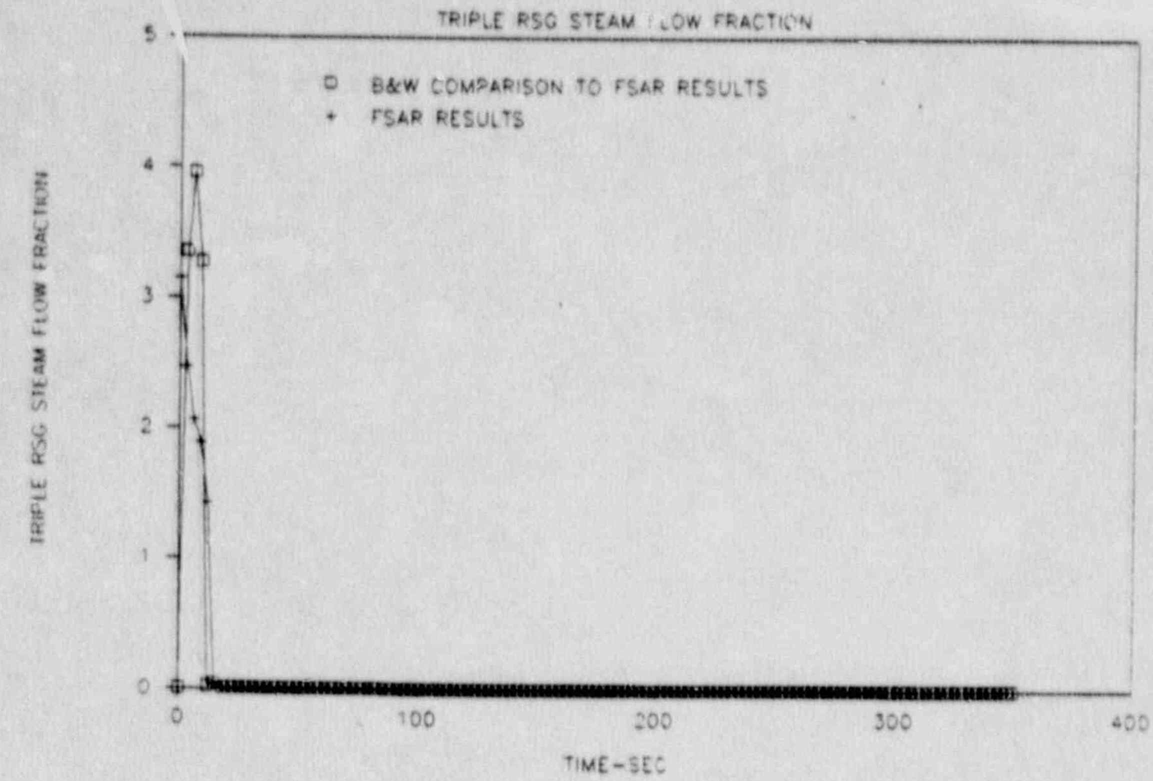

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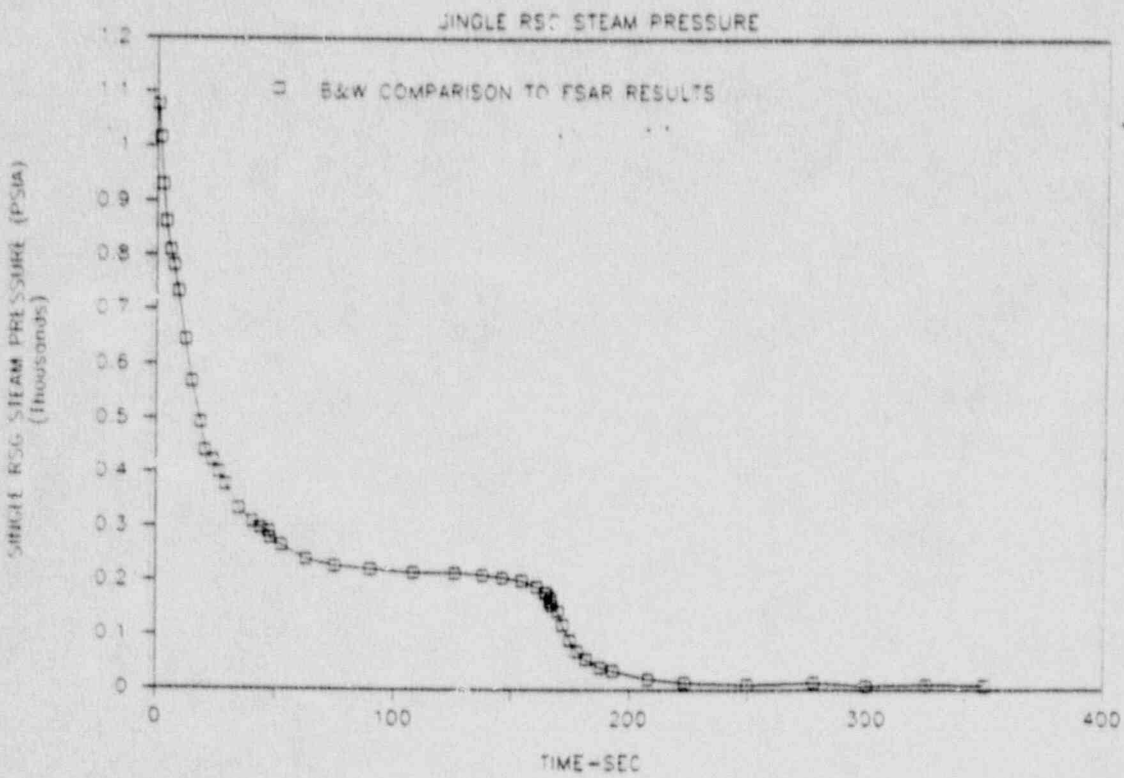

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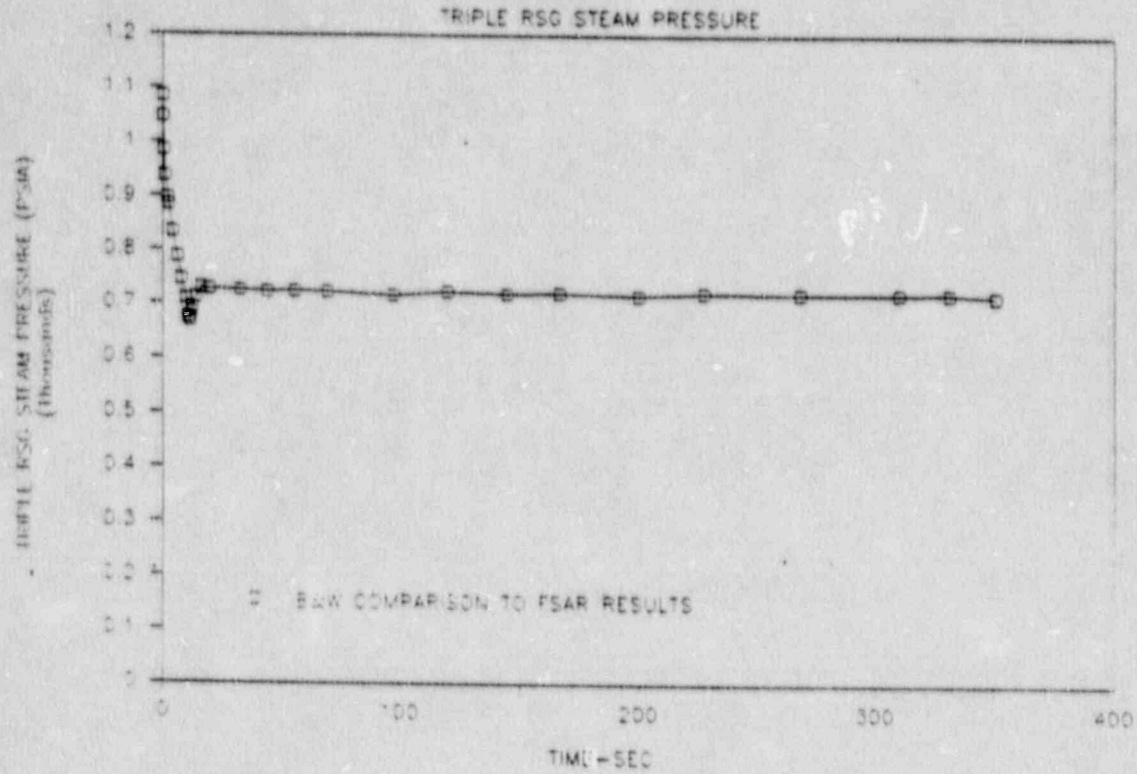

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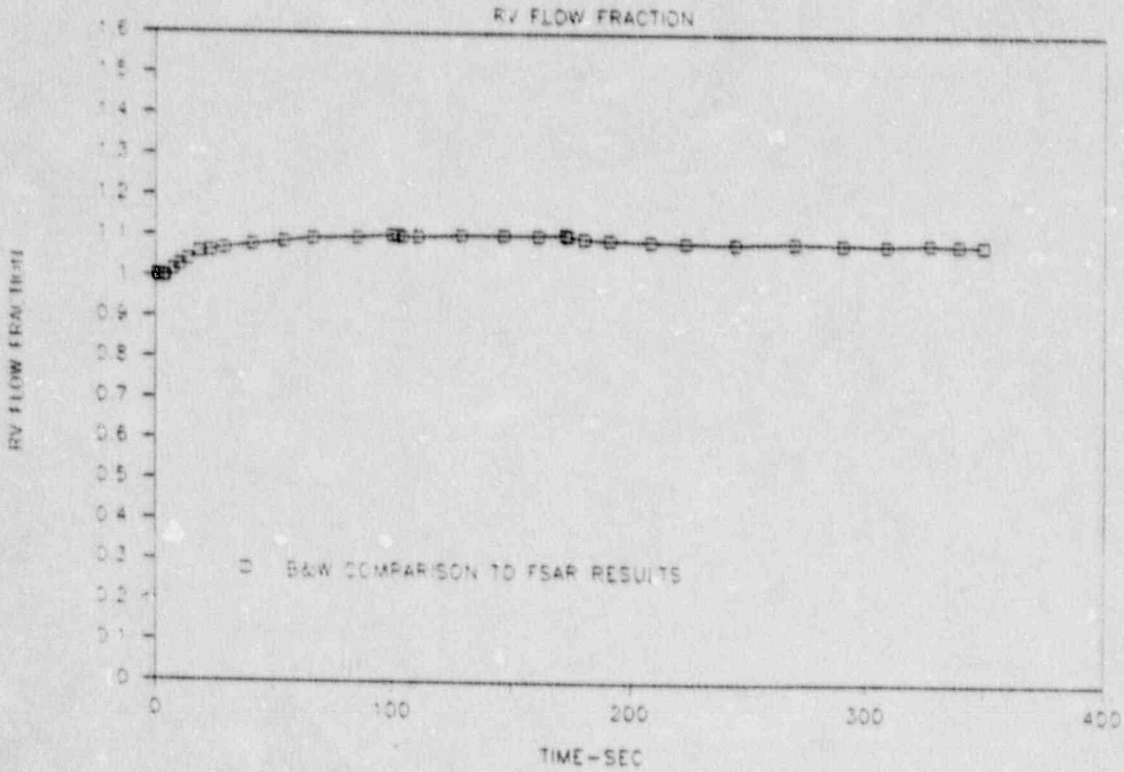

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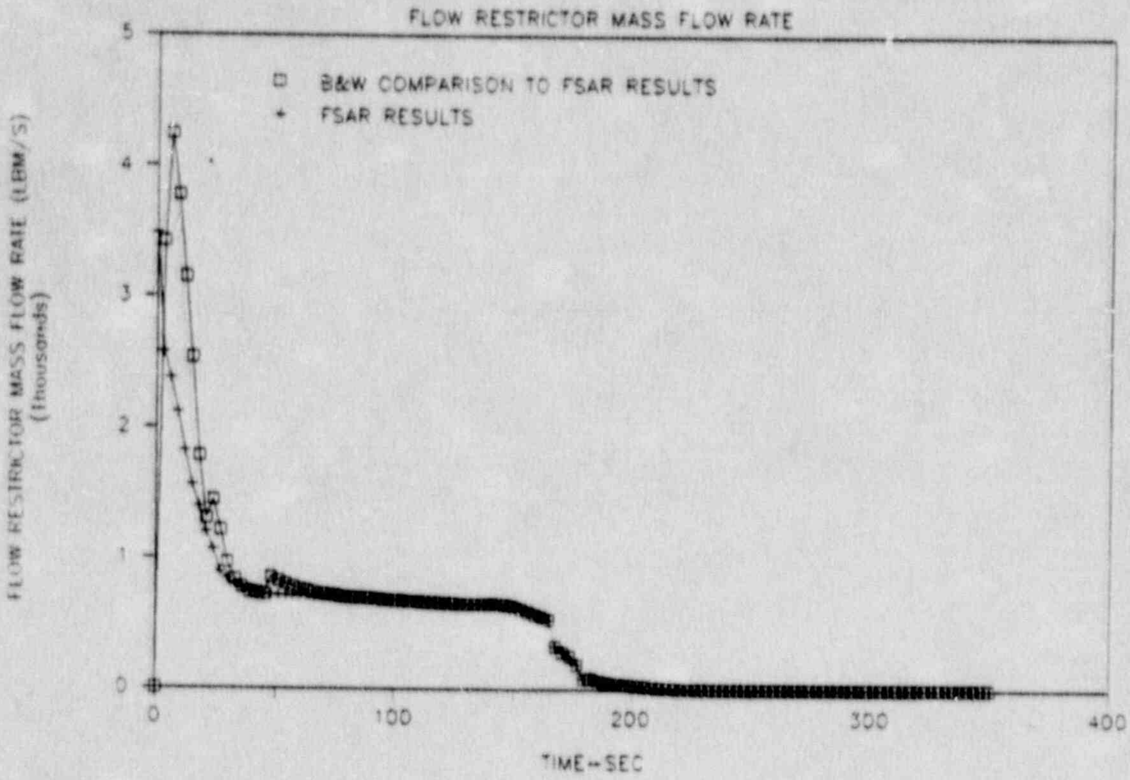


FIGURE 7.5.29. STEAM LINE BREAK

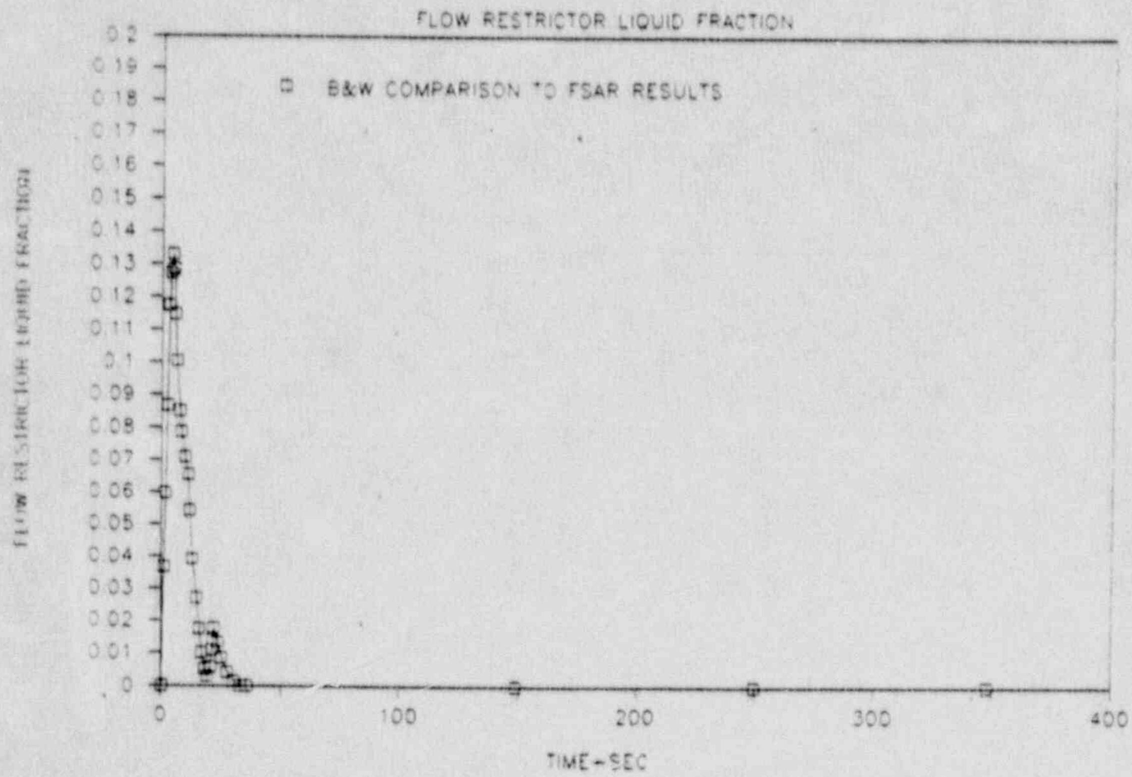


FIGURE 7.5.30. STEAM LINE BREAK

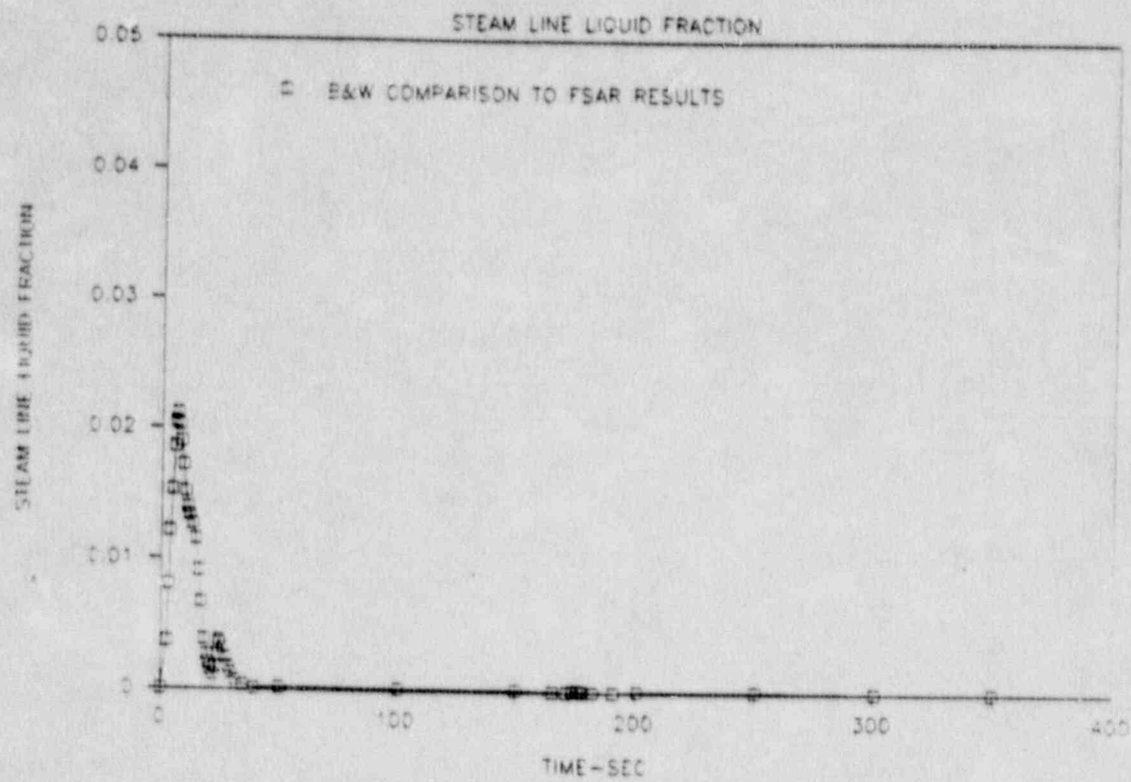


FIGURE 7.5.31. STEAM LINE BREAK

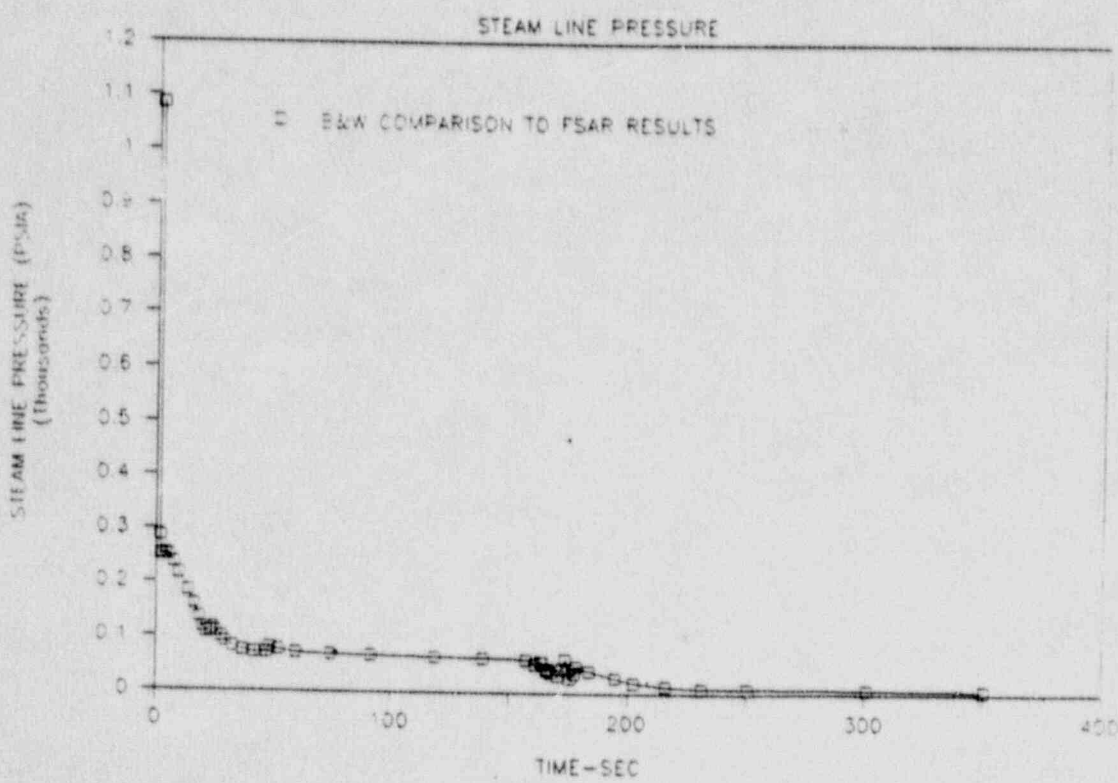
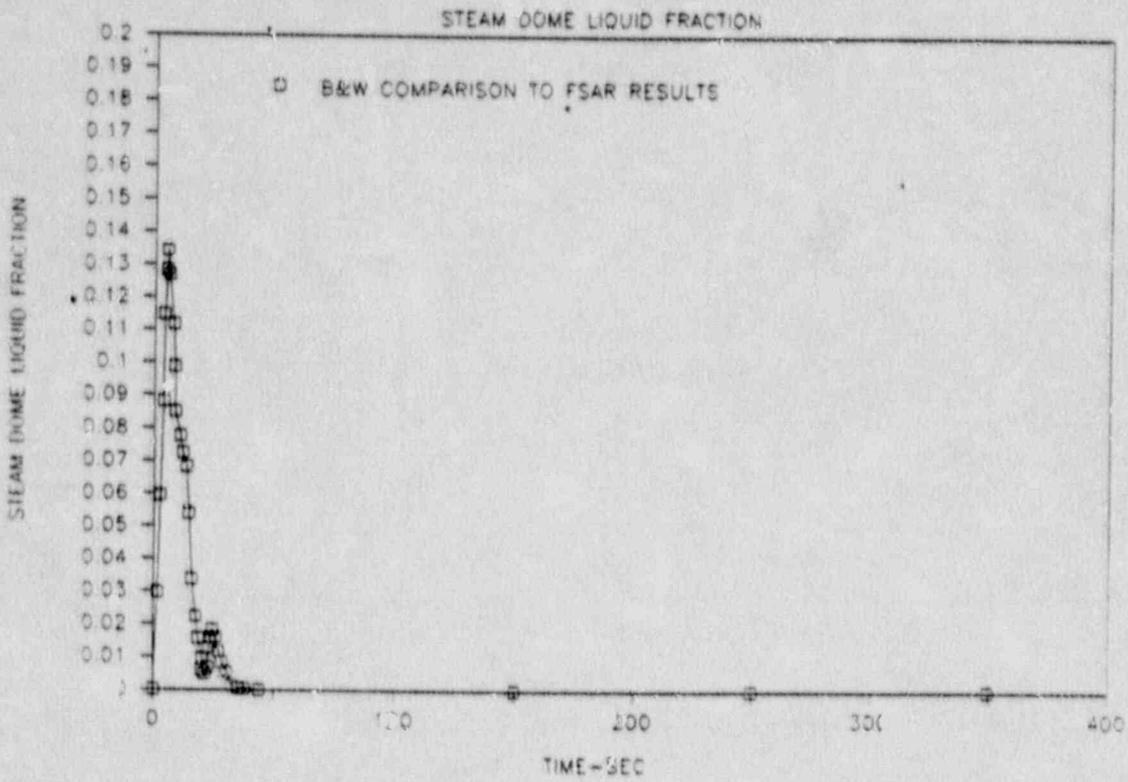


FIGURE 7.5.32 STEAM LINE BREAK



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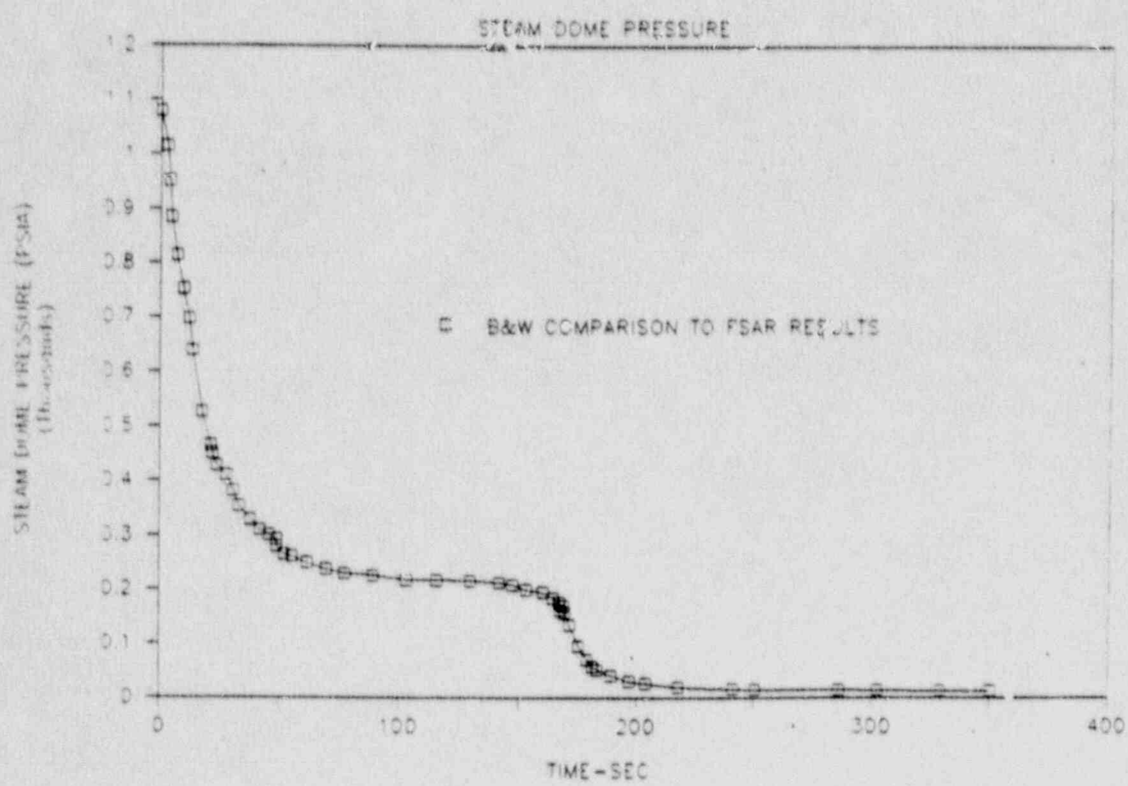

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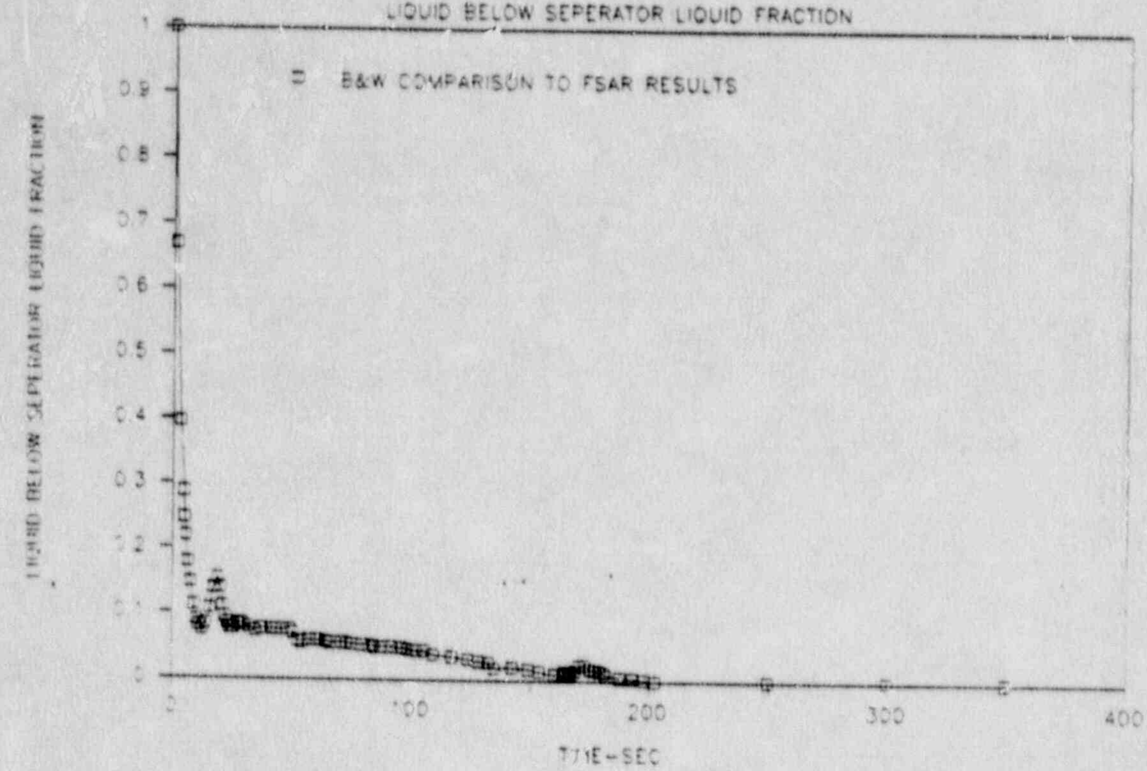

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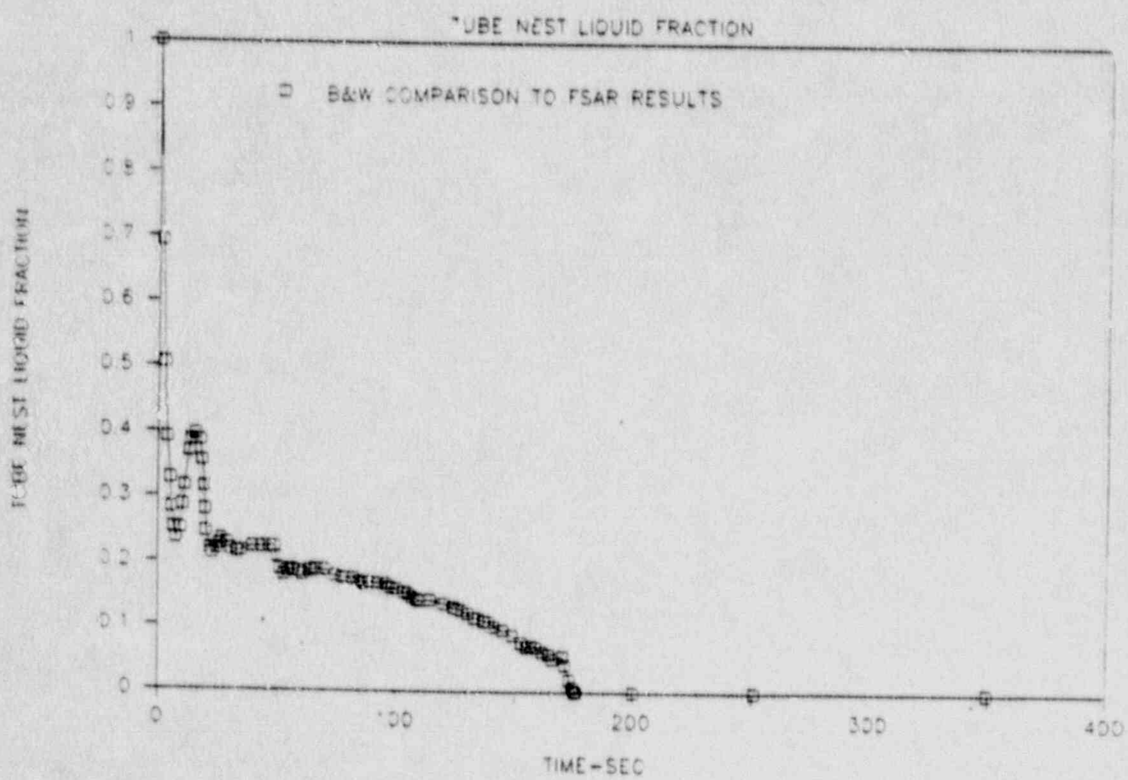

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(804) 385-2000

JHT/89-243

November 28, 1989

Ms. V. H. Wilson
Chief Administrative Section
PMB
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Accepted Version of Topical Report BAW-10169P, "B&W Safety Analysis Methodology for Recirculating Steam Generator Plants."

References: 1. A. C. Thadani to J. H. Taylor, "Acceptance for Referencing of Licensing Topical Report BAW-10169P, RSG Plant Safety Analysis," August 20, 1989.

2. J. H. Taylor to J. A. Norberg, "Topical Report BAW-10169P, Safety Analysis Methodology for Recirculating Steam Generator Plants," JHT/87-228, October 22, 1987.

Dear Ms. Wilson:

Attached are 12 copies each of BAW-10169P-A and BAW-10169-A (non-proprietary version). These are the accepted versions of the topical report entitled, "B&W Safety Analysis Methodology for Recirculating Steam Generator Plants." Included with each report are the NRC acceptance letter (reference 1) and Safety Evaluation Report (SER) and NRC questions with B&W responses that were provided during the review process.

Approval for withholding the proprietary report BAW-10169P from public inspection was requested in the submittal letter (reference 2) that accompanied the topical report. Since the accepted version of BAW-10169P does not include any new information beyond that of the original submittal, B&W does not intend to forward another affidavit defending the proprietary nature of BAW-10169P. It is requested, however, that the NRC approved version of BAW-10169P be treated as proprietary for the reasons given in the affidavit that was attached to reference 2.

Very truly yours,

J. H. Taylor
Manager, Licensing Services

cc: R. B. Borsum
T. L. Baldwin