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Oconee Nuclear Station

UFSAR Chapter 15 Transient Analysis Methodology

DPC-NE-3005-A
Revision 1

August 1999

Nuclear Engineering Division
Nuclear Generation Department
Duke Power Company

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

May 25, 1999

Mr. W. R. McCollum, Jr.
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SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 RE: SAFETY EVALUATION FOR REVISION 1 TO TOPICAL REPORT DPC-NE-3005-P, "UFSAR CHAPTER 15 TRANSIENT AND ACCIDENT ANALYSIS METHODOLOGY" (TAC NOS. MA4713, MA4714, AND MA4715)

Dear Mr. McCollum:

The NRC staff has completed its review of Revision 1 to Topical Report DPC-NE-3005-P, "UFSAR Chapter 15 Transient Analysis Methodology" that was submitted by letter dated February 1, 1999. The revision addresses issues and minor corrections resulting from the staff's previous review of the topical report that was included in the safety evaluation dated October 1, 1998. The principal issue addressed by the change involves credit for main feedwater system isolation that is no longer assumed in the analysis of main steam line breaks. Since isolation of main feedwater at Oconee is not accomplished with safety related equipment, the staff requested that the analyses be preformed with the assumption of continued feedwater.

As described in the enclosed SER, we conclude that the topical report adequately addresses the issues raised in the staff's previous safety evaluation. The methodology in DPC-NE-3005-P, Revision 1 is, therefore, approved and found acceptable for performing the Updated Final Safety Analysis Report (UFSAR) Chapter 15 transient and accident analysis for Oconee Units 1, 2, and 3. The plant analyses contained in the topical report are typical of those that will be incorporated into the present UFSAR. For subsequent core reloads or other plant modifications, you should justify that the analyses in the topical report bound the results that would be obtained for the new plant condition or perform new analyses that are conservative for that purpose.

Sincerely,

David E. LaBarge, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION OF THE OFFICE OF NUCLEAR REACTOR REGULATION OF REVISION 1 TO DPC-NE-3005-P UFSAR CHAPTER 15 TRANSIENT ANALYSIS METHODOLOGY DUKE ENERGY CORPORATION OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

1.0 INTRODUCTION

Duke Energy Corporation (DEC) first received approval from the NRC staff to perform transient analyses for core reloads at the Oconee Nuclear Station Units 1, 2, and 3 in 1981. This methodology has been updated and revised on several occasions. Topical Report DEC-NE-3005-P describes the methodology that is currently used for Oconee (Reference 1). The initial review by the staff of DEC-NE-3005-P is discussed in Reference 2. In Reference 2, the staff found that the DEC methodology, as documented in DPC-NE-3005-P, was acceptable, with the exception of the following four items:

1. A peak reactor coolant pressure acceptance criterion of 110 percent of the design pressure for the postulated locked-rotor accident should be included.

2. The proposed steam generator tube rupture methodology should be modified to assume no operator action at the beginning of the event and for consistency with the updated final safety analysis report (UFSAR), which credits the low-pressure reactor trip.

3. The licensee should modify the proposed methodology for large and small main steamline break (MSLB) events to assume failure of main feedwater to isolate. The Oconee plants are not equipped with safety-related main feedwater isolation capability.

4. Since the occurrence of a small steamline break is considered to be of moderate frequency, the acceptance criteria for events of this type should include the requirement of no fuel failure.

DEC made the required modifications, which were submitted for NRC staff review as Revision 1 to DPC-NE-3005-P (Reference 3).

2.0 EVALUATION

The NRC staff verified that the requirements from the previous staff review had indeed been incorporated into the topical report. The acceptance criteria for a postulated locked-rotor accident now include the requirement that the reactor system pressure remains below 110 percent of design pressure. The analysis of a steam generator tube rupture has been modified to assume automatic reactor trip 20 minutes after the steam generator tube rupture. This value is conservative in comparison with the analysis performed by DEC as documented in the UFSAR. In the UFSAR analysis, reactor trip was calculated to occur in 8 minutes if reactor operators take no remedial action. Actually operators are instructed to reduce reactor power at a rapid rate and to increase charging flow. The reduction of power and increase in charging

may prevent a reactor trip. Preventing reactor trip would lessen the likelihood of a challenge to the main steam safety valves and would, thereby, reduce the offsite dose consequences.

The Oconee nuclear plants are not equipped with safety-related main feedwater isolation valves that would actuate following an MSLB accident. The NRC staff, therefore, required the licensee to perform analyses of an MSLB, assuming continued operation of the main feedwater system. The additional feedwater might cause recriticality in the core and a return of fission power generation even after reactor trip. The combination of low reactor system pressure and core power generation might cause the critical heat flux (CHF) to be exceeded thus leading to fuel cladding damage.

The licensee analyzed the reactor system aspects of MSLBs using the RETRAN-02 computer code. RETRAN-02 calculates, among other parameters, the break flow from the broken steamline, the heat transfer from the primary system through the steam generator tubes, the reactor power, and condition of the fluid entering the reactor core.

A break in a main steamline will cause the coolant loop with the break, the "affected" loop, to cool much more rapidly than the rest of the primary coolant system. The licensee calculates the effect of this asymmetric cooldown by dividing the core into two halves, each associated with one of the coolant loops. Mixing between the two loops at the core inlet plenum is input to RETRAN-02 by means of a mixing coefficient. The mixing coefficient was derived from a series of tests at Oconee Unit 1 in which a temperature differential was introduced between the reactor system cold legs while the reactor was at power. Reactor vessel temperatures at various locations were monitored.

The once through steam generators at Oconee do not contain steam separation equipment. A large steamline break would, therefore, be expected to result in a substantial amount of liquid being expelled through the broken steamline. The licensee uses modeling in the RETRAN-02 code, which inhibits liquid entrainment but, at the same time, maintains full steam generator heat transfer. With continued uncontrolled feedwater flow into the affected steam generator, no liquid entrainment is permitted until the steam generator is almost full. Then liquid is permitted to flow out of the broken steamline. These assumptions are designed to conservatively maximize steam generator heat transfer.

The analysis includes the reactivity and flux peaking effect of the maximum-worth control assembly remaining outside the core following a reactor trip. The amount of local flux peaking is calculated using the SIMULATE-3P computer code. SIMULATE-3P is a core neutronics code used to generate multi-dimensional core power distributions. The maximum-worth control assembly is conservatively assumed to be in the cooler half of the core, which will experience the greatest increase in local reactivity and power. A 10 percent reduction in the worth of the remaining control assemblies is also assumed.

The VIPRE-01 computer code is used to calculate the CHF within the core, including the hot channel. For steamline break analysis, the licensee developed VIPRE-01 input that models the entire core. The entire core was modeled so that flow redistribution from the hottest fuel bundle could be described. Simulation of the entire core with flow redistribution produces a more conservative result than modeling it as a single channel. The staff has previously approved use of RETRAN-02, SIMULATE-3P, and VIPRE–01 computer codes for use in Oconee licensing analyses (Reference 2).

Calculation of heat transfer between the primary system and the affected steam generator is significant in evaluating the cooling of the reactor system. The depressurization of the affected steam generator will cause an increase in boiling heat transfer between the tubing exterior and secondary coolant. The Thom Nucleate Boiling Correlation is used in RETRAN-02. The Chen Nucleate Boiling Correlation is programmed into the RELAP5 MOD3 code, which was developed by the staff. The licensee provided an evaluation of the accuracy of the Thom correlation in predicting nucleate boiling test data (Reference 4). The evaluation showed no significant difference between the predictions of the Thom and Chen correlations. The Thom correlation was shown to calculate slightly higher heat transfer in comparison to the test data which would be conservative for MSLB evaluations.

The core flood tank model in RETRAN-02 assumes the cover gas and liquid to be at the same temperature during the discharge of a flood tank. The NRC staff questioned the conservatism of this assumption since if the cover gas cooled faster than the liquid, a smaller flood tank discharge flow rate might be predicted and less boric acid would be introduced into the core to prevent core recriticality. The licensee stated that it assumes a conservatively low flood tank initial temperature, which compensates for the non-equilibrium effect. The licensee further states that the DEC RETRAN-02 core flood tank model was benchmarked against Oconee full-scale blowdown tests and that the model showed good agreement with the data.

The licensee performed sensitivity studies to evaluate the effects of single failures. Various combinations of reactor coolant pump operation, main feedwater flow control, and availability of offsite power were considered. The most severe single failure was found to be failure of one train of high-pressure injection (HPI). HPI adds boric acid to the reactor system and acts to dampen any return to power calculated to result from overcooling of the reactor system.

With offsite power available, the reactor coolant pumps, as well as the feedwater and condensate systems, could remain operable following an MSLB. Continued main feedwater flow to the affected steam generator would overcool the reactor system, increase core power generation, and reduce the reactor system pressure. Continued reactor coolant pump operation would also increase overcooling by facilitating heat removal from the reactor system. Reactor coolant pump operation would, however, act to maintain core heat transfer and the margin to CHF.

The licensee evaluated continued feedwater system operation with and without assumed operation of the integrated control system (ICS) to control steam generator level. The licensee calculated a return to criticality using both assumptions. The reactor coolant pumps were assumed to be operating, but the reactor coolant pumps in the unaffected loop were assumed to trip at 100 seconds into the event when voiding occurred in that loop. The RETRAN-02 coolant pump model became unstable under these conditions. The greatest return to power (13.09 percent) was calculated to occur if the ICS was assumed to remain operating, controlling steam generator level. The licensee believes that this is the result of increased boiling in the steam generator for the ICS operable case. For the ICS failure case, RETRAN-02 predicts that the steam generator will gradually fill with liquid after the initial blowdown and will experience less boiling.

The staff performed an audit calculation of this event using the RELAP5 MOD3 computer code. In the RELAP analysis, the ICS was assumed to fail so that main feedwater flow was uncontrolled. The affected steam generator filled and liquid flowed out the steamline. All the

reactor coolant pumps were assumed to remain operating. The reactor coolant pump model in RELAP did not become unstable when voiding occurred in the unaffected reactor coolant loop. A very low value of decay heat was assumed to maximize overcooling. The RELAP code calculated slightly less reactor system cooling than did the licensee's analysis. RELAP also calculated lower reactor system pressures. The lower pressures calculated by RELAP resulted from not assuming the tripping of the reactor coolant pumps in the unaffected loop which, if tripped, would allow that loop to become stagnant. The lower reactor system pressure calculated by RELAP allowed additional boric acid to be injected by the HPI system and the core flood tanks. The core remained subcritical in the staff's calculation. Therefore, the licensee's analysis was conservative.

Reactor operators are trained to manually trip all reactor coolant pumps if the difference between the reactor system temperature and the saturation temperature (subcooling margin) becomes sufficiently small. This action is taken because for a certain range of small reactor coolant system break sizes, unacceptable results might be obtained if the reactor coolant pumps were not manually tripped early into the event. A large MSLB would resemble a small reactor system break in that the subcooling margin would also be lost and HPI would be actuated. Reactor operators might be expected to manually trip the reactor coolant pumps following an MSLB. The staff requested the licensee to evaluate this scenario. The licensee preformed the analysis by tripping the reactor coolant pumps in the affected loop at 160 seconds when the core power from re-criticality was calculated to be greatest. The reactor coolant pumps in the unaffected loop were, as before, tripped at 100 seconds because of the stability problem with RETRAN-02.

The ratio between the maximum heat flux in the core and the critical heat flux is called the departure from nucleate boiling ratio (DNBR). The DNBR is evaluated with the VIPRE-01 code. Using VIPRE-01, the licensee calculated a DNBR of 3.28 when the coolant pumps in the unaffected loop are allowed to remain running. A DNBR of 3.15 is calculated when the coolant pumps in the unaffected loop were tripped at the time of maximum return to power in the core. Both these values are in excess of the minimum DNBR limit at Oconee for low reactor system pressure, which is 1.45.

The smallest value of DNBR for a large MSLB was calculated to occur when the concurrent loss of offsite power was assumed. Loss of offsite power would cause the reactor coolant pumps, as well as the main feedwater and condensate pumps, to trip. A minimum DNBR of 1.51 was calculated to occur 1.9 seconds into the event as a result of the loss of forced reactor coolant flow and the decrease in reactor system pressure. The DNBR limit at Oconee is 1.193 for the range of pressures that is calculated for this event.

The licensee evaluated the consequences of main feedwater to isolate for a spectrum of small steamline breaks. The requirement of no fuel failures was included as an acceptance criterion. Using RETRAN-02, the licensee identified a break size so that automatic reactor trip would not occur on either low reactor system pressure or high reactor power. As the steam generators depressurize, the unisolated feedwater flow rate would increase. In addition, as steam is lost from the break, there would be less steam flow to the turbine, less steam flow to the feedwater heaters, and a reduced feedwater temperature. The resulting decrease in reactor system cold leg temperature would act to attenuate the neutron flux leakage leaving the reactor. Neutron flux attenuation would cause an error in the indicated value of reactor core power so that core power might exceed the normal reactor trip setting. The licensee evaluated the core DNBR and

the internal fuel temperature for this event and determined that the applicable safety limits will be maintained to prevent fuel failure. In accordance with Reference 5, the licensee will modify Chapter 16 of DPC-NE-3005-P, Revision 1 to indicate that the acceptance criteria for this analysis are that no fuel damage will occur and that the offsite doses will remain within 10 percent of the 10 CFR Part 100 limits.

3.0 CONCLUSION

On the basis of its review of Revision 1 to DPC-NE-3005-P, including supplemental information provided by the licensee, the staff concludes that the licensee has adequately addressed the conditions contained in the staff's original safety evaluation (Reference 2). The methodology in DPC-NE-3005-P, Revision 1, is, therefore, approved and found acceptable for performing UFSAR Chapter 15 transient and accident analysis at Oconee. The plant analyses contained in the topical report are typical of those that will be incorporated into the UFSAR. For subsequent core reloads or other plant modifications, the licensee should justify that the analyses in the topical report bound the results that would be obtained for the new plant condition or should perform new analyses that are conservative for that purpose.

Principal Contributor: Walton L. Jensen, Jr.

Date: May 25, 1999

REFERENCES

1. UFSAR Chapter 15 Transient Analysis Methodology," DPC-NE-3005-P, Duke Power Company, July 1997.

2. Letter from David E. LaBarge, NRC to W. R. McCollum, Duke Energy Corporation, "Review of Updated Final Safety Analysis Report, Chapter 15, Transient Analysis Methodology Submittal-Oconee Nuclear Station, Units 1, 2, and 3," October 1, 1998.

3. Letter from M.S. Tuckman, Duke Energy Corporation, to NRC Document Control Desk, February 1, 1999.

4. Process, Enhanced, and Multiphase Heat Transfer," a Festscrift for A.E. Bergles, Begell House, 1996.

5. Letter from M.S. Tuckman, Duke Energy Corporation, to NRC Document Control Desk, May 5, 1999.

UNITED STATES
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October 1, 1998

RECEIVED
OCT 7 1998
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JUN 17 1999
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SUBJECT: REVIEW OF UPDATED FINAL SAFETY ANALYSIS REPORT, CHAPTER 15, TRANSIENT ANALYSIS METHODOLOGY SUBMITTAL - OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 (TAC NOS. M99349, M99350, AND M99351)

Dear Mr. McCollum:

By letter dated July 30, 1997, Duke Energy Corporation (DEC) submitted Topical Report DPC-NE-3005-P, "UFSAR [Updated Final Safety Analysis Report] Chapter 15 Transient Analysis Methodology," for NRC staff review and approval. Additional information was supplied by DEC by letter dated July 23, 1993. The topical report describes the methodology DEC used to analyze the nonloss-of-coolant accident (non-LOCA) UFSAR Chapter 15 transients and accidents for the Oconee Nuclear Station (Oconee), Units 1, 2, and 3. The objective of this topical report is to implement a revised non-LOCA transient and accident analysis methodology and establish a new licensing basis for Oconee. As you explained in the submittal, most of the computer codes and simulation models referenced in the report have been previously reviewed by the staff and approved for application to the Oconee reload design and in response to Generic Letter 83-11. This topical report is a specific application of these models to the analysis of the Oconee UFSAR Chapter 15 transients and accidents.

The staff has reviewed the topical report and finds it acceptable with some exceptions, as explained in the enclosed safety evaluation. These exceptions have been discussed with your staff and will be addressed in a revision to the topical report that will be submitted for staff review and approval. As a result, staff activity concerning this topical report is not completed and will be reinitiated using new TAC numbers when the revision is received.

However, some of the methodology described in this topical report is also important in certain design basis considerations for conversion of the current Oconee Technical Specifications (TS) to the improved TS. The submittal for this conversion is under staff review (TAC Nos. M99912,

M99913, and M99914). All of the methodology needed for this application has been found to be acceptable, as described in the enclosed safety evaluation. This action closes TAC Nos. M99349, M99350, and M99351.

Sincerely,

David E. LaBarge, Senior Project Manager
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Docket Nos. 50-269, 50-270, and 50-287

Enclosure: As stated

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO TOPICAL REPORT DPC-NE-3005-P

UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) CHAPTER 15

TRANSIENT ANALYSIS METHODOLOGY

DUKE ENERGY CORPORATION

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

1.0 INTRODUCTION

By letter dated July 30, 1997 (Reference 1), supplemented by letter dated July 23, 1998 (Reference 2), Duke Energy Corporation (Duke) submitted Topical Report DPC-NE-3005-P, "UFSAR Chapter 15 Transient Analysis Methodology," describing the methodology used by Duke for analyzing the nonloss-of-coolant accident (non-LOCA) UFSAR Chapter 15 transients and accidents for the Oconee Nuclear Station (ONS/Oconee), Units 1, 2, and 3. The objective of this topical report is to implement a revised non-LOCA transient and accident analysis methodology and establish a new licensing basis for ONS.

Duke received NRC approval to perform core reload design analyses for ONS in 1981. The set of licensing basis transients and accidents in the current UFSAR has essentially remained the same as in the original FSAR. However, future reload core designs will require reanalysis of the UFSAR Chapter 15 transients and accidents due to advanced fuel assembly designs, longer fuel cycles, increased steam generator tube plugging, and more efficient core designs. Therefore, Duke proposes to use the methodology described in this topical report to reanalyze the ONS Chapter 15 events in order to establish an up-to-date design basis, and to support advanced fuel assembly and core reload designs.

2.0 EVALUATION

The acceptability of Duke's nuclear and thermal-hydraulic analysis models and methods for simulating the ONS UFSAR Chapter 15 non-LOCA analyses is evaluated herein. Since many of these models and methods have previously been reviewed and approved by the NRC staff, the evaluation focused on any new models and methods, and on the specific application of the methods to the reanalysis of ONS transients and accidents.

2.1 RETRAN-02 One-Dimensional Kinetics Model

The RETRAN-02/MOD5.1 code was reviewed generically by the NRC and approved for use provided plant-specific methods have also been reviewed by the NRC (Reference 3). In this report (DPC-NE-3005-P), a new modeling application was made. A one-dimensional (1-D) kinetics model was used to model the core response for transients for which point kinetics does not provide sufficient results. The 1-D kinetics equations are derived from the neutron diffusion equation by assuming that the change in the radial neutron flux with time is relatively small. This model used the three-dimensional (3-D) nodal physics code, SIMULATE-3K (Reference 4), to generate the 1-D nuclear parameters (nuclear cross-sections and kinetics parameters) and a linking utility program (XGEN) to functionalize the nuclear parameters against the RETRAN-02 thermal-hydraulic feedback variables (moderator density and fuel temperature). The 1-D kinetics licensing basis analyses used the nuclear parameters (cross-sections and kinetics parameters) that would yield the same physics parameters (moderator temperature coefficient, Doppler coefficient, control rod worth, etc.), that were used in the licensing basis analyses using the point kinetics model. SIMULATE-3K was used in this limited application because of its ability to iteratively modify the nuclear parameters until the desired physics parameters are achieved. The resultant nuclear parameter modification factors were then used to generate the nuclear parameters for the RETRAN 1-D kinetics model.

Duke has presented comparisons of the RETRAN 1-D kinetics model with SIMULATE three-dimensional calculations. In general, the results indicate that with the 1-D nuclear data generated by this methodology, for any perturbation, the RETRAN 1-D kinetics model (1-D representation of the core) would predict a similar core response as the global response predicted by the SIMULATE 3-D representation of the core. The staff, therefore, finds this limited use of SIMULATE-3K acceptable. The SIMULATE-3K code was also used to calculate the effects of a control rod ejection accident, as discussed in Sections 2.5 and 2.6 below.

2.2 RETRAN-3D in RETRAN-02 Mode

The RETRAN-3D/MOD001F code is a recent version of the RETRAN code which has not been reviewed and approved by the NRC. Duke has submitted the code to the NRC for generic review in a separate licensing action. RETRAN-3D incorporates new models and equations, including additional balance equations to predict non-equilibrium phenomena and 3-D core kinetics, as well as advanced numerical solution schemes and correlations. However, in this report (DPE-NE-3005-P), Duke has not included any of the non-equilibrium or 3-D core modeling techniques. The application of RETRAN-3D was limited to the "RETRAN-02 mode" which is intended to replicate RETRAN-02. Only the advanced solution scheme and correlations of RETRAN-3D were utilized. This limited application of RETRAN-3D was used to analyze the startup accident, loss of flow, locked rotor, and turbine trip events and the results were compared to those obtained with the NRC-approved RETRAN-02 code. Based on the good agreement between the results of RETRAN-02 and RETRAN-3D in the RETRAN-02 mode for these transients discussed below, the staff concludes that this limited use of the RETRAN-3D code is acceptable.

2.3 VIPRE-01 Additional Features

The VIPRE-01/MOD2 code is used for steady-state and transient core thermal-hydraulic analyses and has been approved by the NRC for ONS licensing calculations (Reference 5). The version used in the DPC-NE-3005-P safety analyses is designated as VIPRE-01/MOD2F, and incorporates several additional features. However, the constitutive equations, correlations, and solution schemes of the VIPRE-01/MOD2 code have been preserved. The following changes were incorporated into VIPRE-01/MOD2F:

(1) The BWC, BWCMV, and BWU-Z critical heat flux (CHF) correlations were added.

The NRC Safety Evaluation Report for VIPRE-01 states that whenever Duke intends to use other CHF correlations, power distributions, fuel pin conduction models or any other input parameters and default options which were not part of the original review of the VIPRE-01 code, Duke must submit justification of these changes for NRC review and approval. The B&W BWC correlation with VIPRE-01 has been approved by the NRC with the design limit of 1.18 (Reference 6). Use of the BWCMV correlation has been approved by the NRC for use with VIPRE-01 with a design limit of 1.21 (Reference 7). Use of the BWU-Z correlation with a design minimum DNBR limit of 1.19 with the Mk B11V fuel design is currently under review by the NRC staff in a separate licensing action and preliminary indications are that it will be approved for use with VIPRE-01 (Reference 8). Therefore, the incorporation of the BWC, BWCMV, and BWU-Z CHF correlations into VIPRE-01/MOD2F is acceptable, provided these correlations are used over their approved ranges of applicability.

(2) An option to allow use of either a linear interpolation or a spline fit for the input nodal axial power profile was added.

The spline fit option was originally incorporated to replace inadequacies in the linear interpolation routine. Because of straight line interpolation from point to point, linear interpolation did not conserve area under the curve and therefore would tend to under predict the axial shape uniformly, which is nonconservative for departure from nucleate boiling (DNB) calculations. However, in order to be able to duplicate previous analyses that used linear interpolation, both options were incorporated. The staff finds this acceptable.

(3) An option to allow input of the power hot channel factor and the local heat flux hot channel factor to a subchannel for calculating the DNB ratio (DNBR) in that subchannel was added.

The local heat flux hot channel factor has not been applied in Oconee DNBR analyses beginning with the Oconee Unit 1 Cycle 14 reload. The power hot channel factor has been applied during the entire operating history of Oconee and is not new to the Oconee licensing basis. Since there is no difference being introduced in the proposed methodology, the option is acceptable.

(4) An enhanced iteration scheme was added.

The logic in the original VIPRE-01/MOD2 code did not always cause the iteration to converge when the input parameter yielded a minimum DNBR value significantly different from the target minimum DNBR limit and would at times also cause the iteration process to stall. Logic was therefore added to improve the iteration technique to circumvent these problems. The staff finds the enhanced iteration logic acceptable.

2.4 ARROTTA Code for Rod Ejection Analysis

The core neutronic response to a control rod ejection event was calculated with the Electric Power Research Institute ARROTTA code (Reference 9). ARROTTA is a 3-D, 2-energy group diffusion theory code which uses neutron cross sections, discontinuity factors, and 6 groups of delayed neutron precursor data generated with CASMO-3 (Reference 10). The ARROTTA time-dependent core power distribution is used as input to the subchannel core thermal-hydraulics analysis performed with VIPRE-01. VIPRE-01 calculates the fuel temperatures, the allowable power peaking to avoid exceeding the DNBR limit, and the core coolant expansion rate. The allowable power peaking is then used along with a post-ejected condition fuel pin census to determine the percentage of pins exceeding the DNB limit. The coolant expansion rate is input to a RETRAN-02 model of the reactor coolant system to determine the peak pressure resulting from the core power excursion. Duke has indicated that ARROTTA is only used for the rod ejection accident and, because of the rapid nature of this event, the neutronics solution rather than the moderator feedback effects are most important for this application. The NRC has approved the use of ARROTTA by Duke for rod ejection analysis for the McGuire and Catawba Nuclear Stations (Reference 11) and the use of ARROTTA in DPC-NE-3005-P is consistent with the previously approved methodology. Therefore, the staff concludes that ARROTTA is acceptable for the Duke analysis of the rod ejection event in Oconee.

2.5 SIMULATE-3K Code for Rod Ejection Analysis

The SIMULATE-3K code (Reference 4) was also used by Duke to calculate the core power response and three-dimensional power distribution resulting from a control rod ejection event. SIMULATE-3K is a three-dimensional transient neutronic version of the NRC-approved SIMULATE-3P code and utilizes the same neutron cross section library. It employs a fully implicit time integration of the neutron flux, delayed neutron precursors, and heat conduction models. Code validation was performed by the code vendor (Studsvik of America, Inc.) during development of SIMULATE-3K. The validation included benchmarks of the fuel conduction and thermal-hydraulic model, the transient neutronics model, and the coupled performance of the transient neutronics and thermal-hydraulic models. Duke comparisons of SIMULATE-3K with ARROTTA for the Oconee rod ejection analysis presented in DPC-NE-3005-P indicate very good agreement for core power versus time for the ejection occurring at end-of-cycle from the maximum allowable power level with three and four reactor coolant pumps (RCPs) operating and from both beginning-of-cycle and end-of-cycle hot zero power conditions. Although larger deviations occur for the ejection event initiated at beginning-of-cycle at the maximum allowable power level with three and four RCPs operating, the comparisons indicate that SIMULATE-3K results in more conservative values than ARROTTA. Therefore, the staff concludes that SIMULATE-3K is acceptable for the Duke analysis of the rod ejection event in Oconee.

2.6 Chapter 15 Safety Analyses

The core thermal-hydraulic analysis for most of the transients considered in this topical report is based on the NRC-approved statistical core design (SCD) methodology (Reference 12). This methodology includes uncertainties on the initial conditions of core average power, core inlet temperature, core exit pressure, and core inlet flow. Therefore, when performing an SCD analysis, initial condition uncertainties are not included in these four parameters as they are already included in the SCD DNBR limit. When non-DNB analyses are being performed, the uncertainties are included. For either type of analysis, the uncertainty in the timing of a particular action is accounted for by uncertainty-adjusting the actuation setpoints.

Reactivity and Power Distribution Anomalies

Six transients are considered in this category: (1) startup accident, (2) control rod withdrawal at power, (3) moderator dilution event, (4) cold water event, (5) control rod misalignment, and (6) control rod ejection accident.

The startup accident involves a reactivity addition due to an uncontrolled control rod withdrawal from a subcritical or low power condition. The current licensing basis for this event is that the reactor thermal power does not exceed 112 percent of rated thermal power (RTP) and reactor coolant system (RCS) pressure does not exceed code pressure limits. A high flux level and a high pressure trip are assumed. The proposed new acceptance criteria for this event are that the peak RCS pressure remains below 110 percent of the design pressure of 2500 pounds per square inch gauge (psig) and that no fuel failures result as demonstrated by not exceeding the DNBR limit. This is consistent with Section 15.4.1 of the NRC Standard Review Plan (SRP). The RETRAN-02 code is used to determine the peak transient primary system pressure. If the peak heat flux for the peak primary pressure analysis exceeds the maximum allowed steady-state value, the VIPRE-01 code is used to calculate the minimum DNBR for the transient using the SCD methodology. The core power distribution is analyzed with the SIMULATE-3P code. The event models three RCPs in operation and a maximum reactivity addition rate. Reactor trip is expected to occur on high pressure, high power, or the flux/flow imbalance trip functions.

The rod withdrawal event initiates from an accidental withdrawal of a control rod group while the reactor is at power. The current licensing basis is that the reactor thermal power does not exceed 112 percent of RTP and RCS pressure does not exceed code pressure limits. A high reactor coolant outlet temperature trip, a high reactor coolant system pressure trip, and a high power level (neutron flux level) trip are available to terminate this event. The proposed new acceptance criteria for this event require that the peak RCS pressure remains below 110 percent of design pressure and that the DNBR limit not be exceeded. This is consistent with Section 15.4.2 of the NRC SRP. The peak primary pressure case is analyzed with the RETRAN-02 code and VIPRE-01 is used to calculate the minimum DNBR. The core power distribution is analyzed with the SIMULATE-3P code. It is expected that the thermal-hydraulic conditions at the limiting DNB statepoint will be within the ranges covered by the SCD methodology. Reactor trip is expected to occur on high flux or high pressure, although the high coolant temperature and flux/flow/imbalance trips may also provide protection.

A RCS moderator dilution event occurs when the soluble boric acid concentration of makeup water supplied to the RCS is less than the concentration in the existing reactor coolant. Because Oconee's operating license was issued prior to the NRC SRP, the current licensing basis analyzes the event only from rated power (Mode 1) and refueling (Mode 6) conditions and requires that reactor thermal power not exceed 112 percent of RTP, RCS pressure not exceed code allowable limits, and a minimum shutdown margin of 1 percent Δk/k be maintained. The proposed new acceptance criteria are based on manual operator action at least 15 minutes during power operation and at least 30 minutes during refueling following actuation of the alarm credited for alerting the operator of a moderator dilution occurring. This ensures that the event is terminated by the operator before the DNBR limit or the peak primary pressure limit is violated and is consistent with Section 15.4.6 of the NRC SRP. For reload evaluations, the cycle-specific highest critical boron concentration and the initial boron concentration closest to the critical concentration are used.

The cold water accident is initiated with an inadvertent startup of an idle RCP, which causes a reduction in moderator temperature and a power excursion due to moderator feedback effects. The current licensing basis assumes the event is initiated from 50 percent power with two RCPs operating in one loop and two idle loops. This is no longer permitted by Oconee TS, which do not allow operation while critical with less than three RCPs in operation. Therefore, these analyzed conditions currently represent a bounding case relative to allowed operating conditions. The acceptance criteria would remain, i.e., minimum DNBR does not violate the acceptance criterion and system pressure limits (110 percent of design pressure) are not exceeded. This is consistent with Section 15.4.4 of the NRC SRP. The minimum DNBR would be determined using the SCD methodology. The proposed reanalysis of this event would assume that it is initiated with an inadvertent startup of a fourth RCP from an initial three-pump operating condition using the two-loop RETRAN-02 model.

The most limiting control rod misalignment event is the dropped rod since it presents the greatest potential for violating the minimum DNBR or system pressure limits. Although the event initially causes a rapid reduction in power and moderator temperature, the negative moderator temperature coefficient would subsequently result in a power increase. If the reactor is operating with the Integrated Control System (ICS) in automatic, rod withdrawal by the ICS will add to the increase in power. Although the analog ICS is being replaced by a digital ICS, the same modeling philosophy is retained in the analysis of this event. The transient response is analyzed with the RETRAN-02 code and the DNB analysis is performed with VIPRE-01 using the SCD methodology. The core power distribution is analyzed with the SIMULATE-3P code. The acceptance criteria remain that RCS pressure does not exceed 110 percent of design pressure and minimum DNBR does not violate the DNBR limit. This is consistent with Section 15.4.3 of the NRC SRP.

In addition to the dropped rod event, Duke has evaluated the misalignment event where a control rod is misaligned from the remainder of the rods in its bank. Since this may produce an increase in core peaking which decreases the margin to DNB, SIMULATE-3P was used to confirm that the asymmetric power distribution resulting from the rod misalignment will not result in DNB.

The rod ejection event is initiated by a failure of a control rod drive mechanism housing, which allows a control rod to be rapidly ejected from the core by the RCS pressure differential. The licensing basis criteria are that the accident will not further damage the RCS and that the offsite dose will be within the 10 CFR Part 100 limits. The first criterion is met by demonstrating that the peak fuel enthalpy remains below 280 cal/g. The proposed reanalysis would also require the peak primary pressure to remain within the Service Limit C as defined in Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (120 percent of the 2500 psig design pressure, or 3000 psig). The core power excursion is simulated with the ARROTTA or SIMULATE-3K 3-D transient code. VIPRE-01 is used to calculate the fuel temperatures, the allowable power peaking, and the coolant expansion rate. The allowable power peaking is used along with a post-ejection fuel pin census to determine the percentage of fuel pins exceeding the DNBR limit. All pins in DNB are assumed to experience clad failure for dose calculational purposes. The coolant expansion rate is input to RETRAN-02 to determine the peak pressure. As mentioned in Sections 2.4 and 2.5 above, ARROTTA and SIMULATE-3K are acceptable codes for analyzing the rod ejection accident in Oconee.

The staff concludes that the appropriate reactivity and power distribution anomalies will be reanalyzed using acceptable methods and that current licensing basis acceptance criteria will remain in effect or will be updated to conform to NRC SRP acceptance criteria.

Loss of Coolant Flow

The loss of coolant flow transient could be initiated by an electrical failure to the RCPs and result in one or more RCP coast downs. Either the pump monitor trip or the flux/flow/imbalance trip are used for reactor protection during this event. This event is classified as an event of moderate frequency. The acceptance criterion for this event is that no fuel failures will result as demonstrated by not exceeding the DNBR limit established at Oconee.

There are five bounding scenarios selected for the analysis of this transient including various numbers of RCP(s) coast downs from initial four or three RCP operation. RETRAN-02 and VIPRE-01 are used for thermal-hydraulic analyses and DNBR calculations. Initial and boundary conditions assumed in the analysis are conservatively selected for limiting consequences of the event. The results of the analysis confirmed that the transient minimum DNBR is approximately 1.68, which is sufficiently above the allowable minimum DNBR of 1.3 at Oconee. The staff finds this analysis methodology acceptable.

Locked Rotor

The locked rotor accident is the result of an instantaneous seizure of one RCP due to mechanical failure of the RCP. The analysis of this event assumes that the off-site power is available during this accident which is consistent with the current Oconee UFSAR. The flux/flow/imbalance trip is used for reactor protection during this event. The proposed licensee acceptance criteria include (1) fuel failure should be sufficiently limited to maintain core cooling capability, and (2) radiological consequences should be within the 10 CFR Part 100 guidelines. Since this event will also cause heatup of the RCS, in response to the staff request, the licensee has committed to include a peak RCS pressure acceptance criterion of 110 percent of

design pressure for the locked rotor accident (Reference 2). The results will be included in the UFSAR.

There are three scenarios analyzed for this event including one locked rotor from four pump operation and one locked rotor in the loop either with the idle RCP or without idle RCP from three-pump operation. RETRAN-02 and VIPRE-01 are used for thermal-hydraulic analysis and DNBR and maximum allowable radial peaking limits calculations. Initial and boundary conditions assumed in the analysis are conservatively selected for limiting consequences of the event. The results of this analysis indicate that there are no fuel failures and the acceptance criteria are met for this event. The staff finds the proposed analysis methodology acceptable.

Turbine Trip

The turbine trip transient can be initiated by a generator trip, low condenser vacuum, loss of lubrication oil, turbine trust bearing failure, turbine overspeed, or a manual trip. The turbine trip event could cause an increase in RCS temperature and pressure. The high RCS pressure trip is used for reactor protection during this event. This event is classified as an incident of moderate frequency. The acceptance criterion for this event is that the peak RCS pressure shall not exceed 110 percent of the design pressure. There is no DNB concern with this heatup transient.

The bounding scenario analyzed for the event is that of a turbine trip at full power under four RCP operating conditions. RETRAN-02 and VIPRE-01 are used to calculate the thermal-hydraulic responses. Initial and boundary conditions assumed in the analysis are conservatively selected for limiting consequences of the event. The results of this analysis confirm that the peak RCS pressure during the transient is within the acceptance criterion with sufficient margin. The staff finds that the method of analysis is acceptable.

Steam Generator Tube Rupture

The Steam Generator Tube Rupture (SGTR) analysis documented in the current Oconee UFSAR assumes no operator action at the beginning of the event and a low RCS pressure trip in about 8 minutes. The methodology proposed in the topical report assumes operator action to identify that a tube rupture has occurred and to manually trip the reactor in 20 minutes. Also, immediate action to maximize emergency core cooling system injection is needed.

The staff will require the licensee to modify the proposed methodology for the SGTR analysis to be consistent with the licensing bases established in the UFSAR. We will provide our safety evaluation of this item after the issue is resolved.

Large Steamline Break

The licensee is modifying the analysis methodology for this event. The new methodology will assume no main feedwater isolation during the event, which is consistent with the current Oconee UFSAR. The staff will provide its safety evaluation of this item after the issue is resolved.

The split reactor vessel modeling approach was approved by the NRC for the Oconee containment mass and energy release analysis methodology (Reference 13) and is similar to the method approved by the NRC for the McGuire and Catawba steamline break analysis (Reference 11). Although some differences exist to conservatively model the core response as compared to the mass and energy release, the staff finds the modeling acceptable for Oconee steamline break analyses.

Small Steamline Break

The licensee is modifying the analysis methodology for this event. The new methodology will assume no main feedwater isolation during the event. Also, an acceptance criterion will be added to require no fuel failure (no DNB) during this event since it is an incident of moderate frequency. The staff will provide its safety evaluation of this item after the issue is resolved.

3.0 CONCLUSIONS

Duke Topical Report DPC-NE-3005-P and its supporting documents, including the Duke responses to the NRC request for additional information, were reviewed to determine the acceptability of the revised non-LOCA transient and accident analysis methodology that will establish a new licensing basis to be used for future Oconee Chapter 15 analyses. Since most of the models and methods have been previously reviewed and approved by the NRC, the review focused on any new models and methods as well as on the specific application of the methods to the reanalysis of transients and accidents. Except for the following, the Duke methodology, as documented in DPC-NE-3005-P was found to be acceptable.

The licensee has committed to inclusion of a peak RCS pressure acceptance criterion of 110 percent of design pressure for the locked rotor event (Reference 2). The staff requires that the proposed SGTR methodology be modified to allow no operator action at the beginning of the event and initiation of a low RCS pressure trip in approximately 8 minutes to be consistent with the licensing bases in the UFSAR. The licensee is also modifying the proposed methodology for the large and small steamline break events to assume no main feedwater isolation during the events for consistency with the current Oconee UFSAR. In addition, since it is an incident of moderate frequency, an acceptance criterion will be added to the small steamline break analysis to require no fuel failures. The staff will provide safety evaluations of these items when the issues are resolved.

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Date: October 1, 1998

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Duke Power Company
Oconee Nuclear Station

UFSAR Chapter 15 Transient
Analysis Methodology

DPC-NE-3005-A
Revision 1

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Nuclear Engineering Division
Nuclear Generation Department
Duke Power Company

Abstract

This report describes the Duke Power Company methodology for simulating the UFSAR Chapter 15 transients and accidents for the Oconee Nuclear Station. The report includes details of the computer codes and models, methods for calculating safety analysis physics parameters and setpoints, and detailed modeling assumptions for all of the non-LOCA transients and accidents. The EPRI codes RETRAN-02, RETRAN-3D, VIPRE-01, and ARROTTA/1.10, and the Studsvik of America codes CASMO-3, SIMULATE-3P, and SIMULATE-3K are used for modeling the transient system and core thermal-hydraulic response, and the steady-state and transient core neutronic behavior. The dynamic reactor response is modeled using point, one-dimensional, and three-dimensional kinetics models, depending on the modeling requirements of each transient. This methodology will be used to reanalyze the Oconee UFSAR transients and accidents in order to establish an up-to-date design basis, and to support advanced fuel assembly and core reload designs.

UFSAR CHAPTER 15
TRANSIENT ANALYSIS METHODOLOGY

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1.0 INTRODUCTION AND SUMMARY

1.1 Overview

This report describes the methodologies to be used by Duke Power Company to perform the analyses of the UFSAR Chapter 15 non-LOCA transients and accidents for the Oconee Nuclear Station. The Oconee Nuclear Station is a three-unit, 2568 MWt pressurized water reactor of the Babcock &Wilcox (B&W) 177 fuel assembly lowered-loop design. Units 1 and 2 began commercial operation in 1973, and Unit 3 in 1974. Duke Power received NRC approval to perform core reload design analyses for Oconee in 1981, when the Safety Evaluation Report (SER) (Reference 1-1) for the "Reload Design Methodology" topical report NFS-1001 (Reference 1-2) was issued. The safety analysis methodology approved as part of the NFS-1001 topical report consists of a review of the key safety analysis physics parameters for each reload to confirm that the existing safety analyses in the UFSAR remain valid. These parameters originated primarily from the analyses performed by B&W during the original licensing of Oconee in the early 1970s. Most of these parameters have remained bounding during the history of the plant, thereby enabling the use of a review process rather than reanalysis. However, future reload core designs will require reanalysis of the UFSAR Chapter 15 transients and accidents due to advanced fuel assembly designs, longer fuel cycles, increased steam generator tube plugging, and more efficient core designs. In addition, the need for detailed knowledge of the licensing basis analyses in order to perform accurate and thorough safety reviews necessitates a reanalysis effort to update the 1970s vintage analyses in the UFSAR.

In September 1987 Duke Power submitted topical report DPC-NE-3000, "Thermal-Hydraulic Transient Analysis Methodology" (Reference 1-3) to address the requirements of NRC Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions" (Reference 1-4). This report describes the transient analysis simulation models and validation analyses for the Oconee, McGuire, and Catawba Nuclear Stations, using the Electric Power Research Institute's (EPRI) RETRAN-02 (Reference 1-5) and VIPRE-01 (Reference 1-6) computer codes. The McGuire/Catawba sections of DPC-NE-3000 received an SER from the NRC in November 1991 (Reference 1-7). The Oconee sections of DPC-NE-3000 received an SER from the NRC in August 1994 (Reference 1-8). Revision 1 to DPC-NE-3000 and its SER are References 1-30 and 1-9. Revision 2 to DPC-NE-3000 and its SER are References 1-31 and 1-32. The application of these models for UFSAR Chapter 15 non-LOCA analyses for McGuire

1-11. NRC approval of these methodologies were obtained in SERs dated November 15, 1991 (References 1-12 and 1-13). Additional related SERs are given in References 1-14 and 1-15. Submittal of DPC-NE-3005 completes the submittal of methodology reports for application to UFSAR Chapter 15 non-LOCA analyses for Oconee.

This report details the system thermal-hydraulic, core thermal-hydraulic, and nuclear analysis models and methods for simulating the Oconee UFSAR Chapter 15 non-LOCA analyses. Since most of these models and methods have been previously reviewed and approved by the NRC, the focus will be on any new models and methods, along with the specific application of the methods to the reanalysis of transients and accidents. The approach is very similar to that used by Duke Power, and approved by the NRC, for the McGuire and Catawba UFSAR Chapter 15 analyses. The main difference is in the set of transients and accidents comprising the Oconee UFSAR Chapter 15, due to the vintage of the plant. The set of licensing basis transients and accidents in the current UFSAR, which has remained the same as in the original FSAR, was used as a starting point for selecting which events to include in this topical report and in a future revision to the UFSAR. A review of these events has identified several which are not as limiting as similar events that are included in more modern UFSARs, or are not applicable under the current operating license. Duke Power has decided to replace several of the events in the current UFSAR Chapter 15 with similar, but more limiting or more applicable events. For example, the UFSAR Section 15.8 loss of load analysis will be replaced with a turbine trip analysis. This change is appropriate since the loss of load event from full power at Oconee will result in a reactor trip on high RCS pressure in a few seconds. The current UFSAR loss of load analysis assumes that a reactor trip will not occur, which is not true. In addition, a turbine trip at time zero bounds a loss of load which results in a reactor and turbine trip in a few seconds. With approval of this topical report the updated set of transients and accidents will be reanalyzed consistent with the models and assumptions detailed in this report, and the reanalyses will replace those currently in the UFSAR.

1.2 Computer Codes

Chapter 2 describes the computer codes and models used in the reanalysis of UFSAR Chapter 15 for Oconee. Each computer code is described along with the model and a summary of code and model validation. A brief summary of the codes and models is as follows:

RETRAN-02: The system thermal-hydraulic analysis uses the RETRAN-02/MOD5.1 code, which has been reviewed generically by the NRC (Reference 1-16) and approved for use provided plant-specific methods have also been submitted for review. Two RETRAN-02 modeling applications not included in DPC-NE-3000 are included in this topical report. One is the use of one-dimensional (1-D) kinetics for modeling the core response during transients. This modeling approach will be used for some transients for which the point kinetics model does not provide sufficient results. RETRAN 1-D kinetics has been approved by the NRC for use in BWR applications, but it is our understanding that this may be the first submittal for PWR application. The proposed RETRAN 1-D kinetics modeling is generic and will be applied to McGuire and Catawba also.

The second application not included in DPC-NE-3000 is split reactor vessel modeling for steam line break analysis. The steam line break modeling approach is similar to the method reviewed and approved by the NRC for application to McGuire and Catawba in DPC-NE-3001. There are some differences due to the loop configuration. This steam line break model was reviewed and approved by the NRC (Reference 1-17) for the Oconee containment mass and energy release analysis methodology DPC-NE-3003 (Reference 1-18). Some differences do exist due to the approach necessary to conservatively model the core response as compared to the mass and energy release.

RETRAN-3D: RETRAN-3D (Reference 1-19) will be submitted in the near future for generic NRC review and approval. For several of the UFSAR Chapter 15 reanalyses included in this report, results are also shown as predicted by RETRAN-3D. For these RETRAN-3D analyses, the code is executed in a mode which is intended to replicate RETRAN-02. The advanced models and equations of RETRAN-3D (e.g. the five equation model, flowing non-condensible gas capability, three-dimensional core kinetics) are not utilized in this report. What is utilized is RETRAN-3D's advanced solution scheme and correlations. By demonstrating that using RETRAN-3D in this "RETRAN-02 mode" gives virtually identical results, Duke Power requests NRC approval for this limited method of application of RETRAN-3D prior to the review by NRC of RETRAN-3D's advanced capabilities. Approval of this method of application of RETRAN-3D will greatly improve engineering productivity as well as enable successful application to problems which have proven difficult with RETRAN-02.

VIPRE-01: The core thermal-hydraulic and fuel pin analyses use VIPRE-01/MOD2, which has been reviewed generically by the NRC (Reference 1-20) and approved for use provided plant-specific methods have also been submitted for review. The Oconee VIPRE model used for most of the UFSAR Chapter 15 analyses was approved by the NRC in References 1-8, 1-9, and 1-32. This report includes additional VIPRE models required for specific analyses. The application of VIPRE-01 for Oconee Chapter 15 analyses closely follows that submitted by Duke and approved by the NRC for application to McGuire and Catawba.

CASMO-3: Nuclear constants are generated with the Studsvik of America code CASMO-3 (Reference 1-21). This code is used in Oconee reload design (Reference 1-22), and was approved by the NRC in Reference 1-23. CASMO-3 is also used in the McGuire and Catawba UFSAR Chapter 15 methodology. CASMO-3 is used for generating data used as input to the core models listed below.

SIMULATE-3P: Nuclear parameters and core power distributions are generated with the Studsvik of America code SIMULATE-3P (Reference 1-24). This code is used in Oconee reload design (Reference 1-22), and was approved by the NRC in Reference 1-23. SIMULATE-3P is also used in the McGuire and Catawba UFSAR Chapter 15 methodology.

ARROTTA/1.10: The EPRI code ARROTTA (Reference 1-25) is used for transient three-dimensional (3-D) modeling of the rod ejection accident. This code has been approved by the NRC for analysis of the rod ejection accident for Duke Power's McGuire and Catawba Nuclear Stations (Reference 1-12). The application for Oconee closely follows the methodology developed for McGuire/Catawba.

SIMULATE-3K: The Studsvik of America code SIMULATE-3K (Reference 1-26) is also used for transient 3D modeling of the rod ejection accident. SIMULATE-3K provides the same neutronics solution to steady-state 3-D calculations as SIMULATE-3P. Duke Power intends to use SIMULATE-3K as an equivalent code for any of the steady-state applications in this report that are stated as being analyzed with SIMULATE-3P. Additional features include the time-dependent equations necessary to solve transient 3-D problems. This is the first submittal of this version of the SIMULATE family of codes for NRC approval. It is Duke Power's intent to

transition from ARROTTA to SIMULATE-3K in order to reduce the engineering resources required to perform any 3-D transient analyses in the future.

1.3 Analysis Methodology

Chapter 3 describes the methods to calculate the safety analysis physics parameters which are generic to UFSAR Chapter 15 analyses. These inputs are calculated with the SIMULATE-3P code.

Chapter 4 describes the methodology for calculating the Reactor Protection System (RPS) and the Engineered Safeguards Protective System (ESPS) setpoints used in UFSAR Chapter 15 non-LOCA analyses. Determining initial conditions which incorporate allowances for parameter uncertainties is also discussed.

Chapters 5 through 16 describe the method of analysis for each UFSAR Chapter 15 non-LOCA transient and accident. Six of the transients and accidents are described in detail including a demonstration analysis and results. These six are the startup accident (UFSAR Section 15.2), the loss of coolant flow transient (UFSAR Section 15.6), the locked rotor accident (UFSAR Section 15.6), the turbine trip transient (replacing the loss of load transient in UFSAR Section 15.8), the rod ejection accident (UFSAR Section 15.12), and the steam line break accident (UFSAR Section 15.13). The six other events are described in terms of the analysis methodology and a results summary. These six are the rod withdrawal at power transient (UFSAR Section 15.3), the moderator dilution accident (UFSAR Section 15.4), the cold water accident (UFSAR Section 15.5), the control rod misalignment transients (UFSAR Section 15.7), the steam generator tube rupture accident (UFSAR Section 15.9), and the small steam line break transient. The small steam line break transient is not currently included in the UFSAR Section 15.13 steam line break analysis. Due to the different plant response to a small steam line break relative to a large steam line break, a separate analysis methodology and chapter for this event is included. The small steam line break transient was identified as being significantly different than the large steam line break transient in Reference 1-27. Reference 1-27 was submitted to the NRC in response to an NRC letter (Reference 1-28) requesting information on non-conservative high flux trip setpoints for certain transients in B&W plants. The NRC SER for this issue is Reference 1-29. Detailed analysis results for all transients and accidents will be included in the UFSAR. The purpose,

scope, and contents of this report were discussed in a meeting with the NRC staff on August 15, 1995. During this meeting the NRC staff indicated that the proposed contents were reasonable.

For each analysis methodology described in Chapters 5 through 16 the discussion includes acceptance criteria, model nodalization, initial conditions, boundary conditions, physics parameters, control systems, protection systems, safeguards systems, and operator response modeling. Additional discussion on the approach taken for some of these parts of the methodology are as follows:

Acceptance Criteria

Some of the acceptance criteria in the UFSAR have been updated, such as replacing the peak power level acceptance criterion with the industry standard criterion based on a DNBR limit. The updated acceptance criteria are typical of those used in the industry.

Model Nodalization

The RETRAN models and some VIPRE models (Reference 1-3) have been previously approved by the NRC (References 1-8, 1-9, 1-32). The specific application of these models for UFSAR Chapter 15 transient and accident reanalysis has required some changes to these approved models, and some additional VIPRE models. All of the changes and new models are described in this report. For example, the steam line break analysis requires a more detailed RETRAN nodalization of the reactor vessel and a special VIPRE model.

Initial Conditions

The initial conditions for many transient and accident analyses include allowances for uncertainty for parameters such as power, flow, pressure, and temperature. For analyses using the statistical core design approach some of the key initial condition uncertainties are already included in the statistical DNBR limit, and do not need to be included in the thermal-hydraulic analyses.

Credit for Control Systems and Non-Safety Components and Systems

Control systems are generally assumed to respond as designed or remain in manual control (inactive), whichever assumption is more conservative. Non-safety components and systems are generally not credited in the analyses. The following are specific exceptions to the general

modeling philosophy on control systems, and the situations where non-safety components and systems are credited in the analyses: 1) In the dropped rod event, the Integrated Control System will respond by initiating a plant runback to a reduced power level. Since this plant runback assists in the mitigation of the dropped rod event, no credit is taken for this control system design feature. This assumption is an additional conservatism that is not required by the methodology philosophy. 2) For a loss of all reactor coolant pumps without a loss of the Main Feedwater System, the Integrated Control System is credited for raising steam generator levels to the natural circulation setpoint. This design feature is implicitly credited in the loss of coolant flow event, and involves non-safety equipment. A failure of this design function would be mitigated manually by operator action to start the Emergency Feedwater (EFW) System. 3) The moderator dilution accident credits the non-safety high-flux-at-shutdown alarm and the control rod insertion limit alarm to alert the operator that a boron dilution event is in progress. Both of these alarms rely on non-safety equipment. The rod insertion alarm relies on the plant computer. 4) Many of the transient and accident analyses involve control rod movement. These analyses credit the normal withdrawal sequence, overlap, and rod speed, which are controlled by non-safety control systems. 5) For certain failures in the safety-grade EFW System, credit is taken for realigning EFW flow through the non-safety MFW System. This design aspect has been reviewed and approved by the NRC. 6) Steaming of the steam generators with manual non-safety atmospheric dump valves is credited. 7) The turbine trip circuitry has two channels, one with a one second response time, and one with a fifteen second response time. The faster response time is credited in the current UFSAR Chapter 15 analyses and will be credited in the methodology. A station modification is planned to upgrade the second channel to a one second response time. The turbine trip circuitry is not completely safety-grade. 8) The capability to remotely throttle certain valves is credited. Some of the controls required to remotely throttle these valves are not safety-grade. 9) Electrical bus voltage and frequency control are credited. These are controlled by non-safety components. 10) The steam generator secondary drains are credited in the SGTR analysis to control level in the ruptured steam generator to prevent overfill. These drains are not safety-grade.

Main Feedwater Isolation

The large and small steam line break analyses do not credit automatic isolation of main feedwater by the main steam line break detection and main feedwater isolation instrumentation.

Replacement Integrated Control System

The analyses presented in this report model the original analog Integrated Control System that has been replaced with a digital system. None of the analyses are adversely affected by this control system upgrade. Some of the UFSAR transient and accident analyses will be revised to incorporate the upgraded digital Integrated Control System. The modeling philosophy regarding control systems will remain the same in future reanalysis.

Single Failure and Loss of Offsite Power Assumptions

A limiting active single failure in the Reactor Protective System or in the Engineered Safeguards is assumed. A single failure in the Emergency Feedwater System is also considered. A failure of the manual atmospheric dump valves is not considered. Offsite power is assumed to be lost at time zero for those UFSAR Chapter 15 events which already include that assumption, which are limited to the steam line break accident.

For those transients and accidents for which detailed results are included, the results of each computer code used, the sequence of events, the plant response, and figures of key parameter trends are presented. The results are then compared to the applicable acceptance criteria. The process for evaluating each event for each reload core design is then stated. In general, the UFSAR analyses will remain valid as long as the key safety analysis parameters remain valid. Otherwise a reanalysis using revised and bounding parameters will be performed or the core will be redesigned.

1.4 Summary

The methodology presented in this report describes a conservative approach to performing the UFSAR Chapter 15 analyses for Oconee with modern thermal-hydraulic and nuclear analysis codes. These methods will be used to revise the existing UFSAR analyses which date to the early 1970s. The transient and accident analysis results presented are typical of those that will be used to update the UFSAR. Once implemented in the UFSAR, the revised analyses will enable a complete understanding of what the licensing basis analyses assume in terms of plant systems and component responses. This process will enhance the capability of Duke Power to review and assess plant operations and design in order to ensure compliance with regulations, to ensure consistency with Technical Specifications, and to ensure safe operation.

1.5 References

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2.0 SIMULATION CODES AND MODELS

2.1 RETRAN-02/MOD5.1

2.1.1 Code Description

The RETRAN-02/MOD5.1 DKE code is a modified version of the NRC-approved RETRAN-02/MOD005.1 code (References 2-1 and 2-2). The version used in this topical report differs from the NRC-approved version in that two error corrections to the MOD5.1 version have been inserted to obtain the MOD5.1DKE version. These two error corrections, which will be included in future versions of the RETRAN code are:

- Parameter LPOOL in subroutine INPUT was changed from 125,000 to 175,000. This modification was necessary to allow a large RETRAN input deck to execute.

- The DO statement in loop 920 of subroutine GENTRN is changed from "DO 920 IGS = 1,10" to "DO 920 IGS = 1,30". This correction enhances convergence in the generalized transport model.

Since the RETRAN version used is NRC-approved with the exception of error corrections, it is concluded that this report need not justify the validity of the RETRAN code itself. Therefore, details regarding the theory of the RETRAN code are left to the references.

2.1.2 Simulation Model

The Oconee RETRAN Model is documented in Reference 2-3. The nodalization diagram (Figure 2-1) is generally applicable for non-LOCA transient simulation. For certain transients, transient-specific modeling revisions and/or additions must be included. Details defining these modifications are included in each section of this report for each transient. Other than these modifications, the RETRAN model used for this report is the same as that reviewed by the NRC per Reference 2-3.

2.1.3 Validation of Code and Model

Validation of the RETRAN code and the Oconee RETRAN Model for transient analysis was reviewed and approved by the NRC per References 2-3 and 2-4. These validation efforts employed a range of plant transient data and demonstrated the capability of the methodology. Specific validation of certain transients (such as steam line break and locked rotor) in a B&W NSSS have not been performed due to an absence of applicable data. There have been no representative events in B&W plants, and very limited scaled test facility data. Consequently, no validation results are presented to support these particular transients. The validity and conservatism of the analyses for these transients are ensured by careful selection of initial and boundary conditions. These modeling details have been applied such that each acceptance criterion for each transient is evaluated in a conservative manner. Further discussion is provided in the relevant sections.

2.1.4 One-Dimensional Kinetics

This section describes the RETRAN-02 one-dimensional (1-D) kinetics model and the methodology used to generate the one-dimensional nuclear cross-sections and kinetics parameters required for input to the 1-D kinetics model. The 1-D kinetics model is based on the RETRAN-02 multiple control state model and can be used to perform core-wide transient simulations. The general approach taken is relatively similar to previous methodologies which employ the RETRAN-02 1-D kinetics model and that the NRC has found acceptable. This methodology uses the three-dimensional nodal physics code, SIMULATE-3K, to generate the 1-D nuclear parameters (nuclear cross-sections and kinetics parameters) and a linking utility program (XGEN) to functionalize the nuclear parameters against the RETRAN thermal-hydraulic feedback variables (moderator density and fuel temperature).

2.1.4.1 <u>Overview Of Reactor Kinetics</u>

In reactor kinetics analysis it is important to be able to predict the time behavior of the core neutron power level for a given change in neutron multiplication. The neutron multiplication change could be due to changes in moderator density, fuel temperature, control rods, boron concentration or any other mechanism caused by a system operational transient. In general

terms, reactor kinetics is concerned with the space-time behavior of the neutron population in the reactor core.

The reactor kinetics equations are derived from the time-dependent multi-group formulations of the neutron diffusion equations. The neutron source term includes both the prompt and delayed neutron source terms. In RETRAN, the most basic multi-group formulation of the neutron diffusion equations have been adopted, consisting of two energy groups that partition the neutron energy spectrum into the fast and thermal energy groups. Both the point and the 1-D kinetics models are derived from the same initial equations.

The point kinetics model equations are derived from the neutron diffusion equation by assuming that the spatial (axial and radial) dependence of the neutron flux does not change with time. This model can adequately predict the time behavior of the reactor core in transients where the core spatial neutron flux remains relatively constant. This model is relatively simple and easy to implement. The reactor kinetics response can be described with a few pre-computed coefficients, a set of kinetics parameters and a scram curve.

The 1-D kinetics equations are derived from the neutron diffusion equation by assuming that the radial dependence of the neutron flux does not change (or magnitude of change is relatively small) with time. RETRAN uses the space-time factorization methods to solve the quasi-linear partial differential diffusion theory equations associated with the 1-D kinetics model. The space-time factorization methods are based on the assumptions that the time- and space-dependence of the neutron flux can be separated into a time-dependent amplitude function and a more slowly varying or quasi-static shape function. The 1-D kinetics model permits improved simulation of the axial component of the spatial reactivity response during transients. Details of the 1-D kinetics model theory, equations and computational methodology can be found in Reference 2-1.

2.1.4.2 <u>The 1-D Kinetics Model</u>

In the 1-D kinetics model the reactor core (active fuel region) is axially divided into a finite number of neutronic regions. Also modeled are the top and bottom axial reflector regions. The model treats each region as a homogeneous mixture containing a set of spatially homogenized fast and thermal energy group nuclear parameters (cross-sections and kinetics parameters).

These nuclear parameters, in addition to others, have dependence on moderator density, fuel temperature and control conditions (presence/absence of control rods). During the course of a transient, when changes in moderator density, fuel temperature and control rod positions are considered, each region passes through virtually an infinite number of neutronic states. The nuclear parameters therefore need to be functionalized against these variables.

Changes in the nuclear parameters due to changes in the thermal-hydraulic conditions are modeled by functionalizing the parameters against the moderator density and fuel temperature using the thermal feedback model. Changes in the nuclear parameters due to control rod effects (presence/absence of control rods) are modeled by generating another set of nuclear parameters calculated at the identical thermal-hydraulic conditions, except with a different control rod configuration (control state) using the multiple control state control rod model. The changes due to control rod effects are separated from the other effects because the control rod motion represents an explicit, externally driven reactivity change.

Thermal Feedback Model

Coupling between the RETRAN thermal-hydraulic model and the 1-D kinetics model is through the moderator density and the fuel temperature. These thermal-hydraulic states determine the feedback to the neutronic calculations which in turn controls the core power and hence the heat load. As in most kinetics models the feedback is determined by any deviation from the initial thermal-hydraulic state (the initial conditions in the physics code are representative of the initial RETRAN conditions). The standard industry practice is to use the fractional change in moderator density and the change in square root of the fuel temperature to functionalize the nuclear parameters. The nuclear parameters are assumed to be represented by a general polynomial expression of the form:

$$Z(X1, X2) = \sum_{i=1}^{N1} \sum_{j=1}^{N2} C(i,j)\left[\left(X1^{i-1}\right)\left(X2^{j-1}\right)\right]$$

where: Z = 1-D nuclear parameter listed in Table 2-1

X1 = fractional change in moderator density => $[\rho(t) - \rho(0)] / \rho(0)$

X2 = change in fuel temperature => $\sqrt{T_f(t)} - \sqrt{T_f(0)}$

N1 = order of expansion of feedback quantity X1

N2 = order of expansion of feedback quantity X2

C(i,j) = user supplied coefficient of fit (generated by XGEN)

In the above form of the polynomial equation (where the independent variables X1 and X2 represent changes from the initial state condition) the expansion of the change in variables guarantees that the 1-D nuclear parameters of SIMULATE-3K and RETRAN are equal at time zero. The linking program XGEN performs the functionalization per the above equation. XGEN determines the order of the fit (N's) and the coefficients of the fit (C's) for each of the nuclear parameters in each neutronic region, energy group and control state. These nuclear parameter data are supplied to RETRAN in the appropriate format via an external data file (i.e. TAPE40). RETRAN provides the independent variables (X's) and computes the nuclear parameter at every time step.

Control Rod Model

An integral part of the nuclear parameters representation in the 1-D kinetics model is the method of modeling the control rods. To allow for a dynamic control rod representation for purposes such as a reactor scram, the nuclear parameters must vary over a wide range. This variation is accomplished by specifying both uncontrolled and controlled neutronic states for each of the nuclear parameters. An uncontrolled state is defined as a set of nuclear parameters that represent the absence of control rods, whereas, a controlled state is defined as a set of nuclear parameters that represent the presence of control rods. A controlled state may range from partial control with only some control rods present in the region to maximum control with all the control rods present in the region. When control rods are inserted into a region, the number of possible controlled states is large. To simplify the amount of data manipulation and computer time, the partial insertion of control rods into a region is assumed to be a linear combination of the given control states.

The 1-D kinetics model is based on the RETRAN multiple control state model. In this model the control rod model uses multiple sets of nuclear parameters - one for each control state. The effects of change in control rod position are represented via a different set of polynomials. Along with the nuclear parameter data generated by XGEN, the control rod fraction (one-dimensional representation of the axial control rod distribution used to generate the nuclear

parameters) for each of the neutronic regions is also included in TAPE40. For a situation where all the control rods are positioned out of the core only the all-rods-out (ARO) and the all-rods-in (ARI) neutronic states are required, for which the control fractions are 0.0 and 1.0, respectively for all the neutronic regions.

2.1.4.2.1 Methodology For Generation of the 1-D Nuclear Parameters

SIMULATE-3K is used to generate the spatially homogenized nuclear parameters suitable for the RETRAN 1-D kinetics model. The theory and equations solved by SIMULATE-3K are provided in Reference 2-5. SIMULATE-3K is an advanced 3-D nodal physics code with transient analysis capabilities. The transient analysis capabilities of SIMULATE-3K have been implemented within the existing framework of the steady state advanced 3-D nodal physics code, SIMULATE-3P (Reference 2-6). SIMULATE-3K provides the same steady state solution as SIMULATE-3P. The reason for selecting SIMULATE-3K is because of its capability to modify the nuclear parameters thereby affecting reactivity which is a useful application in licensing basis analyses where bounding reactivity feedback is modeled. The 1-D cross-sections and the diffusion coefficients produced by SIMULATE-3K preserve all the planar reaction rates, the planar interface currents and the core eigenvalue such that, as long as the solutions are converged, the results of the RETRAN 1-D kinetics model would approximate the global results (reactor power level, axial flux shape, etc.) of the SIMULATE-3K model.

SIMULATE-3K provides a convenient way to evaluate the global neutronic response to core perturbations. Perturbation calculations are made from the initial state that is representative of the conditions that could be encountered in the transient being analyzed. Perturbation calculations can be easily performed by separately perturbing either the moderator density or the fuel average temperature, with the other variables such as exposure, fission product inventory, etc. held constant. This is important in the functionalization process of the homogenized 1-D cross section data and kinetics parameters against the RETRAN thermal-hydraulic feedback variables. This functionalization will be performed by the utility program XGEN. The effect of control rods is incorporated into the 1-D kinetics model via a different set of cross-sections and nuclear parameters.

Overview of Methodology

The following steps are performed to generate the 1-D nuclear parameters required for the 1-D kinetics model:

1) SIMULATE-3K is used to characterize the core neutronics response and to calculate the nuclear parameters. Multiple SIMULATE-3K cases are executed. The first case is the base case representing the initial state conditions of the transient being analyzed. The subsequent SIMULATE-3K cases are the perturbations (perturbations in moderator density and fuel temperature - the RETRAN thermal-hydraulic feedback variables) cases.

2) If the transient involves a change in the control rod position (such as reactor trip) then another set of SIMULATE-3K cases needs to be executed at the new control rod configuration. The base case and the perturbation cases are executed at the identical thermal-hydraulic conditions as the previous set of cases in (i).

3) The nuclear parameters generated by each of the SIMULATE-3K cases (the base case and each perturbation case) are written to a "kinetics file" (one file is generated for each control state). This file is accessed by the linking utility program, XGEN, which performs the functionalization of the nuclear parameters.

4) The RETRAN "TAPE40" cross section data file is then created per instructions in the RETRAN user manual (Reference 2-1).

Adjustments Applied To Nuclear Parameters

The methodology adopted for performing licensing basis analyses using the 1-D kinetics model is to utilize the nuclear parameters (cross-sections and kinetics parameters) that would yield the same conservative physics parameters (moderator temperature coefficient, Doppler temperature coefficient, control rod worth, βeff, etc.) used in the licensing basis analyses using the point kinetics model. SIMULATE-3K has a convenient way of modifying the nuclear parameters. The nuclear parameters are iteratively modified until the desired physics parameters are achieved. The resultant nuclear parameter modification factors are then used to generate the nuclear parameters for the RETRAN 1-D kinetics model.

2.1.4.2.2 XGEN Utility Program

The XGEN utility program performs the functionalization of the one-dimensional homogenized cross sections and kinetics parameters and creates the "TAPE40" data file required to execute RETRAN using the 1-D kinetics model. XGEN processes one control state (SIMULATE-3K kinetics file) at a time. For each control state the code performs the following major functions:

1) For each case within the SIMULATE-3K kinetics file, the code reads the one-dimensional homogenized values of each of the nuclear parameters and the associated thermal-hydraulics data (moderator density and fuel temperature) for each energy group in each neutronic region, including the axial reflectors.

2) It is an industry practice to use the normalized fractional change in moderator density and change in square root of the fuel temperature as independent parameters to functionalize the nuclear parameters. Therefore, this program calculates the normalized fractional change in moderator density and the change in square root of the fuel temperature from the base case.

3) For each of the nuclear parameters in each energy group and each neutronic region, the code sets the order of "polynomial fit", determines the polynomial coefficients using the least-square fitting technique by fitting each nuclear parameter against the normalized fractional change in moderator density and change in square root of the fuel temperature.

4) The code will also echo to the output file details of the "polynomial fit", such as the actual independent values (the thermal feedback variables), the actual dependent values (nuclear parameters), the dependent values as calculated by the polynomial equation, the absolute error and the relative percent error. A message will be echoed when any relative percent error is greater than ±0.1%.

2.1.4.3 Methodology Verification

The objective of this methodology is to generate the one-dimensional nuclear data such that for any perturbation the RETRAN 1-D kinetics model (1-D representation of the core) would predict a similar core response as the global response predicted by the SIMULATE-3P or SIMULATE-3K core models (3-D representation of the core). To assess the degree to which this objective is met a [

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2.2 RETRAN-3D/MOD001F

2.2.1 Code Description

The RETRAN-3D/MOD001F code (Reference 2-7) is a recent version of the EPRI RETRAN family of codes. RETRAN-3D was developed to provide analysis capabilities for LWR transients, small-break loss-of-coolant accidents, anticipated transients without scram, natural circulation events, long-term transients, and transients with thermodynamic non-equilibrium phenomena. New fully implicit numerical solution schemes have been provided for both fluid states, including additional balance equations to better predict non-equilibrium phenomena. The code user has the ability to run RETRAN-3D in several modes. The application of RETRAN-3D in this report is limited to the "RETRAN-02 mode", which does not include any of the non-equilibrium, flowing non-condensible gas capability, or three-dimensional core modeling unique to RETRAN-3D. The advanced solution scheme and correlations are used in the analyses in this report.

2.2.2 Simulation Model

The Oconee RETRAN Model discussed in Section 2.1.2 is used with the RETRAN-3D/MOD001F code. Appropriate input deck changes are made to account for the relocation of certain parameters in the RETRAN-3D input deck, including additional input requirements. The transient-specific modeling revisions/additions required for each transient modeled with RETRAN-3D are the same as those made for the RETRAN-02 analysis. Other than these modifications, the RETRAN model used for this report is the same as that reviewed by the NRC per Reference 2-3.

2.2.3 Validation of Code and Model

Validation of the Oconee RETRAN Model for transient analysis is documented in References 2-3 and 2-4 for use with the RETRAN-02 code. Four of the transients simulated with RETRAN-02 in this topical report are also simulated with the RETRAN-3D code. The RETRAN-3D code is run in the "RETRAN-02 mode" to show that application of RETRAN-3D gives very similar results to the NRC-approved RETRAN-02 code. Generic validation of RETRAN-3D was performed by EPRI and is documented in the RETRAN-3D manuals (Reference 2-7).

2.3 <u>VIPRE-01/MOD2</u>

2.3.1 Code Description

The VIPRE-01 code (Reference 2-8) is used for the reactor core thermal-hydraulic analyses. VIPRE-01 is a subchannel thermal-hydraulic computer code developed for EPRI by Battelle Pacific Northwest Laboratories (BPNL). The VIPRE-01 code has been reviewed by the NRC and was found to be acceptable for referencing in licensing applications (Reference 2-9). With the subchannel analysis approach, the nuclear fuel element is divided into a number of quasi one-dimensional channels that communicate laterally by diversion crossflow and turbulent mixing. Given the geometry of the reactor core and coolant channel, and the boundary conditions or forcing functions, VIPRE-01 calculates the core flow distribution, coolant conditions, fuel rod temperatures (if the conduction model is utilized) and the departure from nucleate boiling ratio (DNBR) for steady-state and transient conditions. VIPRE-01 accepts all necessary boundary conditions that originate either from a system transient simulation code such as RETRAN, or a

different transient inlet temperatures, flow rates, heat flux transients, and even different transient assembly and pin radial powers or axial flux shapes can be modeled.

The version of the VIPRE-01 code currently used in the analyses is a Duke version of VIPRE-01/MOD2. The Duke version of the code includes additional features and editorial changes so that the constitutive equations, correlations, and solution schemes of the VIPRE-01/MOD2 code have been preserved. These additional features and editorial changes are described below:

- Add the following critical heat flux (CHF) correlations:
    1. BWC CHF correlation
    2. BWCMV CHF correlation
    3. BWU-Z and BWU-N CHF correlations
- Add the ability to print the friction, form loss elevation, acceleration and cross flow pressure drops for specified channels
- Add the option to allow the user to use either a linear interpolation or spline fit for the input nodal axial power profile
- Add the option to generate a summary file of the minimum DNBR value
- Add the option to allow the user to input the power hot channel factor (F_q) and the local heat flux hot channel factor (F_q'') to a subchannel in order to conservatively calculate the DNBR in that subchannel
- Enhance the logic used when VIPRE-01 is utilized to iterate on a parameter, such as radial power, to converge to a MDNBR limit

2.3.2 Simulation Models

The NRC has approved the VIPRE-01 models described in DPC-NE-3000 (Reference 2-3) for Oconee core thermal-hydraulic analyses. The simplified [ ] channel model described in Section 2.3.2.1 of the same reference will be utilized for the transients requiring a DNBR evaluation. The [ ] channel model (Figure 2-4) is constructed such that [

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[ ]. The justification for using this simplified [ ] channel model is given in DPC-NE-3000. The thermal-hydraulic modeling techniques and correlations utilized are also consistent with DPC-NE-3000. The [

] is used as a boundary condition.

In addition to the [ ] channel model described above, a [ ] channel model (Figure 2-5) is constructed for the VIPRE-01 analysis. This [ ] channel model simulates the thermal-hydraulic conditions in [ ] in the reactor core, and will be utilized for the transients requiring a [ ] calculation. In the [ ] channel model, the [

] channel model. A similar model has also been approved and utilized for the McGuire and Catawba [ ] analysis as described in Section 4.2.2.3 of Reference 2-10.

A [ ] channel model that simulates a [ ] is also constructed for two specific VIPRE-01 analyses. This [ ] channel model will be utilized for the rod ejection [ ] calculation and for some transient DNBR calculations as described in Chapter 14 of this report. The special SLB VIPRE model is described in Section 15.2.2.

2.3.3 Validation of Code and Model

In DPC-NE-3000, the validation of the VIPRE-01 code is performed by comparing the steady-state and transient results with COBRA-IIIC/MIT (Reference 2-11). The basic structure and computational philosophy of the VIPRE-01 code are derived from COBRA-IIIC (Reference 2-12). Therefore, it is appropriate to compare the steady-state as well as transient results calculated by these two codes. An identical COBRA [ ] channel model was constructed for the comparison purpose. Sections 2.3.5.4 and 2.3.5.5 of DPC-NE-3000 show that VIPRE-01 and COBRA-IIIC/MIT [ ] channel models generate essentially identical MDNBR and thermal-hydraulic property results for different steady-state operating conditions and during transients. The simplified [ ] channel model is also validated in DPC-NE-3000 by performing sensitivity studies (Reference 2-13). These sensitivity studies include the radial noding sensitivity, axial

The simplified [ ] channel model is also validated in DPC-NE-3000 by performing sensitivity studies (Reference 2-13). These sensitivity studies include the radial noding sensitivity, axial noding sensitivity, and void correlation sensitivity for both steady-state and transient calculations. Results of the sensitivity studies show that the [ ] channel model generates conservative minimum DNBR and local thermal-hydraulic conditions for both steady-state and transient analyses.

2.4 CASMO-3

CASMO-3 is a multigroup, two dimensional transport theory code for burnup calculations on PWR or BWR fuel assemblies. The code models a geometry consisting of cylindrical fuel rods of varying composition in a square pitch array with allowance for fuel rods loaded with integral burnable absorber, lumped burnable absorber rods, clustered discrete control rods, incore instrument channels, assembly guide tubes, and intra-assembly water gaps. The program utilizes a 40 energy group cross section library based on ENDF/B-IV with some data taken from ENDF/B-V. Two energy group edits of cross sections, assembly discontinuity factors, fission product data, and pin power data are produced for input to ARROTTA, SIMULATE-3P, and SIMULATE-3K core models. Reference 2-14 provides a detailed description of the theory and equations solved by CASMO-3. The use of CASMO-3 in this report is consistent with the previously approved methodologies of References 2-10 and 2-15.

2.5 SIMULATE-3P

SIMULATE-3P is a three-dimensional, two energy group, diffusion theory core simulator program which explicitly models the baffle and reflector regions of the reactor. Homogenized cross sections and discontinuity factors developed with CASMO-3 are used on a course mesh nodal basis to solve the two group diffusion equations using the QPANDA neutronics model. A nodal thermal hydraulics model is incorporated to provide both fuel and moderator temperature feedback effects. Inter- and intra-assembly information from the course mesh solution is then utilized along with the pinwise assembly lattice data from CASMO-3 to reconstitute pin-by-pin power distributions in two and three dimensions. The program performs a macroscopic depletion of fuel with microscopic depletion of iodine, xenon, promethium, and samarium fission products. Reference 2-6 provides a detailed description of the theory and equations solved by SIMULATE-

3P. The use of SIMULATE-3P in this report is consistent with the previously approved methodologies of References 2-10 and 2-15.

2.6 ARROTTA/1.10

ARROTTA is a three-dimensional, two energy group diffusion theory core simulator applicable for both static and transient kinetics simulations. Homogenized cross sections, discontinuity factors, and 6 groups of delayed neutron precursor data are generated with CASMO-3 and used on a course mesh nodal basis to solve the two energy group diffusion equations using the QPANDA neutronics model. The thermal-hydraulic model is comprised of both fluid dynamics and heat transfer models. Reference 2-16 provides a detailed description of the theory and equations solved by ARROTTA. The use of ARROTTA in this report is consistent with the previously approved methodology documented in Reference 2-10.

2.7 SIMULATE-3K

2.7.1 Code Description

The SIMULATE-3K code is a three-dimensional transient neutronic version of the SIMULATE-3P code (Reference 2-5). SIMULATE-3K uses the QPANDA full two-group nodal spatial model developed in SIMULATE-3P, with the addition of six delayed neutron groups. The program employs a fully-implicit time integration of the neutron flux, delayed neutron precursor, and heat conduction models. A calculation of the adjoint flux solution is performed to provide an accurate value of beta for the time-varying neutron flux. The control of time step size may be determined either as an automated feature of the program or by user input. Use of the automated feature allows the program to utilize larger time steps (which may be restricted to a maximum size based on user input) at times when the neutronics are changing slowly and smaller time steps when the neutronics are changing rapidly.

Additional capability is provided in the form of modeling a reactor trip. The trip may be initiated at a specific time in the transient or following a specified excore detector response. Use of the excore detector response model to initiate the trip allows the user to specify the response of

individual detectors as required to initiate the trip, as well as the time delay prior to release of the control rods. The velocity of the control rod movement is also controlled by user input.

The SIMULATE-3K thermal-hydraulic model may include a spatial heat conduction and a hydraulic channel model. The heat conduction model solves the conduction equation on a multi-region mesh in cylindrical coordinates. Temperature-dependent values may be employed for the heat capacity, thermal conductivity, and gap conductances. A single characteristic pin conduction calculation is performed per fuel assembly, with an optional calculation of the peak pin behavior available to monitor local maxima. A single characteristic hydraulic channel calculation is performed per fuel assembly. The model allows for direct moderator heating at the option of the user. This thermal-hydraulic model is used to determine fuel and moderator temperatures for updating the cross-section model, and may additionally be used to provide edits of fuel temperature throughout the transient.

The SIMULATE-3K program utilizes the same cross-section library and reads the same restart file (exposure and burnup-related information) as SIMULATE-3P. Executed in the static mode, SIMULATE-3K performs the same solution techniques, pin power reconstruction, and cross-section development as SIMULATE-3P. Additional features of SIMULATE-3K include the application of conservatisms through simple user input. Also, the inlet thermal-hydraulic conditions can be provided on a time dependent basis through user input.

2.7.2 Code Validation

Several benchmarks were performed by the code vendor (Studsvik of America, Inc.) during development of SIMULATE-3K. These benchmarks and results are described in the SIMULATE-3K manual (Reference 2-5). The fuel conduction and thermal-hydraulics model have been benchmarked against the TRAC code (Reference 2-17). The transient neutronics model has been benchmarked, using standard LWR problems, to reference solutions generated by QUANDRY (Reference 2-18), SPANDEX (Reference 2-19), NEM (Reference 2-20), and CUBBOX (Reference 2-21). Finally, a benchmark of the coupled performance of the transient neutronics and thermal-hydraulic models was provided by comparison of results from a standard NEACRP rod ejection problem to the PANTHER code (Reference 2-22). Steady-state components of the SIMULATE-3K model are implemented consistent with the CASMO-

2.8 References

2-1 RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Revision 4, EPRI, November 1988

2-2 Letter, M. J. Virgilio (NRC) to C. R. Lehman (PP&L), April 12, 1994 (SER for RETRAN-02/MOD 5.1)

2-3 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000, Duke Power Company, July 1987

2-4 Letter, L. A. Wiens (NRC) to M. S. Tuckman (Duke), August 8, 1994 (SER for DPC-NE-3000 - ONS sections)

2-5 SIMULATE-3 Kinetics Theory And Model Description, SOA-96/26, Studsvik of America, April 1996

2-6 SIMULATE-3: Advanced Three-Dimensional Two-Group Reactor Analysis Code, Studsvik/SOA-92/01, Studsvik of America, April 1992

2-7 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-7450, Revision 3, EPRI, September 1998

2-8 VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores; EPRI NP-2511-CCM, Revision 3, EPRI, August 1989

2-9 Letter from C. E. Rossi (NRC) to J. A. Blaisdell (UGRA), "Acceptance for Referencing of Licensing Topical Report, VIPRE-01: A Thermal-Hydraulic Analysis Code for Reactor Cores," EPRI-NP-2511-CCM, Vol. 1-5, May 1986

2-10 Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology, DPC-NE-3001, Duke Power Company, January 1990

2-10 Multidimensional Reactor Transients and Safety Analysis Physics Parameters Methodology, DPC-NE-3001, Duke Power Company, January 1990

2-11 COBRA IIIC/MIT Computer Code Manual, MIT, March 1976

2-12 COBRA IIIC: A digital Computer Program for Steady-State and Transient Core Thermal-Hydraulic Design Methodology, BNWL-1695, PNL, March 1973

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2-15 Nuclear Design Methodology Using CASMO-3 / SIMULATE-3P, DPC-NE-1004, Duke Power Company, November 1992

2-16 ARROTTA: Advanced Rapid Reactor Operational Transient Analysis, EPRI, August 1993

2-17 TRAC-BF1/MOD1, Vol. 1, NUREG/CR-4356, J. Borkowski and N. Wade (Editors), May 1992

2-18 QUANDRY: An Analytical Nodal Method for Solving the Two-Group, Multidimensional, Static and Transient Nodal Diffusion Equations, K. S. Smith, Massachusetts Institute of Technology, 1979

2-19 Development of a Variable Time-Step Transient NEM Code: SPANDEX, Trans. Am. Nucl. Soc. 68- 425, B. N. Aviles, 1993

2-20 NEM: A Three Dimensional Transient Neutronics Routine for the TRAC-PF1 Reactor Thermal Hydraulic Computer Code, B. R. Bandini, Pennsylvania State University, 1990

2-21 CUBBOX: Coarse-Mesh Nodal Diffusion Method for the Analysis of Space-Time Effects In Large Light Water Reactors, S. Langenbuch, W. Maurer, and W. Werner, Nucl. Sci. Eng. 63-437, 1977

2-22 NEACRP 3-D LWR Core Transient Benchmark, NEACRP-L-335, H. Finneman and A. Galati, January 1992

Table 2-1

Two Group Nuclear Parameters

Nuclear Parameter	Symbol
Total delayed neutron fraction	β
Group 1 absorption cross-section	$\Sigma_{a,1}$
Group 1 radial buckling factor	$B^2{}_1$
Group 1 diffusion coefficient	D_1
Group 1 down-scatter cross-section	Σ_s
Kappa * group 1 fission cross-section	$\kappa\Sigma_{f,1}$
Nu * group 1 fission cross-section	$\nu\Sigma_{f,1}$
Group 1 neutron velocity	V_1
Group 2 absorption cross-section	$\Sigma_{a,2}$
Group 2 radial buckling factor	$B^2{}_2$
Group 2 diffusion coefficient	D_2
Kappa * group 2 fission cross-section	$\kappa\Sigma_{f,2}$
Nu * group 2 fission cross-section	$\nu\Sigma_{f,2}$
Group 2 neutron velocity	V_2

Figure 2-1

Figure 2-2

Figure 2-3

Figure 2-4

VIPRE-01 [ ] Channel Model

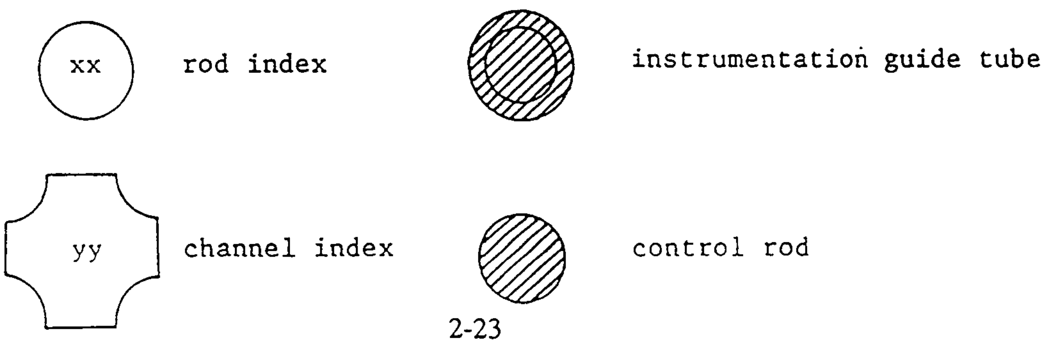


Figure 2-5

VIPRE-01 [ ] Channel Model

xx channel index

yy rod index

3.0 SAFETY ANALYSIS PHYSICS PARAMETERS

UFSAR Chapter 15 transients and accidents must be conservatively analyzed to ensure that the applicable fuel design limits, system overpressure design limits, and dose consequences are not exceeded. Each transient and accident analysis incorporates a set of assumptions, which when combined in a consistent or conservative manner, produce conservative analysis results. These analyses bound the licensed operating conditions and modes for the current plant design and fuel cycle. An important subset of the analysis assumptions includes the core physics parameters necessary to characterize the initial conditions and transient response of the core. The relative importance of various physics parameters and the sensitivity to variations in the values of the parameters varies between transients. However, it is possible to identify for each event a set of physics parameters which are significant and directly affect the results of the analysis. Once these key parameters have been determined, then the impact of variation in the range of values due to a change in the core loading pattern and operating history can be assessed. A conservative or consistent value can then be selected for analysis, or several combinations can be analyzed to ensure the transient response is bounded.

The purpose of this chapter is to review and identify the key physics parameters for each UFSAR Chapter 15 event. The conservative direction for each parameter (e.g. minimum/maximum) is identified where important. Table 3-1 summarizes the key parameters identified in this chapter.

Core physics parameters are calculated as part of the safety analysis for each reload core using NRC-approved methodology to systematically confirm that the physics parameters for a reload core are bounded. Three-dimensional core models such as SIMULATE-3P (Reference 3-1) are used to calculate core physics parameters and power distributions. The models used to perform these calculations are based on the available operating history of the previous reload cycle to assure best estimate calculations. Determination of whether a nuclear-related physics parameter is within the bounding value assumed in the reference safety analysis must be made by performing explicit calculations of the parameter, or by comparison to values generated in previous reload core designs. Comparison to previously calculated physics parameters (to determine if the physics parameter is bounding) is only performed if the reload core being analyzed is similar to previously analyzed reload cores. These comparisons can be performed to determine the bounding nature of a physics parameter because of the predictable behavior of

most physics parameters as a function of reactor power, moderator temperature, burnup, and soluble boron concentration. The parameters are described in the following sections.

3.1 Generic Parameters

Some of the important safety analysis physics parameters can be considered generic in that the value of the parameter is important for many transient analyses. The following are descriptions of the generic parameters.

Reactivity Insertion Following Reactor Trip

The reactivity insertion following reactor trip is a combination of a minimum available tripped rod worth and a normalized reactivity insertion rate. The minimum available tripped rod worth assumed in safety analyses must ensure, as a minimum, that the shutdown margin in Technical Specifications is preserved. This shutdown margin assumes that the most reactive rod remains in the fully withdrawn position and that the other control rods drop from their power dependent insertion limits. The normalized reactivity insertion rate is determined by bounding control rod drop times as determined by plant testing, and by developing a conservative relationship between rod position (% withdrawn) and normalized reactivity worth.

Initial Core Power Distribution

Technical Specifications implicitly require that the core power distribution remains within prescribed limits during power operation, based on explicit operating limits on measurable parameters. The actual operating limits are based on power level, flux imbalance, control rod position , and core power tilt. The 3-D core power distribution is limited by analytical methodologies which relate the measurable parameters to simulated power distributions. These power peaking limits are typically expressed as limits on total peak (F_Q) and radial peak ($F_{\Delta H}$). F_Q limits are typically a function of elevation in the core and might also vary as a function of burnup and power level. The transient and accident analyses assume that any core power distribution permitted within normal operating limits is a valid initial condition. For those transients in which the initial power distribution has a significant impact on the course of the event, perturbed power distributions allowed by operating limits are considered.

Power Distribution At Limiting Transient Statepoint

Many of the transients for which the initial core power distribution is a key parameter must also be evaluated for the power distribution at the limiting transient statepoint. The limiting transient statepoint is usually the time of the transient at which the DNBR reaches its minimum value. This evaluation is necessary when the core power distribution changes from the initial power distribution due to the effects of the transient on the core conditions. These effects can include both changes in moderation resulting from the thermal-hydraulic transient, and changes due to control rod movements.

Effective Delayed Neutron Fractions and Decay Constants

The dynamic behavior of the reactor core is determined to a large degree by the presence of delayed neutrons. The delayed neutron parameters are mainly important during rapid reactivity excursion transients. Delayed neutron fractions and decay constants are calculated for six effective delayed neutron groups. The total beta-effective is the sum of the six group effective fractions and is calculated at BOC and EOC conditions. The values of the fractions and decay constants for each delayed neutron precursor group are not key parameters, and typical values are sufficient.

Prompt Neutron Lifetime

The prompt neutron lifetime is mainly important during rapid reactivity excursion transients. This parameter is not a key parameter, and so typical beginning and end-of-cycle values are used consistent with the limiting core condition for the transient.

3.2 Control Rod Worths

The primary purpose of control rods is to provide adequate shutdown capability during normal plant operation and accident conditions. Control rods are also used to maintain criticality during rapid reactivity changes such as those that occur during typical load follow maneuvers. They can also be used to offset reactivity changes produced from fuel depletion and changes in boron concentration, xenon concentration, and moderator temperature. However, control rods are maintained at or near their all rods out (ARO) position during full power operation and are normally only used to compensate for rapid reactivity changes.

Control rod integral and differential rod worths are sensitive to local and global power distribution changes. Since the placement of fresh and depleted fuel assemblies produces unique power distributions, it is necessary to analyze control rod worths for each reload core. Rod worth related calculations that are evaluated for each reload core are:

- Shutdown margin
- Trip reactivity
- Control rod insertion limits
- Maximum differential rod withdrawal at power
- Maximum differential rod withdrawal from subcritical
- Dropped rod worth
- Ejected rod worth
- Stuck rod worth

Shutdown Margin

Shutdown margin calculations are typically performed for each reload core at beginning-of-cycle (BOC) and end of cycle (EOC) at various power levels including hot full power (HFP) and hot zero power (HZP) conditions. These calculations are typically performed in three dimensions, taking into account the power defect, stuck rod worth, allowance for rods being at their power dependent insertion limits, and rod worth uncertainty.

Control Rod Insertion Limits

Control rod insertion limits serve several functions and are dependent upon the acceptable results of the power peaking analyses, shutdown margin calculations, ejected rod worth calculations, and inserted reactivity assumptions for safety analyses. Verification of the rod insertion limits from a peaking standpoint is performed in the operating limits and RPS setpoint analysis performed for each reload core design. Rod insertion limits also impact the available shutdown margin by influencing the magnitude of the rod insertion allowance. The rod insertion allowance is calculated at various burnups. Rod insertion limits also impact the ejected rod worth and the amount of worth available for withdrawal for accidents sensitive to this parameter.

Maximum Differential Rod Withdrawal from Power

The maximum differential rod worth at power is calculated for each reload core at BOC and EOC. This calculation is performed to assure that inputs to the uncontrolled bank withdrawal at power accident are bounded. The maximum differential rod worth of any two control banks is calculated assuming normal overlap, while adhering to the power dependent rod insertion limits.

Maximum Differential Rod Withdrawal from Subcritical

The maximum differential rod worth from subcritical is calculated for each reload core at BOC and EOC. This calculation is performed to ensure that inputs to the startup accident are bounded. The calculation of this parameter assumes that control banks move in 25% overlap with the reactor at HZP.

Dropped Rod Worth

The maximum allowed dropped rod worth is calculated at both BOC and EOC. This value is compared against the reference analysis value to ensure that the safety analysis remains bounding. Dropped rod worths are calculated by evaluating the reactivity difference produced from a control rod dropped from the HFP ARO condition.

Ejected Rod Worth

Ejected rod worths are calculated at BOC and EOC for both HZP and HFP conditions. Initial conditions for the ejected rod worth calculation are established by assuming that the control rods are at their rod insertion limit. The rod worth calculation is performed by ejecting the control rod from the rod insertion limit to the ARO condition and calculating the reactivity difference. All possible rods are analyzed to determine the highest worth ejected rod.

3.3 Reactivity Coefficients and Kinetics Parameters

The dynamic behavior of a reactor core during load following maneuvers, transients, and accident conditions can be described in terms of reactivity coefficients. The magnitude and sign of these coefficients affect the reactor stability during transient and accident conditions. Reactivity coefficients are defined as the change in reactivity produced from a change in reactor power, moderator density, fuel temperature or boron concentration. The moderator density effects are often expressed in terms of moderator temperature. Since these coefficients are a

strong function of exposure, they are calculated at several exposure statepoints during core life. Reactivity coefficients are also influenced by changes in moderator temperature, reactor power, and soluble boron concentration.

The statepoints at which reactivity coefficients are evaluated are chosen to ensure that the assumptions made in the specific accident analysis remain bounded. For example, the moderator dilution accident at power is sensitive to the least negative moderator coefficient and the steam line break accident is sensitive to the most negative isothermal temperature coefficient. The calculation of the moderator temperature coefficient, the Doppler temperature coefficient, and the statepoints at which these coefficients are evaluated, are discussed below. The calculation of critical boron concentrations, boron worths and kinetics parameters follow.

Moderator Temperature Coefficient

The moderator temperature coefficient (MTC) is defined as the change in core reactivity resulting from a change in moderator temperature. Bounding coefficients (least and most negative) are calculated for each reload core. The following parameters are considered in the evaluation of the MTC to ensure that conservative results are obtained.

- Soluble boron
- Cycle exposure
- Control rods
- Moderator temperature

The MTC is calculated by inducing a change in moderator temperature (and therefore, density) and dividing the resulting reactivity change by the change in moderator temperature.

Doppler Temperature Coefficient

The Doppler temperature coefficient is defined as the change in core reactivity resulting from a change in fuel temperature. The least and most negative Doppler temperature coefficients are calculated for each reload core considering the core burnup and power level. The Doppler temperature coefficient is calculated by performing a set of two cases which vary the fuel temperature. The reactivity difference between the two fuel temperatures divided by the change in fuel temperature is the definition of the Doppler temperature coefficient. Doppler temperature

coefficients are often quoted at various power levels by equating changes in reactor power to changes in mean fuel temperature.

Critical Boron Concentrations and Boron Worths

Critical and shutdown boron concentrations are calculated as a function of reactor power, exposure, temperature, and control rod positions as allowed by the power dependent rod insertion limits. Differential boron worths are also calculated as a function of various combinations of the above variables. The results of these calculations are compared to inputs for several accident analyses.

3.4 Reload Cycle Specific Evaluation

The important physics parameters in Table 3-1 will be evaluated each reload cycle to ensure that values assumed in the current licensing analyses bound the reload core. Accidents for which the physics parameters are not bounded would be re-evaluated to ensure acceptable accident consequences or the core would be redesigned to obtain acceptable results.

3.5 References

3-1 Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004A, Duke Power Co., November 1992.

Table 3-1

Summary of Safety Analysis Physics Parameters

Report Section	Transient or Accident	FSAR Section	Key Parameters	Conservative Direction
3.0	Generic	N/A	• Reactivity insertion following reactor trip	- Minimum worth less stuck rod not to exceed the 1% Δk/k shutdown margin - Slowest insertion
			• Initial core power distribution	- Maximum power peaking per Tech Specs
			• Power distribution at limiting transient statepoint	- Maximum power peaking
			• Effective delayed neutron fraction and decay constants	- Minimum for rapid reactivity transient - Maximum for all other transients - Nominal precursor group fractions and decay constants
5.0	Startup Accident	15.2	• MTC • DTC • Reactivity insertion rate	- Most positive - Least negative - Maximum
6.0	Rod Withdrawal at Power	15.3	• MTC • DTC • Reactivity insertion rate	- Most positive/least negative - Least negative - Minimum and Maximum
7.0	Moderator Dilution Accident	15.4	• Critical boron concentration • Initial boron concentration	- Highest - Closest to critical concentration
8.0	Cold Water Accident	15.5	• MTC • DTC	- Most negative - Least negative
9.0	Loss of Coolant Flow	15.6	• MTC • DTC	- Most positive - Least negative

Table 3-1 (continued)

Report Section	Transient or Accident	FSAR Section	Key Parameters	Conservative Direction
10.0	Locked Rotor	15.6	• MTC	- Most positive/least negative
			• DTC	- Least negative
11.0	Control Rod Misalignment	15.7	• MTC	- Bounding vs. burnup
			• DTC	- Bounding vs. burnup
			• Maximum available Group 7 rod worth curve for withdrawal	- Maximum
12.0	Turbine Trip	15.8	• MTC	- Most positive
			• DTC	- Least negative
13.0	Steam Generator Tube Rupture	15.9	• Boron worth	- Minimum
14.0	Rod Ejection	15.12	• MTC	- Most positive/least negative
			• DTC	- Least negative
			• Ejected rod worth	- Maximum
15.0	Steam Line Break (offsite power available)	15.13	• MTC	- Most negative
			• DTC	- Most negative
			• ECCS boron concentration	- Minimum
			• Boron worth	- Minimum
	Steam Line Break (offsite power not available)	15.13	• MTC	- Least negative
			• DTC	- Least negative

Table 3-1 (continued)

Report Section	Transient or Accident	FSAR Section	Key Parameters	Conservative Direction
16.0	Small Steam Line Break	15.17	• MTC • DTC	- Bound BOC to EOC - Least negative

4.0 SAFETY ANALYSIS SETPOINT METHODOLOGY

4.1 Overview

Oconee UFSAR Chapter 15 analyses introduce conservatism by assuming a bounding conservative value for various key initial conditions. These key initial conditions are chosen to bound the expected operating range for that particular parameter. Once the bounding initial condition value is determined, an uncertainty is conservatively applied to give an uncertainty-adjusted initial condition. This leads to differentiating between a true, or actual, value of an initial condition (uncertainty-adjusted) and the indicated value (non-uncertainty-adjusted). The indicated value is important since it determines when a particular safety or control feature is actuated. The actual value is important since it determines, or is input to a calculation that determines, the success of the transient. As a transient progresses, a particular indicated parameter, or parameters, will approach an actuation setpoint, either a Reactor Protective System (RPS), Engineered Safeguards Protective System (ESPS), or some auxiliary control function. These actuation setpoints are also uncertainty-adjusted such that an earlier or later actuation is achieved, depending on the conservative timing of the actuating function.

By differentiating between an initial condition uncertainty and a setpoint uncertainty, both DNB and non-DNB analyses can be simulated with the same methodology. DNB analyses are typically performed using the statistical core design (SCD) methodology (Reference 4-1). This methodology was approved by the NRC in Reference 4-2. The SCD methodology includes uncertainties on the following initial conditions: core average power, core inlet temperature, core exit pressure, and core inlet flow. Therefore, when performing an SCD analysis, initial condition uncertainties are not included in those parameters as they are already included in the SCD DNBR limit. When non-DNB analyses are being performed, the uncertainties are included. For either type of analysis (DNB or non-DNB), the uncertainty in the timing of a particular action is accounted for by uncertainty-adjusting the actuation setpoints. The combination of the initial condition value along with the initial and setpoint uncertainty adjustments leads to an overall conservative system response for a given transient analysis.

4.2 Initial Condition Uncertainties

Oconee operates with nominal setpoints for various parameters bounded by a specified range. These ranges are usually prescribed by the UFSAR Chapter 15 analyses and can be included in various licensing documents. These are best characterized as initial condition ranges. Some parameters, like RCS average temperature, are not allowed to deviate from their nominal setpoint, while others, like RCS pressure, are allowed to deviate a small amount, and still others, like RCS flow, cannot be controlled at all but still must lie within the prescribed initial condition ranges. A table of the key parameters and their current ranges is included in Table 4-1.

The key parameters, with the exception of core bypass flow and core average fuel temperature, are process indications in the control room. As long as the indicated value lies within the prescribed range, no action is required. The value of the key parameter at the start of the transient is termed the initial condition. It is therefore important that the initial condition lie within the initial condition range. About each indicated parameter lies the actual value of that parameter. The amount by which the actual value deviates from the indicated value is termed the initial condition uncertainty. If the indicated value is controlled to a nominal setpoint or within the initial condition range, then the safety analyses must consider that the actual value of that parameter may be different and may exceed that setpoint or range. The SCD includes the initial condition uncertainty for core average power, core inlet temperature, core exit pressure, and core inlet flow. Some of these (core inlet temperature, core inlet flow, and core exit pressure) are not directly measurable. However, for each of these parameters, there is a measurable key system parameter. This methodology makes the assumption that if the key system parameter is chosen conservatively, then the non-measurable parameter is also conservative.

The initial condition uncertainties are calculated with the square root sum-of-the-squares (SRSS) method. The total uncertainty includes the instrument uncertainty, the uncertainties in the electronic components that make up the signal processing string, and the uncertainty in any device that supplies a readout of that parameter. The total uncertainty also includes any biases that may characterize a particular component. Not included are uncertainties in the reactor trip module or actuation setpoint drift. Controller deadbands are typically not included in the uncertainty but are rather included in the initial condition ranges.

4.3 RPS and ESPS Setpoints

The RPS and ESPS systems are relied upon to mitigate the consequences of licensing basis transients and accidents. It is therefore important to represent these systems in a conservative manner. Since this methodology separates the initial condition uncertainty from the setpoint drift and reactor trip module uncertainty, it is straightforward to derive an uncertainty-adjusted setpoint. If there is only one uncertainty to include in the setpoint, it is a straight algebraic process. If there are two or more and the uncertainties are independent of each other, the SRSS method is employed. In this manner, the setpoint is either increased or decreased in the conservative direction and the timing of the actuation is conservatively accounted for. This is true whether an SCD analysis is being performed or not. The various current actuation setpoints, both nominal and uncertainty-adjusted, are included in Table 4-2.

Additionally, the high flux trip presents a special situation in that the effects of changes in reactor vessel downcomer water temperature and control rod movements on the excore flux indications are not included in the adjusted setpoint. This methodology necessitates inclusion of these effects in the nuclear instrumentation (NI) signal that is compared to the uncertainty adjusted setpoint. The NI signals are therefore adjusted for each transient for which these effects are important.

4.4 Methodology Application

The general application of the methodology is described in this section. Typically, it is conservative to delay safety system actuation as long as possible. To maximize the time to actuation on a particular parameter, the indicated value is chosen to be at the edge of the indicated parameter range that will maximize this time. If a non-SCD analysis is being performed, the actual value of that particular parameter is adjusted by the initial condition uncertainty and is adjusted in the conservative direction. If an SCD analysis is being performed, then no initial condition uncertainty is applied to those parameters that are statistically combined in the SCD DNBR limit. The uncertainty-adjusted actuation setpoint is then determined by applying the setpoint uncertainty to the setpoint value in the conservative direction (both SCD and non-SCD). As the transient progresses, the indicated value will approach the uncertainty-adjusted actuation setpoint and the actual value will lead or lag the indicated value, depending on

the direction of the initial condition uncertainty. Once the indicated value reaches the setpoint, the safety feature is actuated. It is noted that the timing of this actuation is independent of whether an SCD analysis is being performed or not. However, the actual value of the particular parameter at the time of actuation does depend on whether it is an SCD analysis. If it is an SCD analysis, the initial conditions uncertainty is included in the DNBR limit and does not need to be explicitly included in the DNB analysis. Thus, margin is gained due to the SCD while conservatism is maintained through the timing of the actuation. If it is a non-SCD analysis, then the actual parameter will lie in the conservative direction and conservatism is maintained in that both the timing of the actuation and the actual value of that parameter are both conservatively modeled.

The above discussion considers that only one parameter is important to the results of the analysis. This method is applied to all key parameters to ensure that the overall system response is conservative. If a particular parameter will not actuate a safety function, then the initial condition and its uncertainty are chosen to yield a conservative result.

4.5 Summary

The setpoint methodology outlined in this chapter is both flexible and conservative. It is flexible in that it allows both SCD and non-SCD analyses to be performed, and it is conservative in that it differentiates and explicitly accounts for uncertainties on initial conditions and actuation setpoints. This methodology simulates the initial condition ranges observed during normal plant operations and appropriately accounts for normal operational deviations in the various initial conditions. It differentiates between the actual value and the indicated value of a particular parameter such that the timing of a particular actuation is modeled conservatively while at the same time, maintaining conservatism in the magnitude of the actual value for that particular parameter.

Table 4-1
Current Initial Condition Ranges and Uncertainties

Key Parameter	Nominal Value	Initial Condition Range	Initial Condition Uncertainty
Reactor Power #	100 %FP (4 RCP) 75 %FP (3 RCP)	98 - 100 %FP 78 - 80 %FP ***	± 2 %RTP
RCS Avg. Temperature #	579 °F for ≥ 15 %FP 532 °F at HZP	578 - 580 °F 527 - 537 °F	± 1 °F + 3 °F, - 2 °F
NR RCS Pressure #	2155 psig	2125 - 2155 psig	± 30 psig
RCS Flow #	~112 %df (4 RCP) 0.747 x 112 %df (3 RCP) 0.49 x 112 %df (2 RCP)	107.5 - 115 %df 74.7 % of 4 RCP range 49 % of 4 RCP range	[ ]
Core Bypass Flow	5.3 %	5 - 7 %	NA **
Pressurizer Level	220 inches	220 inches	± 25 inches
Core Avg. Fuel Temperature	NA **	1075 - 1250 °F (BOC) 950 - 1150 °F (EOC)	NA **
SG Level	~ 60 %OR (HFP) ~ 69/19 %OR (2/1 RCP) 25 inches XSUR MFW at HZP	55 - 98 %OR 67-98/18-38 %OR 25 inches XSUR MFW	+ 3 %OR, - 5 %OR + 3 %OR, - 5 %OR ± 12.1 inches
SG Pressure	910 psig	This value is chosen to maintain the proper system heat balance	Included in any safety system actuation setpoints
MFW Flow	~1500 lbm/sec/SG	This value is chosen to maintain the proper system heat balance	Included in any safety system actuation setpoints
CFT Inventory	1040 ft^3	1070 - 1010 ft^3	- 40 ft^3 ****
CFT Pressure	600 psig	625 - 575 psig	- 25 psi ****

\# Indicates that the initial condition uncertainty in this parameter is included in the SCD.

* [ ]%df uncertainty applicable for 1 pump in each loop.[ ]%df uncertainty applicable for 2 pumps in one loop and zero pumps in the other loop.

** NA denotes that this value is not applicable since there is not one value that is either repeatable (nominal fuel temperature) or measured (uncertainties).

*** The initial condition range for the 3 RCP UFSAR Chapter 15 analyses assumes that the initial indicated power level is 78-80 % FP

**** These values are for steam line break and not LOCA

(Table 4-1 cont.)

NOTE: The above values are current values and may change in the future. If future values lie outside the analyzed values, then the UFSAR Chapter 15 analyses will be re-evaluated or reanalyzed as appropriate.

Actual values can be derived from the above values by adding or subtracting the initial condition uncertainty from the upper or lower bound on the initial condition range, depending on the conservative direction of the indicated and actual values.

Table 4-2
Current RPS and ESPS Actuation Setpoints

Safety Function	Nominal Setpoint	Uncertainty Adjusted Setpoint	Maximum Response Time *
RPS:			
High Flux	105.5 %FP	106.5 %FP	0.4 sec
High Pressure	2355 psig	2362 psig	0.5 sec
Low Pressure	1800 psig	1793 psig	0.5 sec
Variable Low Pressure-Temperature	trip if: (P is psig) P<11.14(Thot)-4706	trip if: (P is psig) P<11.14(Thot)-4716	0.7 sec
Flux/Flow	trip if: ϕ>109.4 x Fm **	trip if: ϕ>109.4 x Fm + 2.2 %FP	1.2 sec
High Temperature	618 °F	618.85 °F	0.7 sec
Pump Monitor	NA	NA	0.6 sec
ESPS:			
HPI	1590 psig	1480 psig # 1400 psig ##	15 sec (no-LOOP) 38 sec (LOOP)
CFT ***	2.0 psid	+ 6.5 psid (CFT A) ### - 2.5 psid (CFT B)	NA

* Note that the RPS trip response times include a minimum 0.14 sec delay for the control rod gripper coils to de-energize.

** Fm is measured flow. 109.4 is in units %FP/flow

*** The nominal actuation setpoint is based on a ΔP across the CFT check valve. The RETRAN values assume the nominal value, but account for the RETRAN modeling to obtain the nominal ΔP. Since this is a passive safety system, no response times are applicable.

\# Includes 50 psi uncertainty and 60 psi margin

\## For large steam line break only due to harsh containment environment allowance

\### These setpoints are adjusted to account for elevation differences

4.6 References

4-1 Thermal-Hydraulic Statistical Core Design Methodology, DPC-NE-2005P-A, Duke Power Company, September 1992

4-2 Letter, G. M. Holahan (NRC) to H. B. Tucker (DPC), February 24, 1995 (SER for DPC-NE-2005P-A)

5.0 STARTUP ACCIDENT

5.1 Overview

5.1.1 Description

The objective of a normal startup is to bring a subcritical reactor to the critical or slightly supercritical condition, and then increase power in a controlled manner until the desired power level and system operating temperature are obtained. During a startup, an uncontrolled reactivity addition could cause a nuclear power excursion. Since the heat removal capability of the secondary is not increased during the excursion, the resultant power mismatch would cause an increase in the primary system temperatures and pressures. Because of the relatively short duration of the power excursion, the effect on the secondary system pressure and temperature would be minimal. The rod motion and core temperature feedback would also cause the core power peaking to change. With the change in core power peaking and the changes in system thermal-hydraulics, a departure from nucleate boiling (DNB) condition could occur.

5.1.2 Acceptance Criteria

The acceptance criteria for the startup accident are that the peak RCS pressure remains below 110% of the design pressure of 2500 psig, and that no fuel failures will result as demonstrated by not exceeding the DNBR limit.

5.1.3 Analytical Approach

The startup accident requires a limiting set of physics parameters to be determined for use as initial and boundary conditions. These parameters are input to the Oconee RETRAN-02 (Reference 5-1) model for the system thermal-hydraulic analysis. The RETRAN-02 analysis generates the peak transient primary system pressure, and the transient core boundary conditions for detailed core DNB modeling. If the peak heat flux for the peak primary pressure analysis remains below the maximum allowed steady-state value, then DNB will not be of concern, and detailed core thermal-hydraulic modeling using the VIPRE-01 (Reference 5-2) code will not be necessary. This conclusion includes the consideration of changes in the core radial power

distribution during the rod group withdrawal. Otherwise, the Oconee VIPRE-01 model will be used to calculate a set of maximum allowable radial peaking (MARP) curves as core power peaking limits such that DNB will not occur. The MARP curves will be compared against SIMULATE-3P core power distributions to determine if any of the fuel rods exceed the DNBR limit.

5.2 Simulation Codes and Models

5.2.1 RETRAN-02

The Oconee two-loop base model described in Reference 5-3 serves as the basis of the RETRAN-02 model utilized in this analysis. Since the impact of the number of operating reactor coolant pumps is to be examined, a RETRAN model that represents the four cold legs is desired. The RETRAN-02 model that is used in this analysis is shown in Figure 5-1. The two loop base model is modified such that the [

]

During three-pump operation, the loop with two active RCPs is at a slightly lower hot leg pressure than the loop with one active RCP. As a result, if the pressurizer is attached to the loop with two active RCPs, the pressurizer safety valves (PSVs) will lift at a later time. Delaying PSV lift is conservative with respect to maximizing the RCS pressure. Therefore, the pressurizer is assumed to be attached to the loop with two RCPs in operation.

5.2.2 VIPRE-01

Should a DNB analysis become necessary, the VIPRE-01 code is used to calculate the minimum DNBR for the startup accident. VIPRE thermal-hydraulic boundary conditions (core heat flux, core inlet flow, core inlet temperature and core exit pressure) are obtained from the RETRAN simulation. The [ ] channel VIPRE model described in Section 2.3 of Reference 5-3 is used to calculate the limiting statepoint local properties and DNBR. The VIPRE analysis will employ the SCD methodology for the startup accident.

5.2.3 SIMULATE-3P

SIMULATE-3P is a core neutronics code used to generate safety analysis physics parameters and three-dimensional core pin power distributions for the startup accident. The conservatism of the physics parameters will be confirmed each cycle as described in Section 5.4 SIMULATE-3P will also be used to calculate the pin power distributions for the accident conditions if a DNBR analysis is necessary. The pin power distributions will then be used to determine if any fuel failures occur.

5.3 Peak Primary System Pressure and Core Cooling Capability Analysis

The startup accident analysis presented herein is concerned with maximizing the core heat flux, which therefore maximizes the RCS pressure response. If the predicted heat flux for the peak RCS pressure analysis does not exceed the allowable steady-state heat flux for three-pump operation, then DNB is not of concern for this event. Otherwise a VIPRE-01 analysis is performed to calculate the minimum DNBR.

5.3.1 Initial Conditions

Power Level

An initial critical power level of 1E-9 of the nominal full power level is assumed. This very low initial power level maximizes the power excursion.

RCS Pressure

A high initial pressure is generally conservative for peak RCS pressure analyses. However, a low initial indicated pressure will maximize the time from the start of the transient until a high pressure reactor trip actuates. This will maximize the power level if the trip occurs on high pressure, and thus will maximize the peak pressure. Sensitivity studies have determined that a low initial indicated RCS pressure is the most conservative assumption.

Pressurizer Level

A high initial pressurizer level minimizes the volume of the pressurizer steam bubble. This maximizes the pressure increase following an insurge (i.e., heatup), which is conservative for peak RCS pressure analyses.

RCS Temperature

Initial RCS temperature is not expected to be an important parameter in the peak RCS pressure analysis, and thus a nominal HZP average temperature of 532 °F is assumed.

RCS Flow

A low initial flow will maximize the core heatup, and thereby maximize the positive reactivity inserted by an assumed BOC most positive MTC. Maximizing the positive reactivity insertion will maximize the core heat flux, and thereby maximize the RCS pressure response. Preliminary analyses indicated that acceptable results could not be obtained with an assumed two-pump initial condition. Therefore, the analysis will assume a conservatively low three RCP initial flow condition.

Core Bypass Flow

A high core bypass flow is assumed. This is an important parameter for the DNB analysis since it minimizes the core flow. It also maximizes the positive reactivity insertion and core heat flux, thereby maximizing the RCS pressure response

Fuel Temperature

A high fuel rod gap conductivity will maximize the heat transfer (i.e., heat flux) from the fuel into the coolant during the power excursion. This is conservative with respect to maximizing the RCS pressure response. Fuel rod gap conductivity is maximized by assuming low initial fuel

temperatures. Lower transient fuel temperatures will insert less Doppler feedback, which will maximize the power excursion. At HZP, the initial fuel temperature is equal to the coolant temperature. Gap conductivity is known to increase as power increases. Therefore, it is conservative to use low fuel temperatures at an assumed high power level in determining an appropriate gap conductivity.

Steam Generator Mass

Sensitivity studies have determined that initial SG mass is not an important parameter for the startup accident. This is due to the rapid nature of the power excursion, which does not allow enough time for much primary-to-secondary heat transfer during the time frame of interest.

Steam Generator Pressure

The initial SG pressure is determined by the initialization such that the SG saturation temperature is approximately equal to the initial HZP RCS temperature.

Steam Generator Tube Plugging

A high SG tube plugging percentage will degrade any primary-to-secondary heat transfer. A high tube plugging percentage will also minimize the RCS inventory. Both of these effects will maximize the RCS pressure increase. Therefore, a conservatively high SG tube plugging level is modeled.

5.3.2 Boundary Conditions

Control Rod Group Withdrawal Rate and Worth

The Oconee control rod configuration consists of eight groups. Banks 1-4 are the safety groups, and are fully withdrawn during power operation and during an approach to criticality. Groups 5-7 are the control groups, and are partially withdrawn in sequence to approach criticality after the RCS has been diluted to an estimated critical boron concentration. After reaching criticality, Groups 5-7 are further withdrawn to increase power. Group 8 rods are part-length rods used for axial power distribution control.

The analysis will assume that Groups 1-4 are fully withdrawn at the start of the event. The maximum withdrawal rate (i.e., reactivity insertion rate) for any of Groups 5-7 is examined. The

rods are assumed to be moving along the steepest part of the integral rod worth curve, with nominal assumptions for control rod speed, overlap and withdrawal sequence. The maximum withdrawal rate will result in the largest core power excursion, which will maximize the heat flux into the reactor coolant, and thereby maximize the coolant expansion.

Pressurizer Safety Valves

The pressurizer code safety valves are modeled with conservative assumptions for drift, blowdown and relief capacity such that the RCS pressure is maximized during the transient.

Pressurizer Inter-Region Heat Transfer Coefficient

For this analysis, a conservatively low pressurizer inter-region heat transfer coefficient is assumed. This will maximize the rate of RCS pressurization and worsen the approach to the high pressure acceptance criterion.

Secondary Heat Removal

Steam generator heat removal does not have a significant effect on the transient RCS temperature and pressure. This is due to the short duration of the transient which does not allow enough time for the heat added to be transported to the steam generator during the time frame of interest. For this reason, secondary side heat removal is not modeled in detail. No feedwater is modeled. No main steam dump or relief capacity is modeled.[

]

Single Failure

The loop with two reactor coolant pumps in operation will indicate a lower hot leg pressure than the loop with only one active reactor coolant pump. Therefore, the analysis assumes a single failure of one of the narrow range pressure channels on the loop with only one active RCP. This requires the high pressure reactor trip to be generated by the loop with the lower RCS pressure, which is conservative for a later reactor trip.

5.3.3 Physics Parameters

Moderator Temperature Coefficient

The peak core power response is achieved using BOC feedback parameters. At BOC the moderator temperature coefficient is most positive. A table of reactivity as a function of moderator density is input to account for moderator reactivity effects.

Doppler Temperature Coefficient

At BOC the Doppler temperature coefficient is least negative. A table of reactivity as a function of fuel temperature is input to model Doppler reactivity effects.

Beta-effective and Neutron Lifetime

The effective delayed neutron fraction (β_{eff}) is also largest at BOC. A large β_{eff} will result in a slower power decrease upon reactor trip for a given rod worth. A large prompt neutron lifetime will also result in a slower power decrease upon reactor trip. However, a large β_{eff} and a large prompt neutron lifetime will tend to retard the transient power increase. Sensitivity studies indicate that large values of β_{eff} and prompt neutron lifetime produce the highest core power response, and thus maximize the peak RCS pressure. Typical BOC decay constants and delayed neutron precursor fractions are also utilized in the analysis.

Scram Curve and Worth

A slow control rod drop time consistent with the Technical Specification value is assumed. This delays the core power decrease, leading to higher RCS pressures. A conservatively bottom peaked normalized rod worth versus position curve is also assumed. A minimum trippable worth, including an allowance for the most reactive stuck rod out of the core, is utilized in the analysis. The worth of the control rods inadvertently withdrawn is credited for insertion on reactor trip.

5.3.4 Control, Protection, and Safeguards Systems

Reactor Trip

For the startup accident, a late reactor trip is conservative in that the energy addition to the RCS is maximized. Reactor trip is expected to occur on the high pressure, high power, or the

flux/flow imbalance trip functions. Control rod shadowing and nuclear instrumentation calibration errors present at HZP are accounted for. Conservative setpoints are assumed for all credited trip functions with the appropriate conservative trip delay times.

RCS Pressure Control

The pressurizer spray and PORV are assumed inoperable, and the pressurizer heaters are assumed to be operable.

Pressurizer Level Control

No charging/letdown flow is modeled. It is assumed there would be no affect on the results of the analysis if charging/letdown was explicitly modeled due to the short duration of the transient.

5.3.5 Results

The startup accident models three reactor coolant pumps in operation and a maximum reactivity addition rate of 11.5 pcm/sec. Table 5-1 gives the sequence of events for this case. Figure 5-2 shows the neutron and thermal power as a function of time. Neutron power does not begin to appreciably increase until the inserted reactivity begins to approach $1. This occurs approximately 45 seconds into the rod withdrawal. Reactor trip occurs on high RCS pressure at 51.9 seconds with neutron power at approximately 125 % FP. After reactor trip, neutron power decreases rapidly as the control rods are inserted, and then decreases slowly due to delayed neutron fissions. The shape of the core thermal power response is similar to the neutron power response, but lags the neutron power peak by approximately 0.5 seconds. The core thermal power rises to a peak value of 73 % FP at 52 sec. Since the peak core thermal power is below the permissible power level with three RCPs in operation, DNB is not a concern for this transient and no VIPRE analysis is required for the assumed core physics parameters.

Figure 5-3 shows the kinetics response for this case. The reactivity insertion due to rod withdrawal is linear with time until reactor trip, when rod withdrawal ceases. Fuel heatup causes negative reactivity insertion due to the negative Doppler temperature feedback until reactor trip, which then causes a fuel temperature decrease and positive reactivity insertion. System heatup prior to reactor trip causes the moderator temperature to increase, which inserts positive reactivity due to the positive moderator temperature feedback assumed. As the negative

reactivity insertion due to the Doppler feedback becomes significant, the total reactivity increases more slowly, and actually begins to decrease prior to reactor trip. Upon reactor trip, total reactivity decreases rapidly in response to control rod insertion.

Figures 5-4 and 5-5 show the cold leg and hot leg temperatures as a function of time. Because of the reduced flow due to an inactive RCP, the loop B cold leg temperature response lags the loop A cold leg temperature response. After reactor trip the temperatures in both loops decrease.

Figure 5-6 shows the RCS pressure as a function of time. RCS pressure rises to a maximum value of approximately 2640 psig at 54.7 seconds, and then decreases due to pressurizer safety valve lift and blowdown. The peak RCS pressure of 2727 psig occurs at the bottom of the reactor vessel. The peak pressure remains below the 2750 psig acceptance criterion.

Figure 5-7 shows the pressurizer level response. During the thermal power excursion and associated RCS pressurization, level rises quickly due to the insurge of liquid into the pressurizer. After reactor trip and the opening of the pressurizer safety valves, the pressurizer level rises more slowly and begins to stabilize.

5.3.6 RETRAN-3D Comparison

In addition to the results described above, simulation results using the RETRAN-3D code are also provided on Figures 5-2 through 5-6. It is apparent from these figures that the RETRAN-02 and RETRAN-3D predictions are in very good agreement for the startup accident.

5.4 Reload Cycle-Specific Evaluation

Physics parameters that are checked for each reload core are:

- Moderator temperature coefficient
- Doppler temperature coefficient
- Minimum scram curve worth
- Maximum reactivity insertion rate

5.5 References

5-1 RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Revision 4, EPRI, November 1988

5-2 VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM-A, Revision 3, EPRI, October 1989

5-3 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000, Duke Power Company, July 1987

Table 5-1

Sequence of Events

Startup Accident - Peak RCS Pressure

Time (sec)	Event Description
0.0	Rod Withdrawal Begins
49.6	Pressurizer Control Heaters De-energize
51.4	High RCS Pressure Reactor Trip Setpoint Reached
51.9	Control Rod Insertion Begins
54.1	Pressurizer Safety Valves Open
54.7	Peak RCS Pressure Occurs
57.0	Pressurizer Safety Valves Reseat

Figure 5-1

RETRAN-02 Model Nodalization

Figure 5-2

STARTUP ACCIDENT

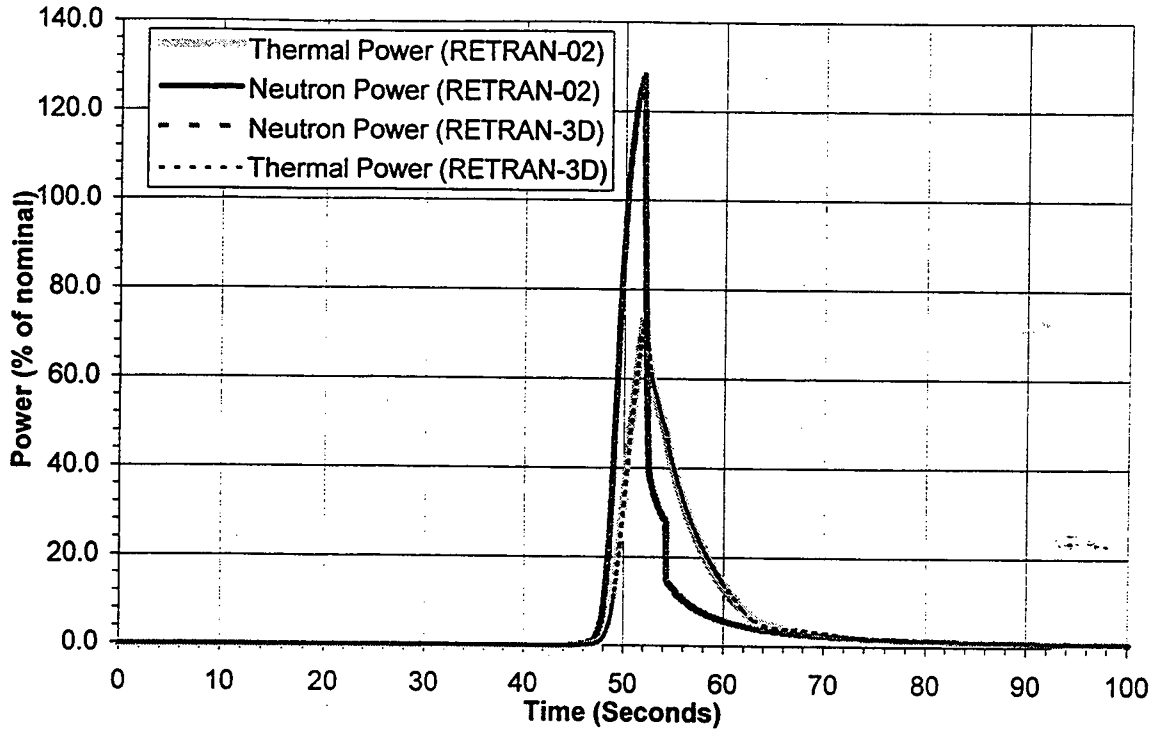


Figure 5-3

STARTUP ACCIDENT

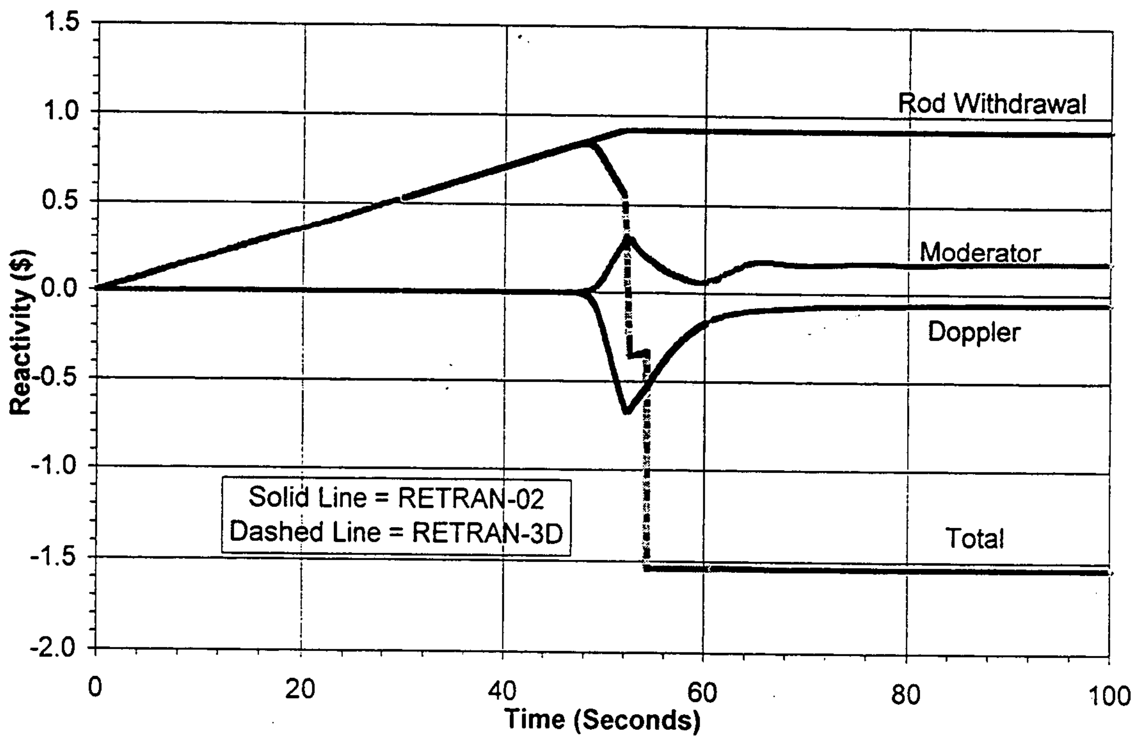


Figure 5-4
STARTUP ACCIDENT

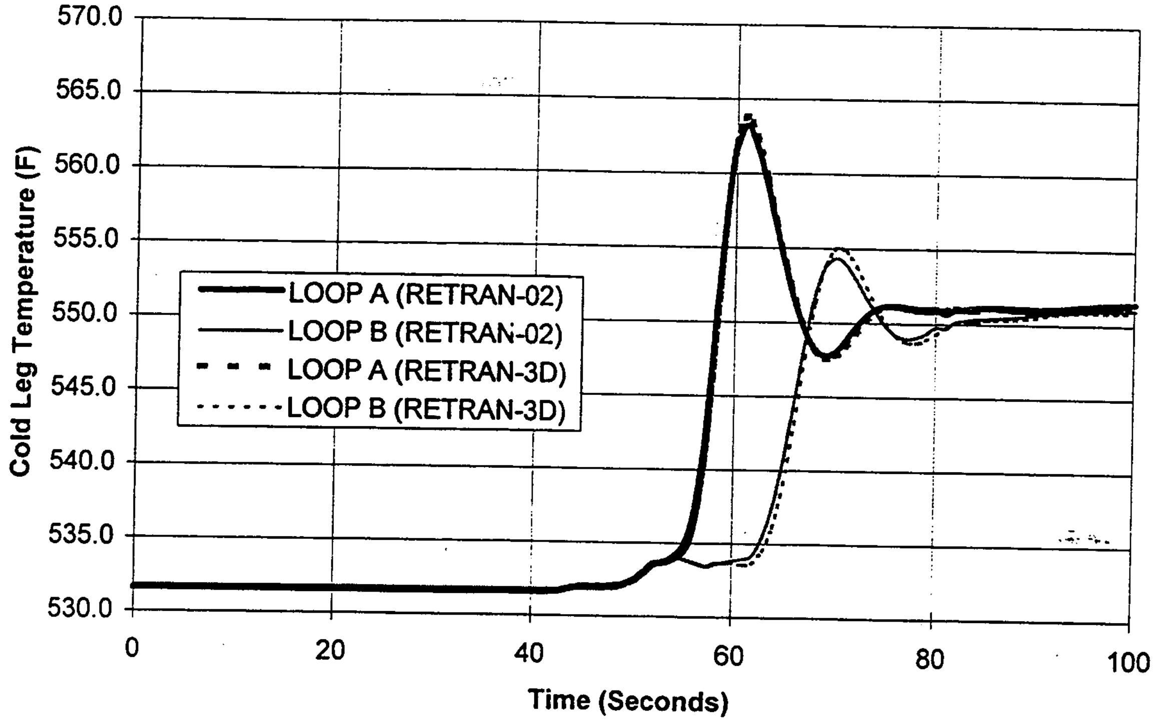


Figure 5-5
STARTUP ACCIDENT

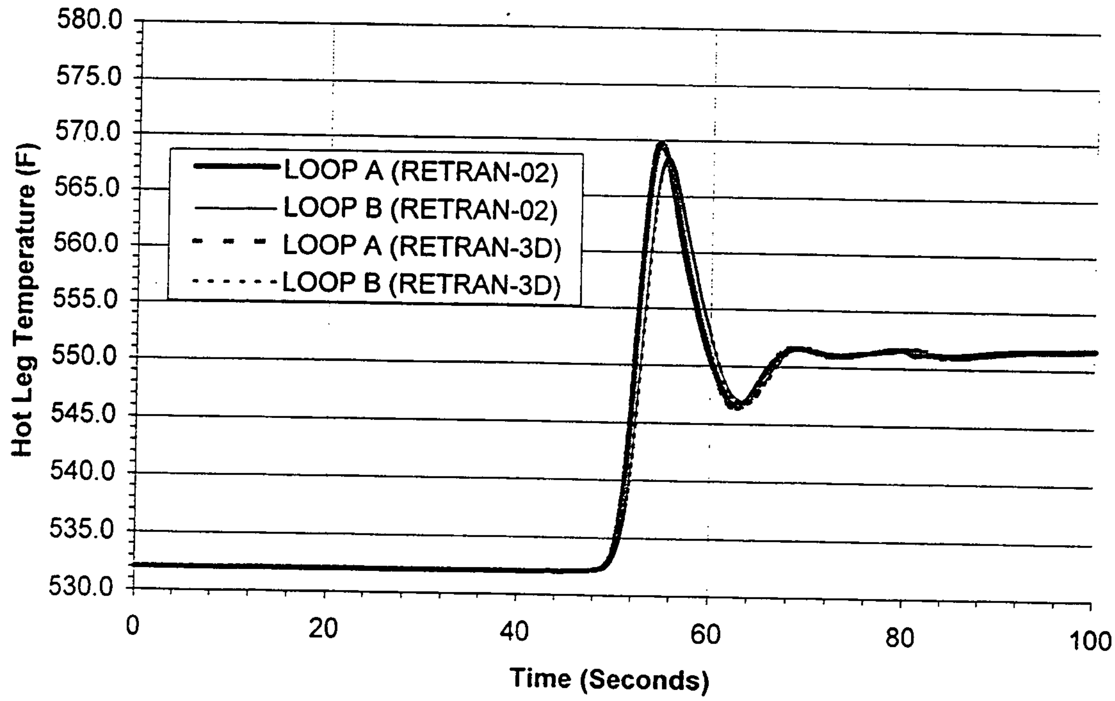


Figure 5-6

STARTUP ACCIDENT

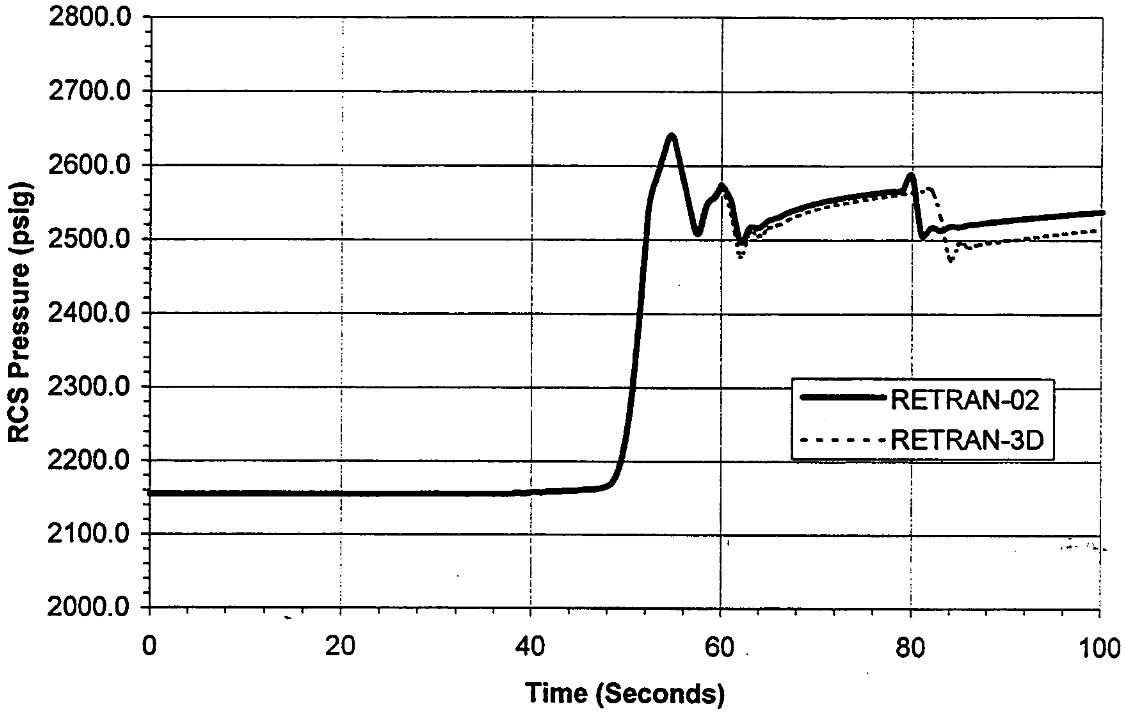
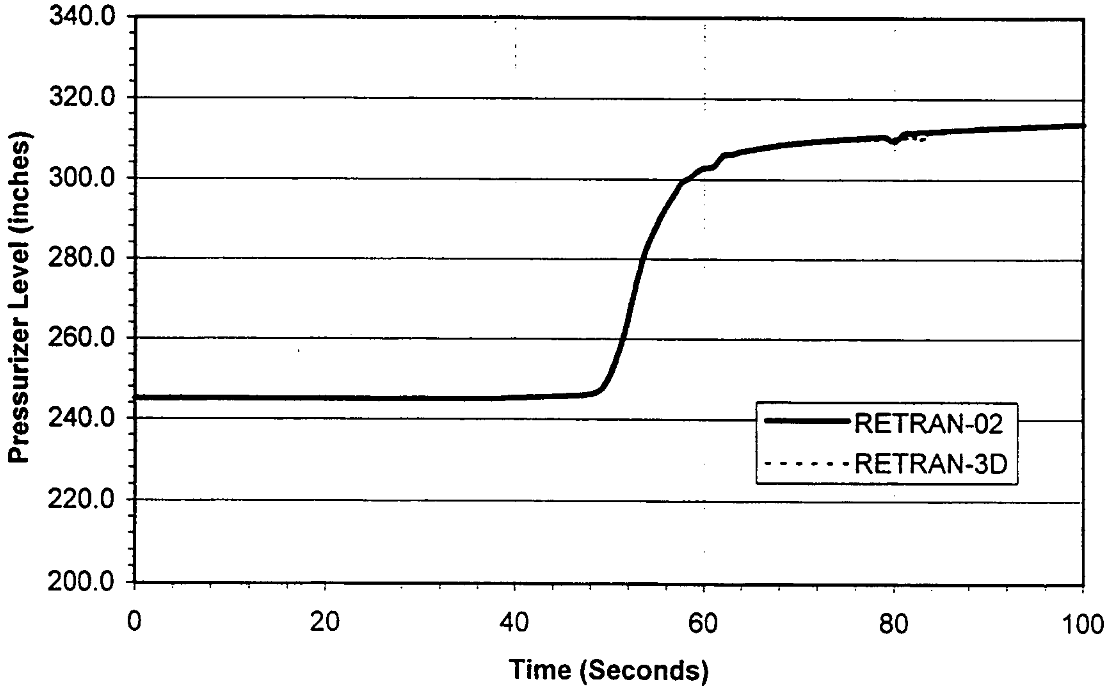


Figure 5-7

STARTUP ACCIDENT



6.0 ROD WITHDRAWAL AT POWER

The rod withdrawal accident initiates with an operator error or an equipment failure which results in accidental withdrawal of a control rod group while the reactor is at power. The rod withdrawal causes a reactor power excursion and a resultant heatup of both the Reactor Coolant System and the secondary system. If the accident is not adequately mitigated by the Reactor Protective System, the power excursion could lead to fuel rod failure or the overpressurization of the primary system. The peak primary pressure case is analyzed with the RETRAN-02 code (Reference 6-1). The acceptance criterion is that the peak RCS pressure remains below 110% of design pressure.

For the DNB analysis the RETRAN system transient response is utilized as input to a detailed core thermal-hydraulic analysis using the VIPRE-01 code (Reference 6-2). The core power distribution is analyzed with the SIMULATE-3P code (Reference 6-3). It is expected that the thermal-hydraulic conditions at the limiting DNB statepoint are within the ranges covered by the statistical core design (SCD) methodology, so the RETRAN DNB cases will use nominal initial conditions where appropriate.

6.1 Peak Primary System Pressure Analysis

6.1.1 Nodalization

Since the analysis includes both four and three reactor coolant pump operation, the two-loop RETRAN-02 Oconee base model as described in Section 2.2 of Reference 6-4 is used.

6.1.2 Initial Conditions

Power Level

Lower initial power is expected to be limiting with respect to peak RCS pressure since the initial margin to a high flux reactor trip is greater. The startup accident is analyzed starting from HZP. Therefore this analysis will initiate from full power to cover the range of possible initial power levels.

RCS Pressure

High initial RCS pressure is generally conservative with respect to peak RCS pressure analyses. However, low initial indicated RCS pressure will maximize the margin to a high pressure reactor trip. If the reactor trip occurs on high pressure, this could maximize the peak reactor thermal power level, and consequently the peak RCS pressure. Therefore, sensitivity cases are performed concerning initial RCS pressure.

Pressurizer Level

A high initial pressurizer level minimizes the volume of the steam bubble and therefore maximizes the RCS pressure increase following an insurge.

RCS Temperature

High initial RCS average temperature is assumed in order to maximize the primary coolant stored energy and minimize the primary system mass.

RCS Flow

Low flow is expected to be conservative with respect to peak RCS pressure since this minimizes the primary-to-secondary heat transfer.

Core Bypass Flow

A low core bypass flow is assumed in order to maximize the coolant flow along the fuel rods.

Fuel Temperature

Sensitivity cases are analyzed using both high and low initial fuel temperatures to determine whether high or low initial fuel temperature is more conservative. The more conservative initial temperature is then assumed.

Steam Generator Mass

A low initial steam generator mass is assumed since this will minimize the primary-to-secondary heat transfer, and thus maximize the primary system heatup/pressurization.

Steam Generator Tube Plugging

A high SG tube plugging percentage will degrade primary-to-secondary heat transfer and also minimize RCS inventory. Both of these effects will maximize the RCS pressure increase.

6.1.3 Boundary Conditions

Control Rod Group Withdrawal Rate

Sensitivity cases are performed on the withdrawal rate between the physical limits such that the most severe challenge to the respective acceptance criterion is obtained. Nominal assumptions are made for control rod speed, overlap and withdrawal sequence.

Pressurizer Safety Valves

The pressurizer code safety valves are modeled using conservative assumptions for drift, blowdown and valve capacity that minimize relief flow.

Main Steam Safety Valves

The main steam safety valves are modeled using conservative assumptions for drift, blowdown and valve capacity that minimize relief flow and maximize the secondary side pressure response. The increased secondary side temperatures associated with the higher pressure will yield reduced primary-to-secondary heat transfer, which is conservative for peak primary pressure.

Pressurizer Inter-Region Heat Transfer Coefficient

For this analysis, a conservatively low pressurizer inter-region heat transfer coefficient is assumed. This will maximize the rate of RCS pressurization and worsen the approach to the high pressure acceptance criterion.

Single Failure

No credible single failure has been identified which adversely impacts the results of the cases initiated from four-pump operation.

6.1.4 Physics Parameters

Moderator Temperature Coefficient

The BOC most positive/least negative values are used. This conservatively minimizes the negative reactivity feedback resulting from the coolant heatup during the power increase. Although this negative feedback could potentially delay reactor trip on high flux, this effect is compensated for in the withdrawal rate sensitivity.

Doppler Temperature Coefficient

The BOC least negative Doppler temperature coefficient values are used. This conservatively minimizes the negative reactivity feedback resulting from the fuel heatup during the power increase. Although this negative feedback could potentially delay reactor trip on high flux, this effect is compensated for in the withdrawal rate sensitivity.

Beta-effective and Neutron Lifetime

For a given withdrawal rate, a large β_{eff} will moderate the neutron power increase and therefore delay reactor trip on high flux. Also, a large β_{eff} will slow the post-trip neutron power decrease. Therefore, since the critical point for both acceptance criteria is reached shortly after reactor trip, the maximum value for β_{eff} is used in this analysis. Similarly, a maximum value for the prompt neutron lifetime is used, although this parameter has been found to have little impact on RETRAN results. Nominal BOC delayed neutron fractions and decay constants are assumed since these are insensitive parameters.

Scram Curve and Worth

A conservatively slow BOC scram curve along with a minimum BOC at power scram worth is used.

6.1.5 Control, Protection, and Safeguards Systems

Reactor Trip

Delaying reactor trip will increase the duration of the core power excursion, which will increase the peak power and the amount of RCS heatup and ultimately lead to a greater pressure increase. Reactor trip is expected to occur on high flux or high pressure for this transient. Conservative

setpoints are assumed for all credited trip functions with the appropriate conservative trip delay times.

RCS Pressure Control

The pressurizer spray and PORV are assumed inoperable, and the pressurizer heaters are assumed operable. These assumptions will maximize the RCS pressure increase.

Pressurizer Level Control

No charging/letdown flow is modeled. It is assumed there would be no affect on the results of the analysis if charging/letdown was explicitly modeled due to the short duration of the transient.

Main Feedwater System

Feedwater flow is held constant until the time of reactor trip. This is conservative since the ICS cross limits would increase feedwater flow in an attempt to match the increase in reactor power. After trip, feedwater flow is assumed to be lost, running back to zero flow in a conservatively short period of time. Lack of feedwater flow is expected to maximize the secondary side pressurization. The increased secondary side temperatures associated with the higher pressure will yield reduced primary-to-secondary heat transfer, which is conservative for the peak primary pressure.

Turbine Control

Prior to reactor trip, the turbine is under manual control. Turbine trip is assumed to occur coincident with reactor trip with no response time delay. The turbine stop valves are assumed to close rapidly.

Turbine Bypass System

The Turbine Bypass System is assumed to be inoperable to maximize peak secondary pressure. The increased secondary side temperatures associated with higher pressure will yield reduced primary-to-secondary heat transfer, which is also conservative for peak primary pressure.

Emergency Feedwater System

No credit is taken for emergency feedwater flow. The acceptance criteria for this transient are challenged within seconds of reactor trip, well before emergency feedwater initiation could occur.

6.2 Core Cooling Capability Analysis

6.2.1 Nodalization

Since the scope of this analysis encompasses both four and three reactor coolant pump operation, a two-loop RETRAN-02 Oconee base model as described in Section 2.2 of Reference 6-4 is used.

6.2.2 Initial Conditions

Power Level

The uncontrolled bank withdrawal event is analyzed with a set of initial power levels that covers the range of permissible power levels given the number of operating RCPs. Initial power levels below 15% are considered to be bounded by the startup accident. The uncertainty between actual and indicated power is included in the DNB limit (SCD).

RCS Pressure

Low RCS pressure is conservative with respect to DNB. The instrument uncertainty associated with the pressure indication is accounted for in the SCD limit. For the three-pump cases, the pressure in the loop with only one operating pump is set to the target value. Due to the impact of the lower loop flow rate on the pressure drop between the core exit and the hot leg pressure taps, the pressure in this loop is higher than that in the loop with two operating pumps. Coupled with the single failure discussed below, this will conservatively delay reactor trip on high RCS pressure.

Pressurizer Level

A low initial pressurizer level maximizes the volume of the steam bubble and therefore minimizes the pressure increase following an insurge. Low pressure is conservative with respect to DNB.

RCS Temperature

Between 15% and 100% FP, the Integrated Control System controls the RCS average temperature indication to a constant value. The instrument uncertainty associated with RCS T-ave is included in the DNB limit (SCD).

RCS Flow

Low flow is conservative with respect to DNB. Since the uncertainty in the RCS flow indication is accounted for in the SCD limit, the actual RCS flow is assumed to be equal to the minimum indicated value.

Core Bypass Flow

A high core bypass flow is assumed to minimize the coolant flow along the fuel rods.

Fuel Temperature

Sensitivity cases are analyzed using both high and low initial fuel temperatures to determine whether high or low initial fuel temperature is more conservative. The more conservative initial temperature is then assumed.

Steam Generator Mass

A high initial steam generator mass maximizes the available heat sink which should slow the primary system pressurization.

Steam Generator Tube Plugging

The minimum actual SG tube plugging is assumed. Low tube plugging will lessen the rate of RCS heatup/pressurization and delay reactor trip on high pressure. Following reactor trip, the lower tube plugging will help the Turbine Bypass System minimize the RCS pressurization. Both of these effects are conservative with respect to DNB.

6.2.3 Boundary Conditions

Control Rod Group Withdrawal Rate

Sensitivity cases are performed on the withdrawal rate between the physical limits such that the most severe challenge to the acceptance criterion is obtained. Nominal assumptions are made for

control rod speed, overlap and withdrawal sequence.

Pressurizer Safety Valves

The pressurizer code safety valves are modeled using conservative assumptions for drift, blowdown and valve capacity that maximize relief flow.

Main Steam Safety Valves

The DNBR acceptance criterion is challenged within two seconds of reactor trip, so the MSSV modeling will have no impact on the transient results.

Pressurizer Inter-Region Heat Transfer Coefficient

For this analysis, a conservatively high pressurizer inter-region heat transfer coefficient is assumed. This will decrease the rate of RCS pressurization, which will both delay (or avoid) reactor trip on high pressure and minimize the core exit pressure at the DNB statepoint.

Single Failure

No credible single failure has been identified which adversely impacts the results of the cases initiated from four-pump operation. For the cases initiated from three-pump operation, the failure of one of the RCS pressure transmitters on the loop with only one pump in operation is assumed. Due to the 2/4 coincidence logic, this requires that the high pressure reactor trip be generated in the loop reading a lower RCS pressure, which conservatively delays reactor trip.

6.2.4 Physics Parameters

Moderator Temperature Coefficient

The BOC most positive/least negative moderator temperature coefficient values are used. This conservatively minimizes the negative reactivity feedback resulting from the coolant heatup during the power increase. Although this negative feedback could potentially delay reactor trip on high flux, this effect is compensated for in the withdrawal rate sensitivity.

Doppler Temperature Coefficient

The BOC most positive Doppler temperature coefficient values are used. This conservatively minimizes the negative reactivity feedback resulting from the fuel heatup during the power

increase. Although this negative feedback could potentially delay reactor trip on high flux, this effect is compensated for in the withdrawal rate sensitivity.

Beta-effective and Neutron Lifetime

For a given withdrawal rate, a large β_{eff} will moderate the neutron power increase and therefore delay reactor trip on high flux. Also, a large β_{eff} will slow the post-trip neutron power decrease. Therefore, since the critical point for both acceptance criteria is reached shortly after reactor trip, the maximum value for β_{eff} is used in this analysis. Similarly, a maximum value for the prompt neutron lifetime is used, although this parameter has been found to have little impact on RETRAN results. Nominal BOC delayed neutron fractions and decay constants are assumed since these are insensitive parameters.

Scram Curve and Worth

A conservatively slow BOC scram curve along with a minimum BOC at power scram worth is used.

6.2.5 Control, Protection and Safeguards Systems

Reactor Trip

Delaying reactor trip will increase the duration of the core power excursion, which will increase the peak power and the amount of RCS heatup. Reactor trip is expected to occur on high flux or high pressure for this transient, although the high coolant temperature and flux/flow/imbalance trips may also provide protection. Conservative setpoints are assumed for all credited trip functions with the appropriate conservative trip delay times.

RCS Pressure Control

The pressurizer spray and PORV are assumed to be operable. The pressurizer heaters are assumed to be inoperable. Conservative assumptions are made for the PORV opening and closing times with respect to maximizing relief flow, although PORV lift is not expected. These assumptions will minimize the pressure increase, which is conservative with respect to DNB.

Pressurizer Level Control

No charging/letdown flow is modeled. It is assumed there would be no affect on the results of the analysis if charging/letdown was explicitly modeled due to the short duration of the transient.

Main Feedwater System

Sensitivities on feedwater control are performed. To maximize the RCS heatup, feedwater is assumed to be under manual control and held constant until the time of reactor trip. To maximize the duration of the reactor power excursion, and thus the peak power, feedwater is assumed to be under Integrated Control System (ICS) control with the reactor-to-feedwater cross limits active. As reactor power increases, the cross limits will increase feedwater flow, which will delay reactor trip on high pressure. Under manual control, feedwater is not isolated at reactor trip. Under automatic control, feedwater would be throttled by the ICS to achieve the post-trip steam generator level setpoint.

Turbine Control

Prior to reactor trip, if main feedwater is under ICS control, the turbine will also be controlled via the ICS. Similarly, if main feedwater is under manual control, the turbine is under manual control. Turbine trip is assumed to occur on reactor trip with a conservatively long response time delay. This modeling will minimize the post-trip RCS pressurization.

Turbine Bypass System

The Turbine Bypass System is assumed to be operable and controlling pre-trip and post-trip steam generator pressure. This minimizes steam generator pressure and increases primary-to-secondary heat transfer. This is conservative since it will minimize the RCS pressurization. The reduction in RCS temperature will not reach the core inlet prior to the minimum DNBR statepoint.

Emergency Feedwater System

No credit is taken for emergency feedwater flow. The acceptance criteria for this transient are challenged within seconds of reactor trip, well before emergency feedwater initiation could occur.

6.3 VIPRE-01 Analysis

The forcing functions necessary to perform the DNB analysis (core average heat flux, core inlet flow and temperature, core exit pressure) are obtained from the RETRAN-02 analysis results and input to VIPRE-01. The VIPRE-01 14 channel model (Reference 6-4) is then used to determine the time of the minimum DNBR statepoint for the transient conditions analyzed. At these statepoint conditions a set of maximum allowable radial peak (MARP) curves is developed for determining if the DNBR limit is exceeded.

6.4 Results

The peak primary pressure reached in the limiting case is approximately 2600 psig. This is well below the acceptance criterion of 2750 psig. The results of the DNBR analysis have demonstrated that the power peaking predicted by SIMULATE-3P will remain below the DNBR limits.

6.5 Reload Cycle-Specific Evaluation

Physics parameters that are checked for each reload core are:

- Moderator temperature coefficient
- Doppler temperature coefficient
- Minimum scram worth curve
- Minimum and maximum reactivity insertion rates
- Maximum allowable radial peak limits

6.6 References

6-1 RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI-NP-1850-CCM, Revision 4, EPRI, November 1988

6-2 VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM-A, Revision 3, EPRI, August 1989

6-3 SIMULATE-3: Advanced Three-Dimensional Two-Group Reactor Analysis Code, Studsvik/SOA-92/01, Studsvik of America, April 1992

6-4 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000, Duke Power Company, July 1987

7.0 MODERATOR DILUTION ACCIDENT

7.1 Description

A Reactor Coolant System moderator dilution event occurs when the soluble boric acid concentration of makeup water supplied to the RCS is less than the concentration of the existing reactor coolant. The moderator dilution accident postulates that such a dilution occurs, potentially resulting in a loss of shutdown margin, approaching the DNBR limit, or challenging the peak primary pressure limit. This accident is conservatively analyzed to ensure that the boron dilution is terminated by the operator such that these criteria are not exceeded. The acceptance criteria for manual operator action is at least 15 minutes during power operation and at least 30 minutes during refueling following the actuation of the alarm credited for alerting the operator of a moderator dilution event.

This accident is analyzed at the conditions of power operation (Mode 1) and refueling (Mode 6). Manual operator action is relied on to terminate the dilution in both modes. Mode 1 is analyzed to demonstrate that there is adequate time for the operator to terminate the dilution when maximum dilution source flowrates are assumed. Mode 6 is analyzed assuming administrative controls on potential dilution sources such that the results of the accident analysis give exactly the 30 minute operator response time. Flowrates are restricted through administrative controls to values which are less than these analyzed flowrates. This results in actual longer operator response times. Additional operator response time margin is provided by the margin between the assumed boron concentrations for Mode 6 and the actual concentrations for the reload core.

7.2 Initial Conditions

Dilution Volume

A dilution event progresses faster for smaller RCS water volumes. Therefore, the analysis considers the smallest RCS water volume in which the unborated water is mixed. For Mode 1, the forced circulation provided by the reactor coolant pumps will mix the RCS inventory in the reactor vessel and each of the reactor coolant loops. The pressurizer and the pressurizer surge line are not included in the volume available for dilution in Mode 1. For Mode 6, the reactor coolant water level may be drained to the top of the main coolant loop piping, with at least one

train of the Low Pressure Injection (LPI) system in operation. The volume available for dilution in this mode is limited to the smaller volume LPI train plus the portions of the reactor vessel and reactor coolant loop piping below the minimum water level.

Boron Concentrations

The Technical Specifications require that the shutdown margin in the various modes be above a certain minimum value. The difference in boron concentration between the value at which an alarm function is actuated and alerts the operator, and the value at which the reactor is just critical, determines the time available to mitigate a dilution event. This time is a function of the ratio of these two concentrations, where a large ratio corresponds to a longer time. During the reload safety analysis for each reload core, the above concentrations are checked to ensure that the value of this ratio for each mode is larger than the corresponding ratio assumed in the accident analysis. For the Mode 1 initial conditions in which the control rods are withdrawn, it is conservatively assumed when calculating the critical boron concentration that the most reactive rod does not fall into the core at reactor trip.

7.3 Boundary Conditions

In the absence of administrative flowrate restrictions, the dilution flowrate assumed to enter the RCS is greater than or equal to the maximum volumetric flowrate of those pumps supplying the dilution flow. In a dilution event, these pumps are assumed to deliver unborated water to the suction of the HPI pumps. Since the water delivered by these pumps is typically colder than the RCS, the unborated water expands within the RCS, causing a volumetric flowrate measured at the colder temperature to correspond to a higher effective volumetric dilution flowrate. This density difference in the dilution flowrate is accounted for in the analysis. For Mode 6, the maximum permissible dilution flowrate is determined in the manner previously discussed.

7.4 Control, Protection, and Safeguards Systems

Mitigation of a boron dilution accident is not assumed to begin until an alarm has alerted the operator to the abnormal circumstances caused by the event. For Mode 1 with manual rod control, the alarm function is provided by the earliest reactor trip setpoint reached. During Mode 1 with automatic rod control, the alarm function is provided by the alarm which occurs when the

control rods reach their insertion limits. This alarm is monitored by the plant computer. For Mode 6, the alarm function is provided by the source range high-flux-at-shutdown alarm exceeding its setpoint. This alarm is provided by the source range stripchart recorder, and is manually set by the operator. The alarm setpoints are determined by analysis to ensure that the acceptance criteria are met.

7.5 Reload Cycle-Specific Evaluation

The highest critical boron concentration and the initial boron concentration closest to the critical concentration must be checked on a cycle-specific basis.

8.0 COLD WATER ACCIDENT

The cold water accident initiates with an inadvertent startup of the fourth reactor coolant pump (RCP) from an initial three-pump operating condition. This event will cause a reduction in moderator temperature, which will result in a power excursion due to the moderator feedback effect. If the power increase is large enough, DNB may result with subsequent fuel damage. The cold water accident is analyzed with the RETRAN-02 code (Reference 8-1).

The acceptance criteria for this analysis are to ensure that there is adequate core cooling capability and that the pressure in the Reactor Coolant System (RCS) remains below 110% of design pressure. The core cooling capability analysis demonstrates that fuel cladding integrity is maintained by ensuring that the minimum DNBR remains above the DNBR limit. The minimum DNBR can be determined using the statistical core design (SCD) methodology. The peak RCS pressure limit is not approached during this event. The initial conditions and boundary conditions chosen for this analysis are therefore those that will result in the lowest DNBR.

8.1 Nodalization

This asymmetric transient is analyzed using the two-loop RETRAN-02 Oconee base model (Reference 8-2). A junction is added to the base model to connect the steam lines since an asymmetric steam generator response will occur during this event.

8.2 Initial Conditions

Power Level

A high initial power level for three-pump operation maximizes the primary system heat flux. The uncertainty for this parameter is incorporated in the SCD methodology.

RCS Pressure

Low initial RCS pressure is conservative for DNBR. The uncertainty for this parameter is incorporated in the SCD methodology.

Pressurizer Level

Low initial pressurizer level increases the volume of the pressurizer steam space which minimizes the pressure increase resulting from the power increase and subsequent insurge.

RCS Temperature

The nominal temperature is assumed, with the uncertainty for this parameter incorporated in the SCD methodology.

RCS Flow

Low initial RCS flow is conservative for DNB analyses. The uncertainty associated with this parameter is incorporated in the SCD methodology.

Core Bypass Flow

A high core bypass flow is assumed to minimize the coolant flow across the fuel rods.

Fuel Temperature

A smaller increase in fuel temperature is seen during this transient when starting from a lower initial fuel temperature. With a negative Doppler coefficient, this results in a smaller negative reactivity insertion which will result in a higher peak power. Therefore, a low initial fuel temperature is assumed.

Steam Generator Mass

A high initial steam generator mass is assumed to maximize primary-to-secondary heat transfer.

Steam Generator Tube Plugging

Low steam generator tube plugging is assumed to maximize primary-to-secondary heat transfer.

8.3 Boundary Conditions

RCP Operation

Three RCPs are initially operating with the fourth RCP being started from this condition. A range of RCP start times are assumed which bound the nominal RCP start time.

Pressurizer Inter-Region Heat Transfer Coefficient

A high inter-region heat transfer coefficient is assumed to maximize the heat transfer rate at the liquid-steam interface of the pressurizer. This will minimize RCS pressurization due to the power increase and subsequent pressurizer insurge.

Single Failure

No credible single failure has been identified which adversely impacts this transient.

8.4 Physics Parameters

Moderator Temperature Coefficient

The EOC most negative moderator temperature coefficient value is used. This conservatively maximizes the positive reactivity feedback resulting from the coolant temperature decrease after the fourth RCP is started.

Doppler Temperature Coefficient

The EOC least negative Doppler temperature coefficient value is used. This conservatively minimizes the negative reactivity feedback resulting from the fuel heatup during the power increase.

Beta-Effective and Neutron Lifetime

B_{eff} impacts both the moderator and Doppler reactivity feedback during the transient. Since positive moderator feedback and negative Doppler feedback occur, a sensitivity study must be performed on this parameter to determine if a minimum or maximum value should be used. The prompt neutron lifetime associated with the minimum and maximum B_{eff} values are used. Nominal EOC delayed neutron fractions and decay constants are assumed since these are insensitive parameters.

Scram Curve and Worth

A conservatively slow EOC scram curve along with a minimum EOC at power scram worth are used.

8.5 Control, Protection, and Safeguards Systems

Reactor Control

The reactor control subsystem of the Integrated Control System (ICS) is assumed to be in manual. A power increase would result in the ICS inserting rods to maintain the initial power level, so the ICS is not credited.

Reactor Trip

The reactor trip function credited is the high flux trip. A conservative trip delay time is assumed. A penalty for a reduction in the excore flux signal due to a decrease in the reactor vessel downcomer temperature is modeled.

RCS Pressure Control

Pressurizer spray is assumed to be operable to minimize the pressure increase resulting from the power increase. Pressurizer heaters are assumed to be inoperable.

Pressurizer Level Control

No net charging/letdown flow is modeled. It is assumed there would be no affect on the results of the analysis if charging/letdown was explicitly modeled due to the short duration of the transient.

Main Feedwater System

The Main Feedwater System is assumed to be in automatic control. As reactor power increases, the ICS will increase flow to the steam generators. This will maximize primary-to-secondary heat transfer which will limit RCS heatup.

Turbine Control

The turbine control subsystem of the ICS is assumed to be in automatic. As the power increases and additional steam generator inventory boils off, the ICS will attempt to prevent the increase in steam line pressure by opening the turbine control valves. This will maximize primary-to-secondary heat transfer and limit RCS heatup.

The results of the RETRAN-02 analysis are that the peak heat flux is 96.7% of the full power heat flux, and that the flow has reached the four-pump full power flowrate. Therefore a detailed analysis of DNB is not required since the DNBR is greater than the full power steady-state DNBR.

8.7 Reload Cycle-Specific Evaluation

Physics parameters that are checked for each reload core are:

- Moderator temperature coefficient
- Doppler temperature coefficient
- Minimum scram worth curve

8.8 References

8-1 RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI-NP-1850-CCM, Revision 4, EPRI, November 1988

8-2 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000, Duke Power Company, July 1987.

9.0 LOSS OF COOLANT FLOW

9.1 Overview

9.1.1 Description

The loss of coolant flow accident is the result of an electrical failure to the reactor coolant pumps (RCPs) and results in one or more RCPs coasting down. Depending on the assumed single failure, either the pump monitor trip or the flux/flow/imbalance trip will trip the reactor. The major concern in the loss of coolant flow accident is the potential for departure from nucleate boiling (DNB).

9.1.2 Acceptance Criteria

The acceptance criterion for the loss of flow accident is that no fuel failures will result as demonstrated by not exceeding the DNBR limit.

9.1.3 Analytical Approach

The loss of flow transient requires a limiting set of physics parameters along with conservative initial and boundary conditions. These parameters are input to the Oconee RETRAN-02 model (References 9-1 and 9-2) for the system thermal-hydraulic analysis. The RETRAN-02 analysis generates the transient forcing functions to be input to the Oconee VIPRE-01 model (References 9-2 and 9-3) to calculate the transient minimum DNBR.

9.2 Simulation Codes and Models

9.2.1 RETRAN-02

The RETRAN-02 code is used for the loss of flow system analyses. The two-loop Oconee RETRAN model is used for the loss of flow analysis. This two-loop model is used for its capability to simulate all combinations of RCP coastdown events. RETRAN predicts core exit pressure, core inlet temperature, core inlet flow, and core heat flux for the RCP coastdown

events. The RETRAN analysis initial conditions are based on the statistical core design (SCD) methodology.

9.2.2 VIPRE-01

The VIPRE-01 code is used for the loss of flow core thermal-hydraulic analyses. VIPRE thermal-hydraulic boundary conditions are obtained from the RETRAN system transient simulation. The [ ] channel VIPRE model described in Section 2.3.2 of this report is used to calculate the limiting statepoint local coolant properties and DNBR. Since the RETRAN surface heat flux is used, the VIPRE conduction model is not used for the loss of flow analyses. The critical heat flux (CHF) correlations used are the BWC or BWU-Z CHF correlations (References 9-4 and 9-5). The VIPRE analysis employs the SCD methodology for the loss of flow core thermal-hydraulic analyses.

9.2.3 SIMULATE-3P

SIMULATE-3P (Reference 9-6) is a core neutronics code used to generate the safety analysis physics parameters and three-dimensional core pin power distributions for the loss of flow transient.

9.3 Core Cooling Capability Analysis

9.3.1 RETRAN-02 Analysis

The Reactor Protective System (RPS) includes a flux/flow/imbalance trip and a pump monitor trip to provide DNB protection for RCP coastdown events. The flux/flow/imbalance trip trips the reactor when the setpoint is reached, and the pump monitor trip trips the reactor when any two of the four RCPs have tripped if the reactor power is greater than 2% FP. Either the flux/flow/imbalance trip or the pump monitor trip will trip the reactor when different number of RCPs coast down from either four or three-pump operation. Therefore, the possible RCP coastdown events from initial four or three RCP operation are determined. Since some of these RCP coastdown events are bounded by others, only the five RCP coastdown events shown below need to be analyzed for the loss of flow transient.

Case Analyzed	RCP Coastdown*	Power Level (% FP)	Trip Function
1	4/1	100	flux/flow
2	4/2**	100	flux/flow
3	4/4	100	pump monitor
4	3/1**	80	flux/flow
5	3/3	80	pump monitor

* 4/1 means 1 RCP coasting down from an initial four RCP operation.

** The RCPs can be in the same loop or different loops.

9.3.1.1 Initial Conditions

Core Power

Nominal rated thermal power is assumed for the four-pump initial condition and 80% FP for the three-pump initial condition since the uncertainty in power is included in the DNB limit (SCD).

RCS Pressure

Low RCS pressure is conservative for the DNB analysis. The instrument uncertainty is included in the SCD limit.

Pressurizer level

Low pressurizer level is conservative to minimize the pressure increase during the insurge.

RCS Temperature

High RCS average temperature is conservative for the DNB analysis. The instrument uncertainty is included in the SCD limit.

RCS Flow

Low flow is conservative for the DNB analysis. The instrument uncertainty is included in the SCD limit.

Core Bypass Flow

Consistent with the RCS flow assumption, high bypass flow is assumed.

Fuel Temperature

Fuel temperature is chosen to maximize the heat flux which is conservative for the DNB analysis. A high initial temperature is appropriate.

Steam Generator Mass

Steam generator mass has a negligible effect on the results of the analysis.

Steam Generator Tube Plugging

Tube plugging has a negligible effect on the results of the analysis.

9.3.1.2 Boundary Conditions

Reactor Coolant Pump Coastdown

Reactor coolant pumps are assumed to coast down at time zero.

Pressurizer Inter-Region Heat Transfer Coefficient

A conservatively large pressurizer inter-region heat transfer coefficient is assumed to decrease the rate of RCS pressurization.

Main Steam Safety Valves

The main steam safety valves (MSSVs) are modeled with conservative assumptions for drift, blowdown, and relief capacity to minimize the post-trip secondary pressure response, which in turn will minimize the primary system pressure response.

Single Failure

A single failure of the pump monitor trip is assumed.

9.3.1.3 Physics Parameters

Moderator Temperature Coefficient

The most positive/least negative BOC moderator temperature coefficient is assumed to minimize feedback due to system heatup.

Doppler Temperature Coefficient

The least negative Doppler temperature coefficient minimizes the decrease in the pre-trip power level and minimizes the post-trip power decrease.

Beta-Effective and Neutron Lifetime

The greater the delayed neutron fraction, the slower reactor power will decrease post-trip. Since the point of minimum DNBR occurs after reactor trip, the maximum BOC value for β_{eff} is assumed. The prompt neutron lifetime is not a key parameter for this analysis.

Scram Curve and Worth

A conservatively slow BOC scram curve along with a minimum BOC at power scram worth are used.

9.3.1.4 Control, Protection and Safeguards Systems

Reactor Trip

The Reactor Protective System (RPS) includes a flux/flow trip and a pump monitor trip to provide DNB protection for RCP coastdown events. A conservative trip delay time is assumed.

RCS Pressure Control

Since low pressure is conservative with respect to the DNB analysis result, pressurizer heaters are assumed inoperable. Pressurizer spray is assumed operable for the same reason. The pressurizer PORV and the pressurizer safety valves do not open during the transient.

Main Feedwater System

Main feedwater control is assumed to be in manual to minimize secondary temperature and thus RCS pressure.

Turbine Control

A slow turbine stop valve stroke time (1.0 sec) is assumed to minimize secondary pressure, thereby minimizing RCS pressure, which is conservative for DNB.

Turbine Bypass System

The Turbine Bypass System is assumed operable with the secondary pressure controlled to a low post-trip value. This minimizes secondary pressure, thereby minimizing primary pressure, which is conservative for DNB.

9.3.1.5 Results

The five RCP coastdown events described in Section 9.3.1 are analyzed. The maximum heat flux to core inlet mass flow ratio during the transient is used as an indicator to determine which is the worst coastdown event. Results show that for the four-pump initial condition from 100% power, two RCPs in the same loop coasting down is the bounding event. For the three-pump initial condition a loss of the RCP in the same loop as the initially idle RCP is the bounding event.

Two RCPs Trip From Four RCP Initial Condition

Two RCPs in the same loop trip at time zero. The duration of the simulation is 19 seconds. The sequence of events is shown in Table 9-1. Prior to control rod motion, the reactor power (Figure 9-1) has already decreased because of the negative moderator temperature feedback as a result of the increase in coolant temperature in the core due to the decrease in flow. The flux/flow trip trips the reactor and the insertion of control rods begins following the delay time. The reactor power decreases rapidly as the control rods insert. The reactor core average heat flux (Figure 9-1) follows the trend of the reactor power with a thermal delay.

RCS pressure (Figure 9-2) increases initially due to the insurge of the reactor coolant into the pressurizer as a result of the RCS average temperature increase due to the flow coastdown. After the reactor trips, the RCS pressure decreases due to the post-trip shrinkage. The pressurizer level (Figure 9-3) starts to increase after the RCPs trip due to the increase in the RCS average temperature. The pressurizer level then decreases due to the decrease of the RCS average temperature after the reactor trips.

The core flow (Figure 9-4) decreases after the RCPs trip, and approaches the equilibrium 2-pump flowrate at the end of the simulation. The faulted loop flow decreases toward zero flow, while the intact loop flow increases from its initial value at the end of the simulation.

The hot leg temperatures (Figure 9-5) increase initially due to the decrease of coolant flow. After the reactor trips, the hot leg temperatures begin decreasing to the normal post-trip temperature. After the RCPs trip, the cold leg temperature in the faulted loop decreases due to the decrease in primary flow. After the turbine trips, the temperature starts to increase due to the increase in steam pressure. The cold leg temperature of the intact loop remains stable after the initiation of the RCP trip, then it increases due to the increase in steam pressure resulting from the turbine trip.

One RCP Trip From Three RCP Initial Condition

The RCP in the same loop as the initially idle RCP trips at time zero. The duration of the simulation is 19 seconds. The sequence of events is shown in Table 9-2.

Figures 9-6 to 9-10 show the key parameter trends for the 3/1 RCP coastdown transient. The transient behavior of these parameters trend those of the 4/2 RCP transient. Therefore, no discussion of the trends will be given with the exception of RCS flow. The RCS flow (Figure 9-9) is approaching the two RCP equilibrium flowrate at the end of the simulation. While the faulted loop flow decreases and reverses direction, the intact loop flow increases from its initial value.

9.3.1.6 RETRAN-3D Comparison

The RETRAN-02 and RETRAN-3D results comparisons are shown in Figures 9-1 to 9-5. The results are in good agreement with the exception of the primary system pressure. A difference of 10 psi (approximately 0.4%) between the RETRAN-02 and RETRAN-3D primary pressure predictions develops in the first 10 seconds of the transient. Pressure is very sensitive to rapid increases in pressurizer level, and in this case the deviation in the pressurizer level is 1 inch. This level deviation results from a small difference in primary-to-secondary heat transfer beginning at around 7 seconds when the main steam safety valves begin to lift. This is attributed to small differences in the predicted critical flow through the main steam safety valves between RETRAN-02 and RETRAN-3D. The critical flow at a junction is a function of the pressure and enthalpy of the fluid. These properties are calculated in slightly different ways in RETRAN-02 and RETRAN-3D. Both calculation methods are appropriate approximations based on known local fluid properties. These small differences are insignificant considering the much greater

effect of the numerous conservatisms included in the loss of flow analysis, and noting that the time of minimum DNBR (5 seconds) precedes the onset of the deviation.

9.3.2 VIPRE-01 Analysis

Utilizing the VIPRE[ ]channel model described in Section 9.2.2 and the RETRAN boundary conditions, VIPRE-01 calculates the transient minimum DNBR.

9.3.2.1 Initial and Boundary Conditions

The RETRAN analyses generate the transient core exit pressure, core inlet temperature, core inlet flow rate, and core average heat flux for the RCP coastdown events described in Section 9.3.1.3. These boundary conditions are input to the VIPRE[ ]channel model as transient forcing functions.

9.3.2.2 Axial and Radial Power Distributions

The axial power distribution is a chopped cosine shape with an axial peak of[ ]peaked at X/L = [ ]. The radial power distribution is the base model radial power distribution with a pin radial power of[ ](Reference 9-2).

9.3.2.3 Conservative Factors

Since the SCD methodology is utilized for predicting the DNBR, the SDL accounts for most of the uncertainties in key parameters. Based on the vessel model flow test and Oconee core pressure drop measurement, the core inlet flow maldistribution is modeled as a reduction in the hot assembly flow. The hot assembly flow reduction factor is shown below for different RCP combinations.

# of RCPs	Flow Reduction Factor
4 RCPs	5%
3 RCPs	[
2 RCPs	]

9.3.2.4 Critical Heat Flux Correlation

The critical heat flux (CHF) correlation used for the loss of flow transient DNBR calculation is the BWC CHF correlation (Reference 9-4). The range of applicability for the BWC CHF correlation is:

Pressure (psia)	1600 to 2600
Mass flux (Mlbm/hr-sqft)	0.43 to 3.8
Quality	-0.20 to 0.26

The BWC CHF correlation SCD limit for the loss of flow transient will be determined utilizing the minimum DNBR statepoint boundary conditions described in Section 9.3.2.5.

9.3.2.5 Results

The transient VIPRE minimum DNBR for the 4/2 and 3/1 RCP coastdown transients are shown below:

RCP Coastdown Case	DNBR	Time of MDNBR (seconds)
4/2	1.69	5.3
3/1	2.02	4.8

Figures 9-11 and 9-12 show the transient minimum DNBR versus time for the two RCP coastdown cases. The above statepoints at which the minimum DNBR occur are used to determine the SCD limit for the RCP coastdown for four and three-pump initial conditions. The minimum DNBR results are greater than the DNB limit, and therefore no fuel failure occurs. These statepoints are then used in the determination of the core power peaking limits for normal operation according to the methodology described in Reference 9-7.

9.4 Reload Cycle-Specific Evaluation

Physics parameters that are checked for each reload core are:

- Moderator temperature coefficient
- Doppler temperature coefficient
- Minimum scram worth curve

9.5 References

9-1 RETRAN-02: A Program for transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Revision 4, EPRI, November 1988

9-2 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000, Duke Power Company, July 1987

9-3 VIPRE-01 : A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM-A, Revision 3, August 1989

9-4 BAW-10143-PA, BWC Correlation of Critical Heat Flux, April 1985

9-5 BAW-10199-PA, The BWU Critical Heat Flux Correlation, April 1996

9-6 DPC-NE-1004A, Duke Power Company Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, November 1992

9-7 DPC-NE-2003P-A, Core Thermal-Hydraulic Methodology Using VIPRE-01, October 1989

Table 9-1

Four RCP Operation

Sequence of Events

Time (sec)	Event Description
0.0	Two RCPs trip
3.51	Flux/flow reactor trip setpoint reached
4.71	Rod motion begins
4.87	Turbine trip on reactor trip
5.99	PZR spray initiates
7.15 - 10.32	MSSVs lift
19.0	End of simulation

Table 9-2

Three RCP Operation

Sequence of Events

Time (sec)	Event Description
0.0	One RCP trips
3.05	Flux/flow reactor trip setpoint reached
4.25	Rod motion begins
4.41	Turbine trip on reactor trip
6.56 - 8.93	MSSVs lift
19.0	End of simulation

Figure 9-1
LOSS OF FLOW
2 RCPS COASTDOWN
FROM 4 RCP OPERATION

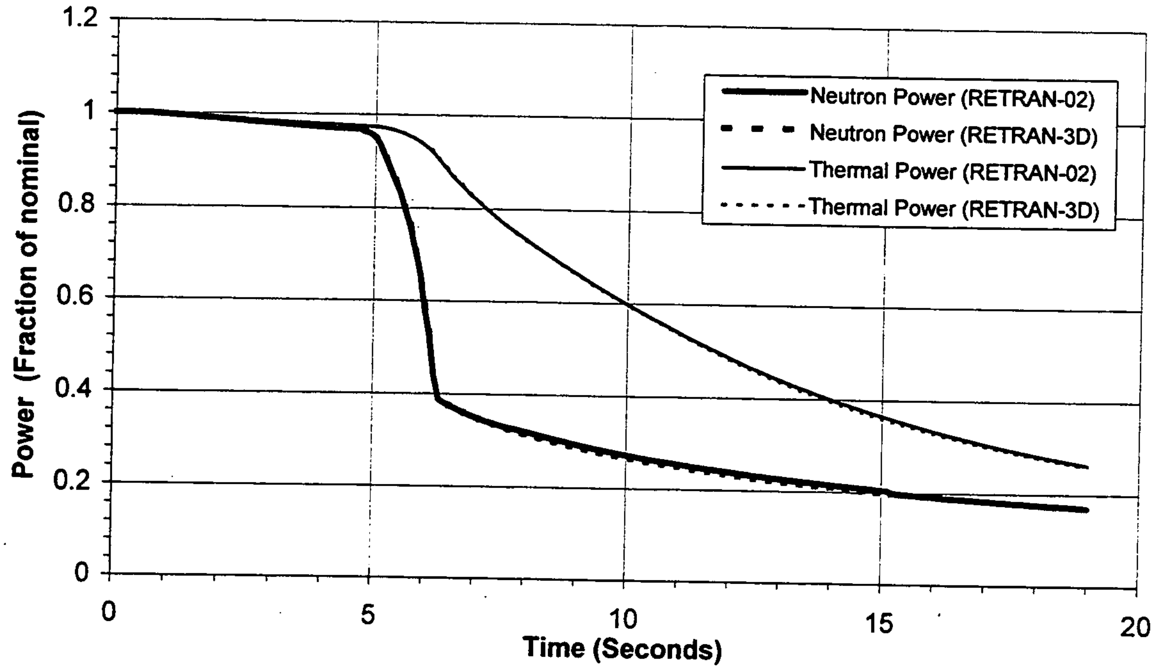


Figure 9-2
LOSS OF FLOW
2 RCPS COASTDOWN
FROM 4 RCP OPERATION

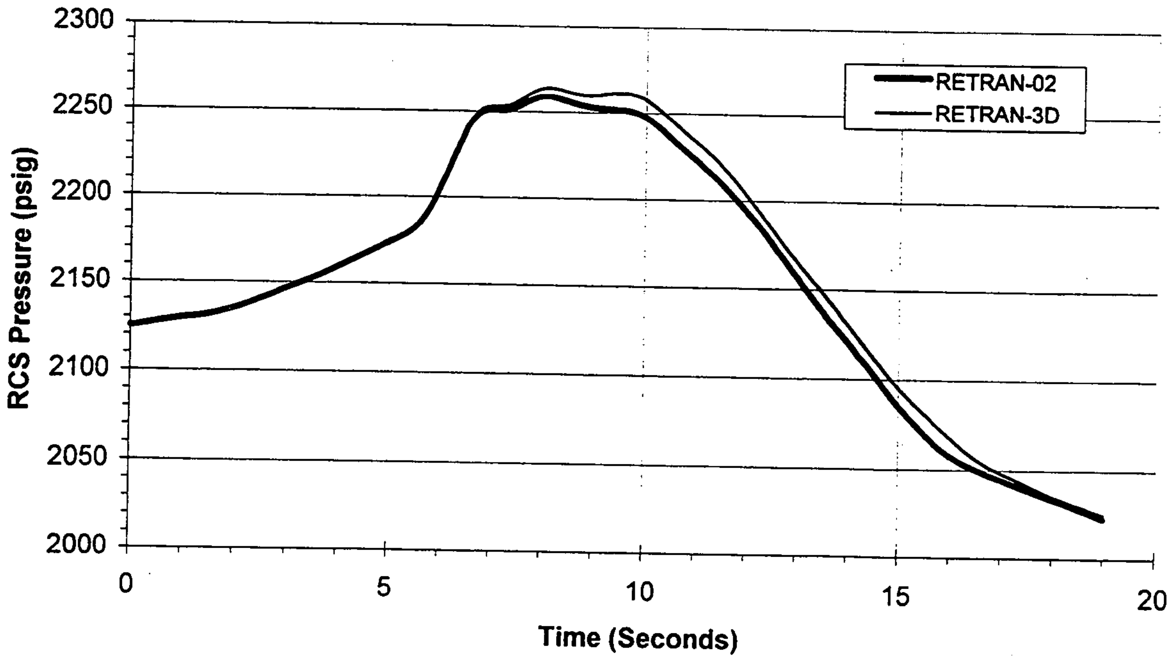


Figure 9-3

LOSS OF FLOW

2 RCPS COASTDOWN

FROM 4 RCP OPERATION

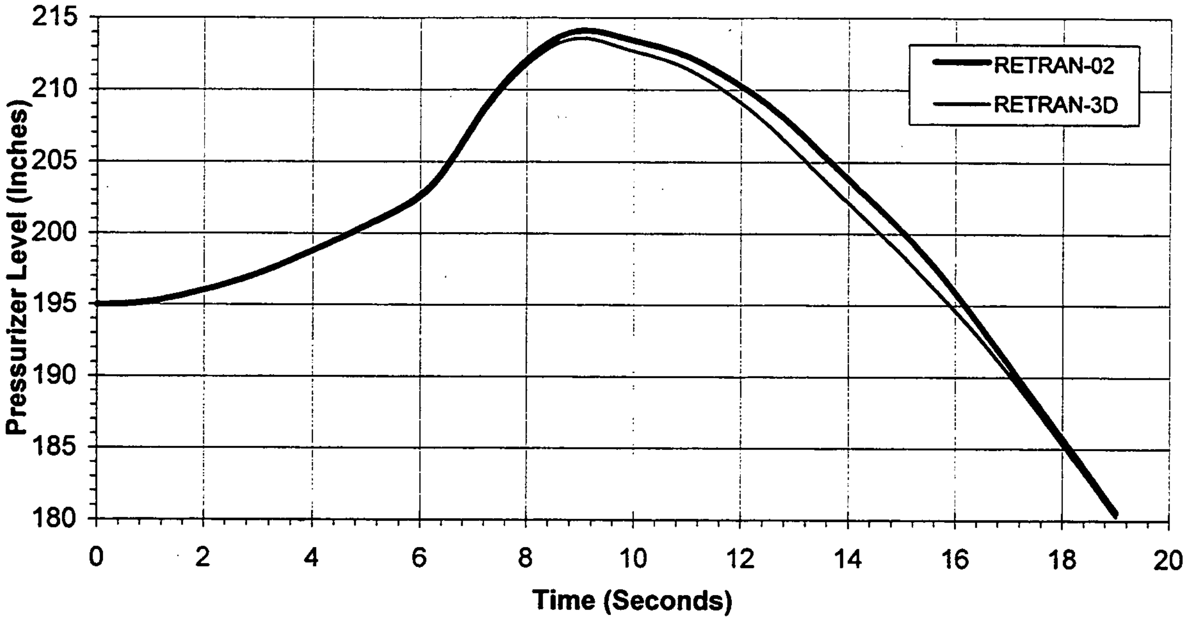


Figure 9-4

LOSS OF FLOW

2 RCPS COASTDOWN

FROM 4 RCP OPERATION

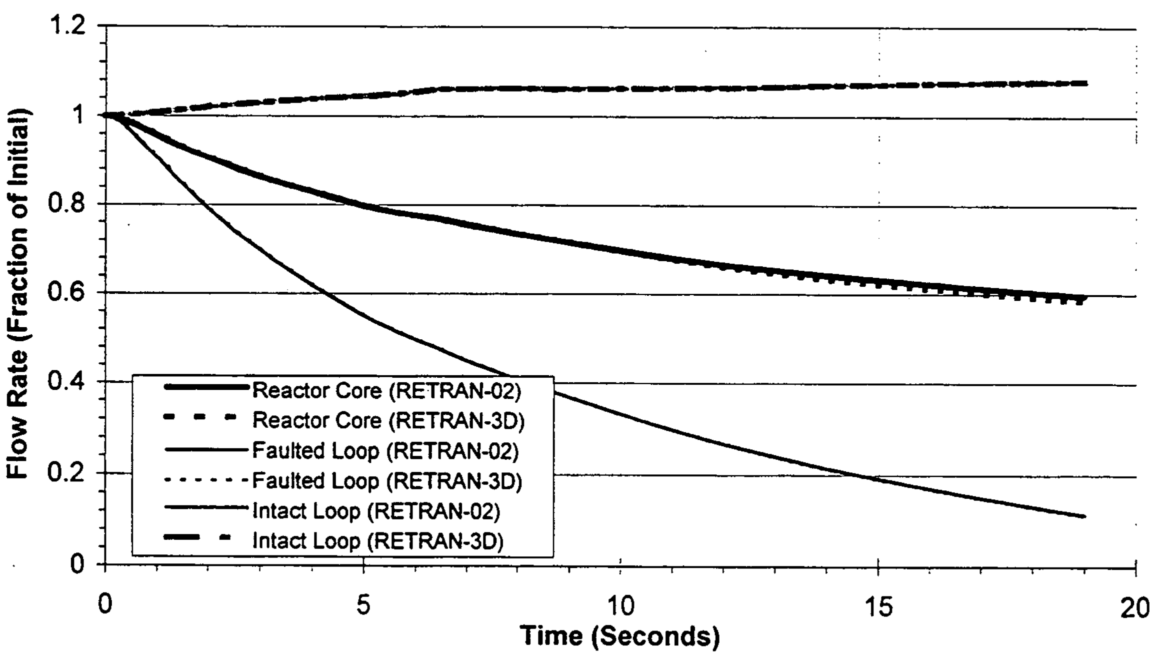


Figure 9-5

LOSS OF FLOW

2 RCPS COASTDOWN

FROM 4 RCP OPERATION

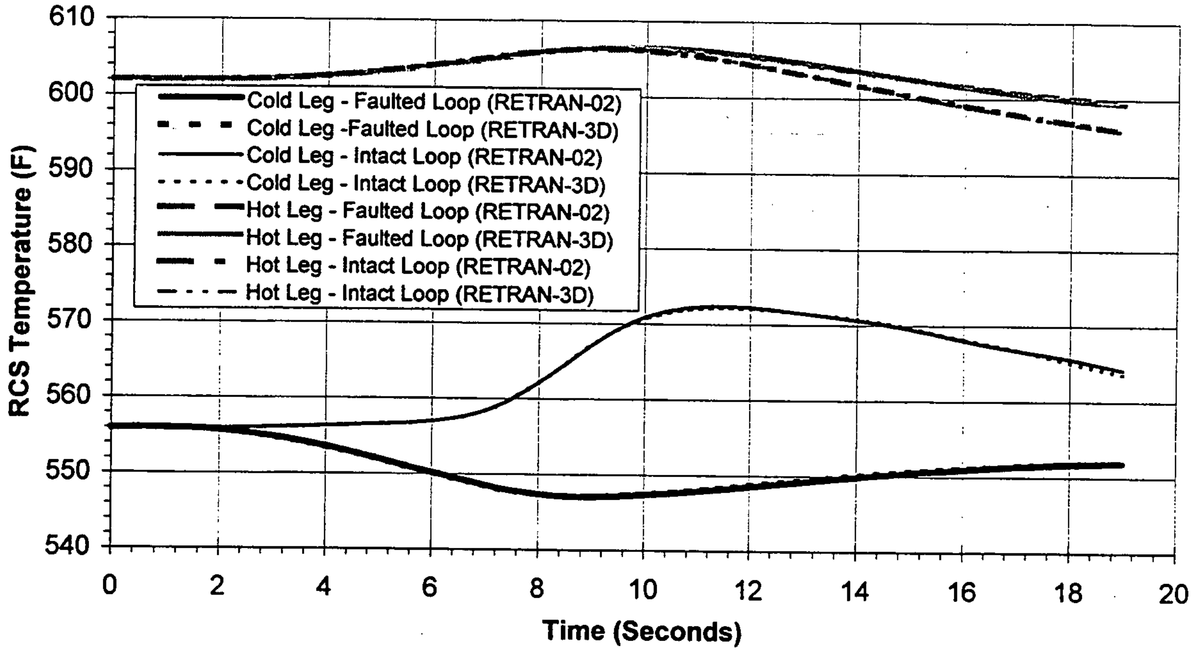


Figure 9-6

LOSS OF FLOW

1 RCP COASTDOWN

FROM 3 RCP OPERATION

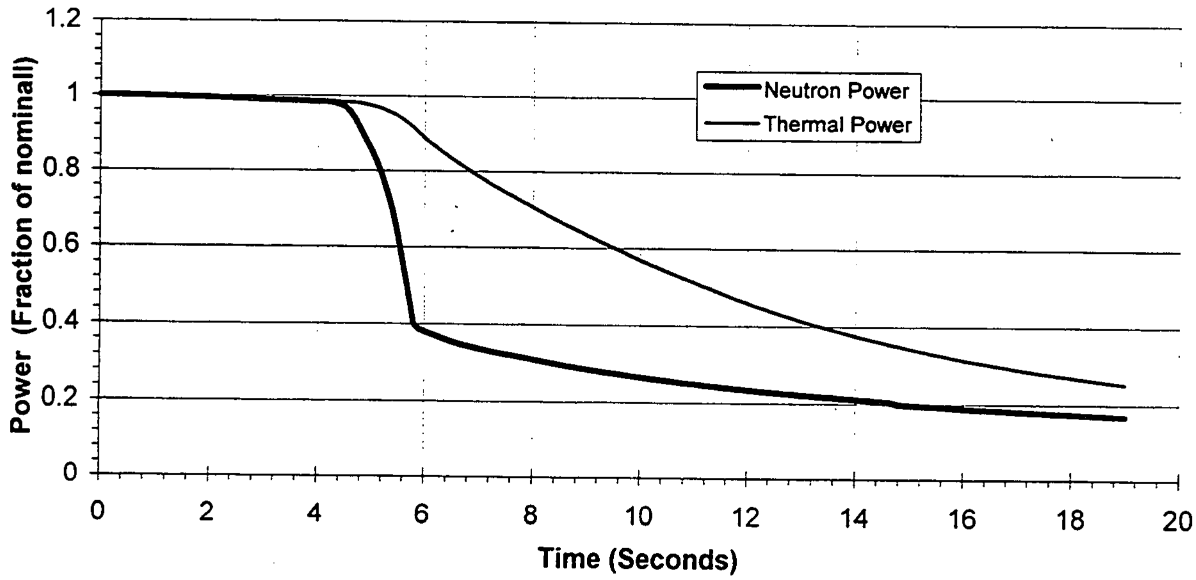


Figure 9-7

LOSS OF FLOW
1 RCP COASDOWN
FROM 3 RCP OPERATION

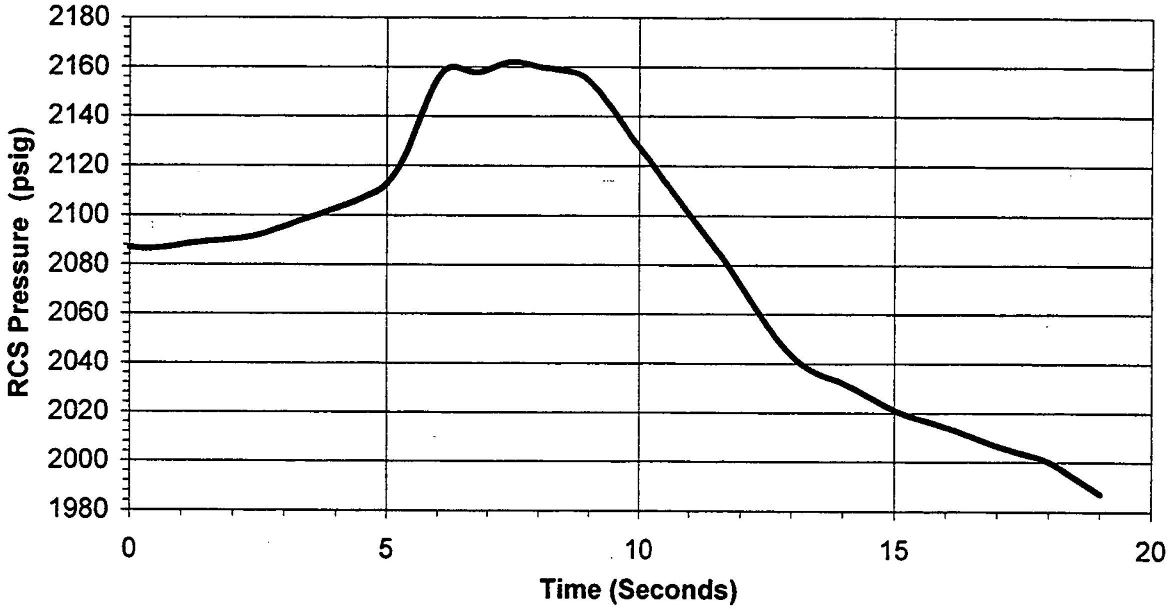


Figure 9-8

LOSS OF FLOW
1 RCP COASDOWN
FROM 3 RCP OPERATION

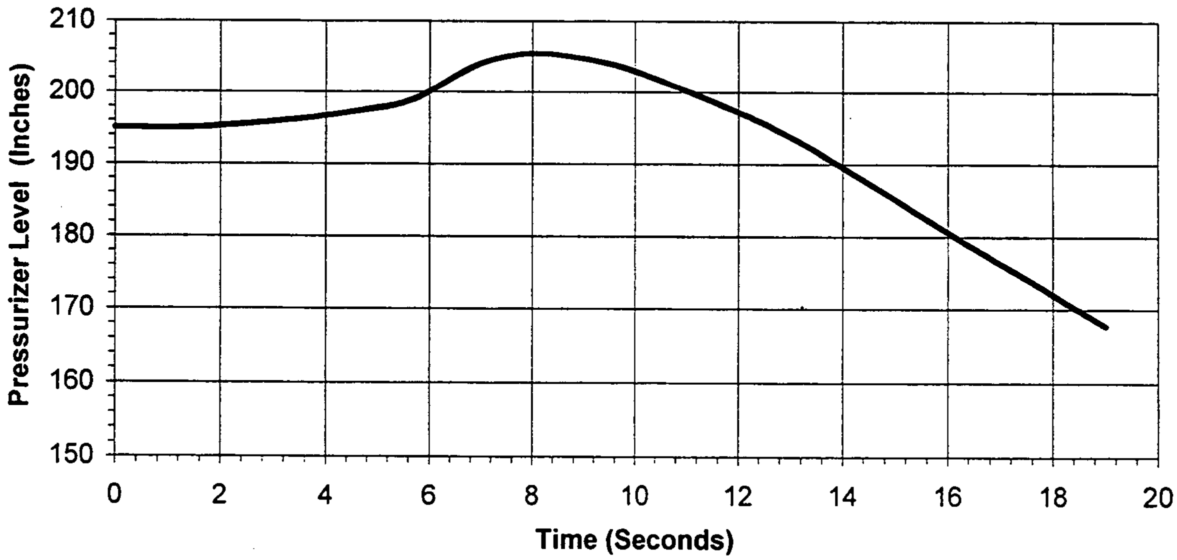


Figure 9-9

LOSS OF FLOW

1 RCP COASTDOWN

FROM 3 RCP OPERATION

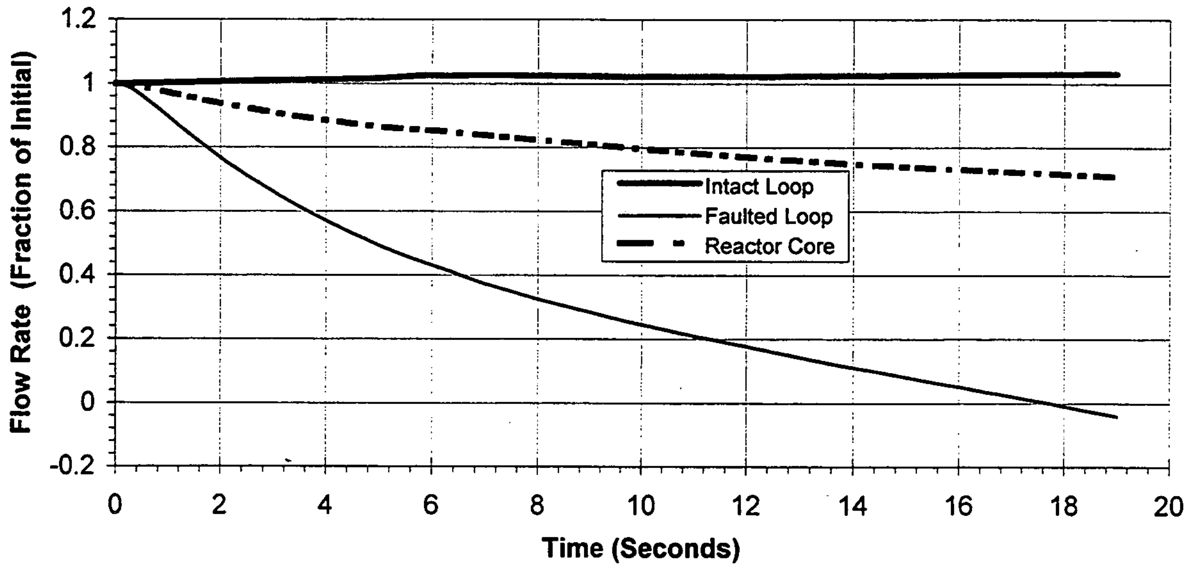


Figure 9-10

LOSS OF FLOW

1 RCP COASTDOWN

FROM 3 RCP OPERATION

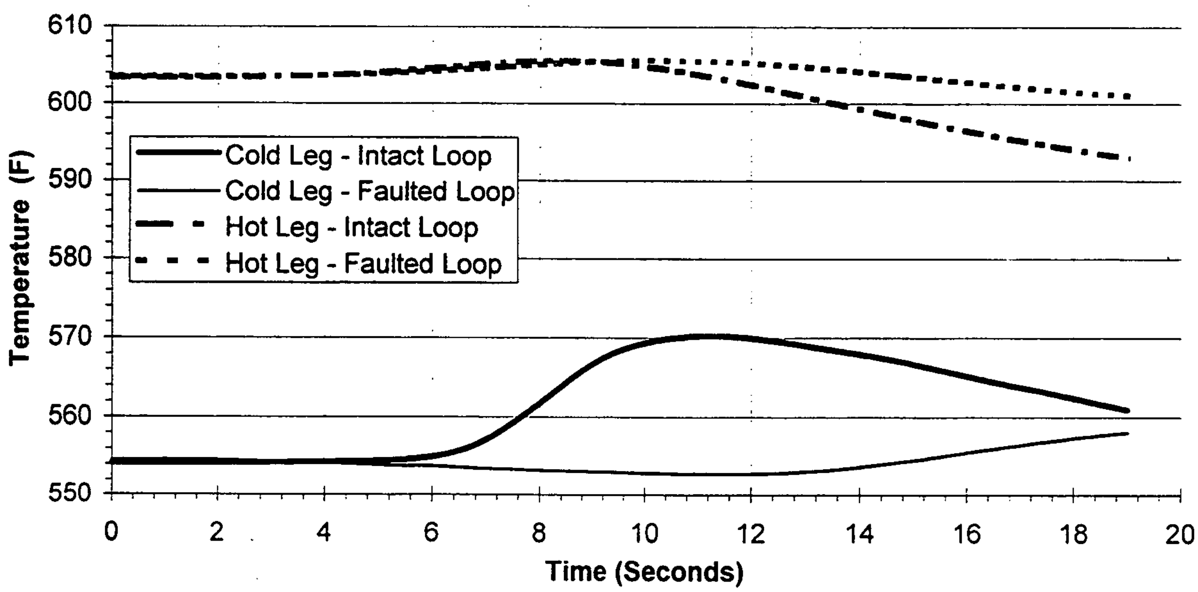


Figure 9-11
LOSS OF FLOW
2 RCPS COASTDOWN
FROM 4 RCP OPERATION

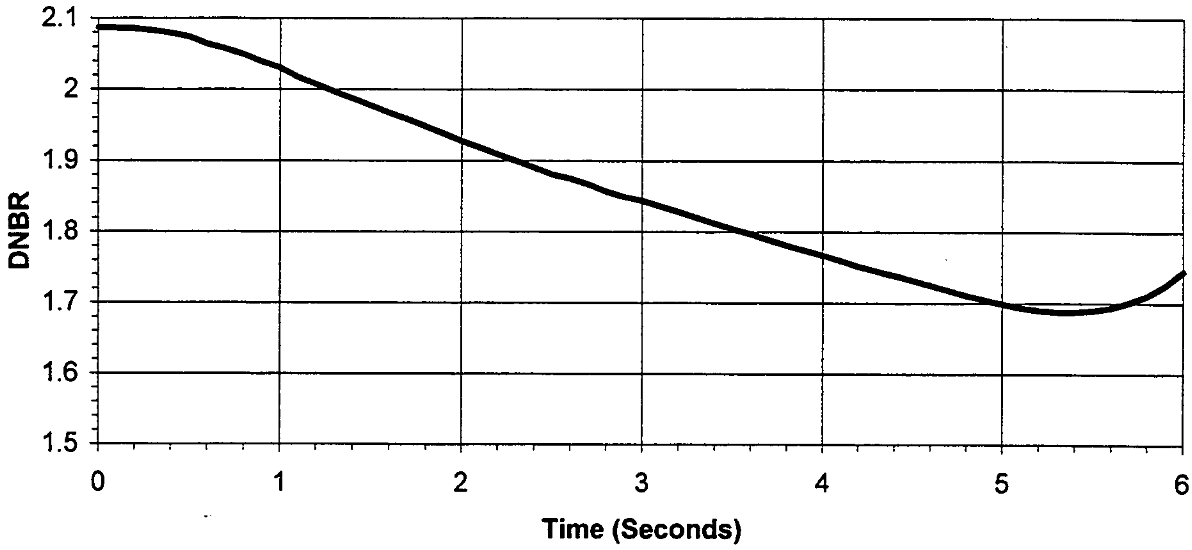
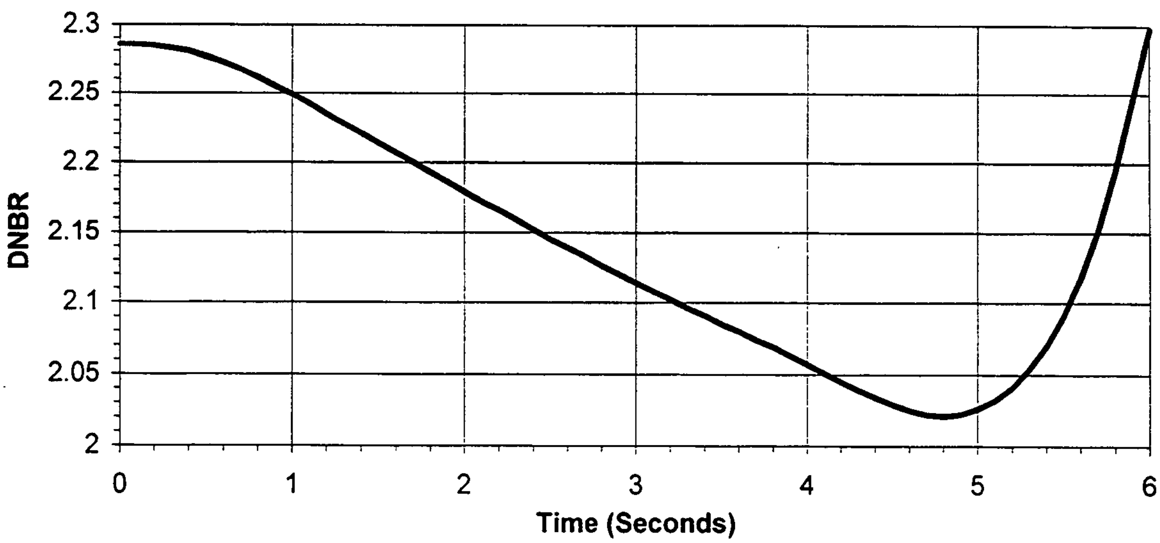


Figure 9-12
LOSS OF FLOW
1 RCP COASTDOWN
FROM 3 RCP OPERATION



10.0 LOCKED ROTOR

10.1 Overview

10.1.1 Description

The locked rotor accident is the result of an instantaneous seizure of one reactor coolant pump (RCP) rotor. Coolant flow in that loop rapidly decreases, causing the Reactor Protective System (RPS) to initiate a reactor trip on flux/flow/imbalance. The mismatch between power generation and heat removal capability due to the degraded flow condition causes a heatup of the primary system. The major concern in the locked rotor accident is departure from nucleate boiling (DNB). Based on the analysis results, peak RCS pressure is not a concern during this event.

10.1.2 Acceptance Criteria

The acceptance criteria for the locked rotor accident are:

- Any fuel damage calculated to occur must be of a sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.
- Peak RCS pressure remains below 110% of design pressure
- Any activity release must be such that the calculated doses at the site boundary are less than 100% of the 10 CFR Part 100 guidelines.

10.1.3 Analytical Approach

The locked rotor accident requires a limiting set of physics parameters along with conservative initial and boundary conditions. These parameters are input to the Oconee RETRAN-02 model (References 10-1 and 10-2) for the system thermal-hydraulic analysis. The RETRAN-02 analysis generates the transient forcing functions to be input to the Oconee VIPRE-01 (Reference 10-3) model. The VIPRE-01 model calculates the maximum allowable radial peaking (MARP) limits during the transient such that DNB will not occur. The MARP results are used to determine the number of fuel pins in the core exceeding the DNB limit, which are considered to be failed fuel pins.

10.2 Simulation Codes and Models

10.2.1 RETRAN-02

The RETRAN-02 code is used for the locked rotor system analyses. The two-loop Oconee RETRAN model is used for the locked rotor analysis. This two-loop model is used due to the asymmetric nature of the locked rotor accident. RETRAN predicts core exit pressure, core inlet temperature, core inlet flow and neutron power for the locked rotor events. The RETRAN analysis initial conditions are based on the statistical core design (SCD) methodology.

10.2.2 VIPRE-01

The VIPRE-01 code is used for the locked rotor core thermal-hydraulic analyses. VIPRE thermal-hydraulic boundary conditions are obtained from the RETRAN system transient simulation. The[ ]channel VIPRE model with fuel conduction described in Section 2.3.2 of this report is used to calculate the MARP limits. The critical heat flux (CHF) correlations used to evaluate DNBR are the BWC or BWU-Z CHF correlations (References 10-4 and 10-5). The VIPRE analysis employs the SCD methodology for the locked rotor core thermal-hydraulic analyses.

10.2.3 SIMULATE-3P

SIMULATE-3P (Reference 10-6) is a core neutronics code used to generate safety analysis physics parameters and three-dimensional core pin power distributions for the locked rotor accident. The RETRAN predicted conditions of the locked rotor accident will be input to SIMULATE-3P which is used to calculate the pin power distribution at these conditions. The pin power distribution will then be compared to the MARP limits generated in the VIPRE analysis to determine if DNB will occur during the transient, and the number of failed fuel pins.

10.3.2.6 Heat Transfer Correlations

For the DNBR calculations, only the single-phase forced convection and nucleate boiling heat transfer modes are applicable. The [ ] is used for the single-phase forced convection mode. The [ ] is used for the nucleate boiling region. The critical heat flux correlation used to define the peak of the boiling curve is the same as that used to predict the DNBR.

10.3.2.7 Results

The transient VIPRE minimum DNBR for the RCP locked rotor from four and three-pump operation are shown below (Figures 10-11 and 10-12):

RCP Locked Rotor Case	MDNBR	MDNBR at Time (seconds)
4-pump	1.50	2.1
3-pump	1.33	2.2

The above statepoints at which the minimum DNBR occur are used to determine the SCD limits for the RCP locked rotor from four and three-pump initial conditions. The minimum DNBR results shown in the above table are less than the respective SCD limits. MARP limits are therefore generated such that the SDLs are not exceeded. The MARP results are shown in Figures 10-13 and 10-14.

10.3.3 Fuel Pin Census

The MARP limits as a function of axial peak and location are used for the fuel pin census. When the radial power peak of the fuel pin predicted by SIMULATE-3P exceeds the MARP limit during the transient, DNB and cladding failure is assumed to occur. The fuel pin census is performed to determine the number of failed fuel pins during the locked rotor accident. Results are shown below.

Operating Conditions	% of Fuel Pins Experiencing DNB
4 RCP operation	0
3 RCP operation	0

10.4 Reload Cycle-Specific Evaluation

Physics parameters that are checked for each reload core are:

- Moderator temperature coefficient
- Doppler temperature coefficient
- Minimum scram worth curve

10.5 References

10-1 RETRAN-02: A Program for transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Revision 4, EPRI, November 1988

10-2 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000, Duke Power Company, July 1987

10-3 VIPRE-01 : A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM-A, Revision 3, August 1989

10-4 BAW-10143-PA, BWC Correlation of Critical Heat Flux, April 1985

10-5 BAW-10199-PA, The BWU Critical Heat Flux Correlation, April 1996

10-6 DPC-NE-1004A, Duke Power Company Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, November 1992

10-7 TACO-3, Fuel Pin Thermal Analysis Code, BAW-10162P-A, BWFC, November 1989

10-6 DPC-NE-1004A, Duke Power Company Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, November 1992

10-7 TACO-3, Fuel Pin Thermal Analysis Code, BAW-10162P-A, BWFC, November 1989

Table 10-1

Four RCP Operation

Sequence of Events

Time (sec)	Event Description
0.0	Locked rotor occurs
0.51	Flux/flow reactor trip setpoint reached
1.71	Rod motion begins
1.87	Turbine trip on reactor trip
3.01	Pressurizer spray initiates
4.18 - 7.05	MSSVs lift
9.00	Simulation ends

Table 10-2

Three RCP Operation

Sequence of Events

Time (sec)	Event Description
0.0	Locked rotor occurs
0.13	Flux/flow reactor trip setpoint reached
1.33	Rod motion begins
1.49	Turbine trip on reactor trip
3.27	Pressurizer spray initiates
5.04 - 6.55	MSSVs lift
9.0	Simulation ends

Figure 10-1

LOCKED ROTOR

LOCKED ROTOR FROM 4 RCP OPERATION

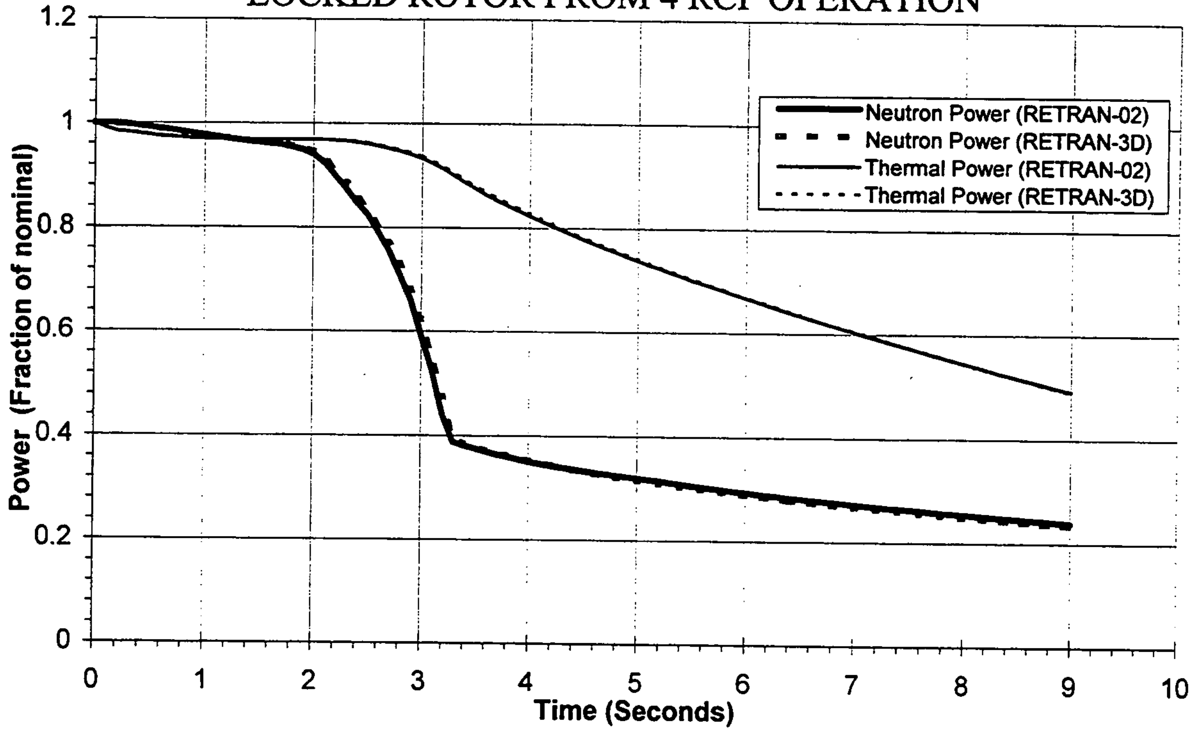


Figure 10-2

LOCKED ROTOR

LOCKED ROTOR FROM 4 RCP OPERATION

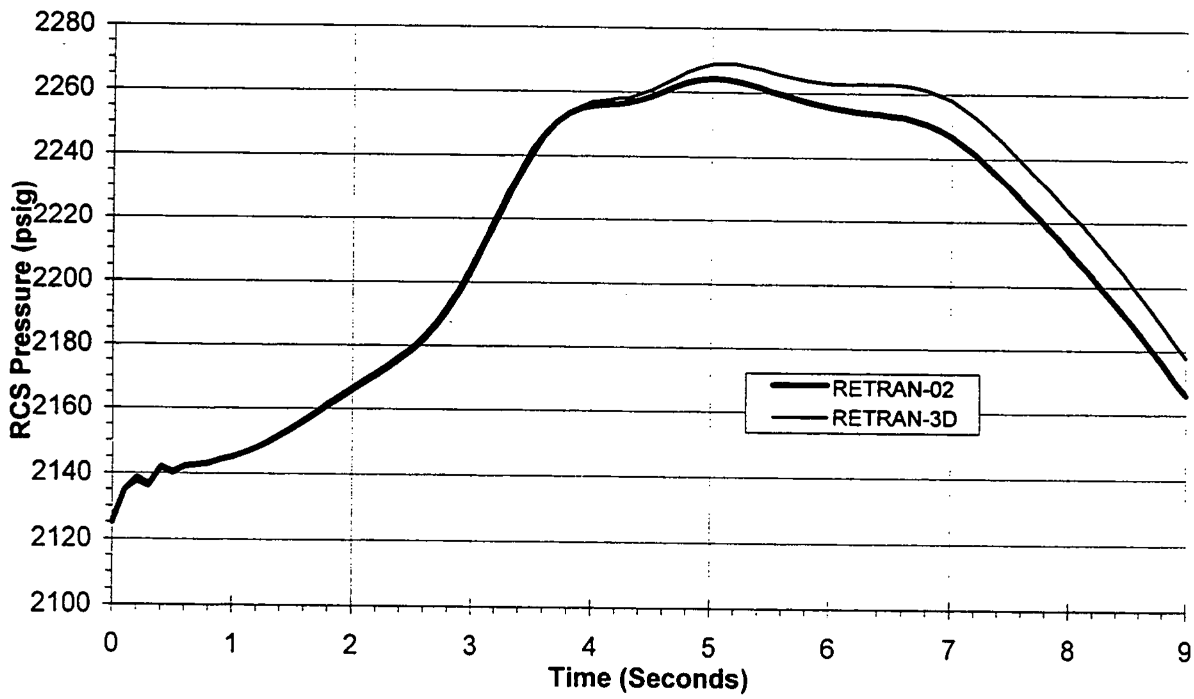


Figure 10-3
LOCKED ROTOR
LOCKED ROTOR FROM 4 RCP OPERATION

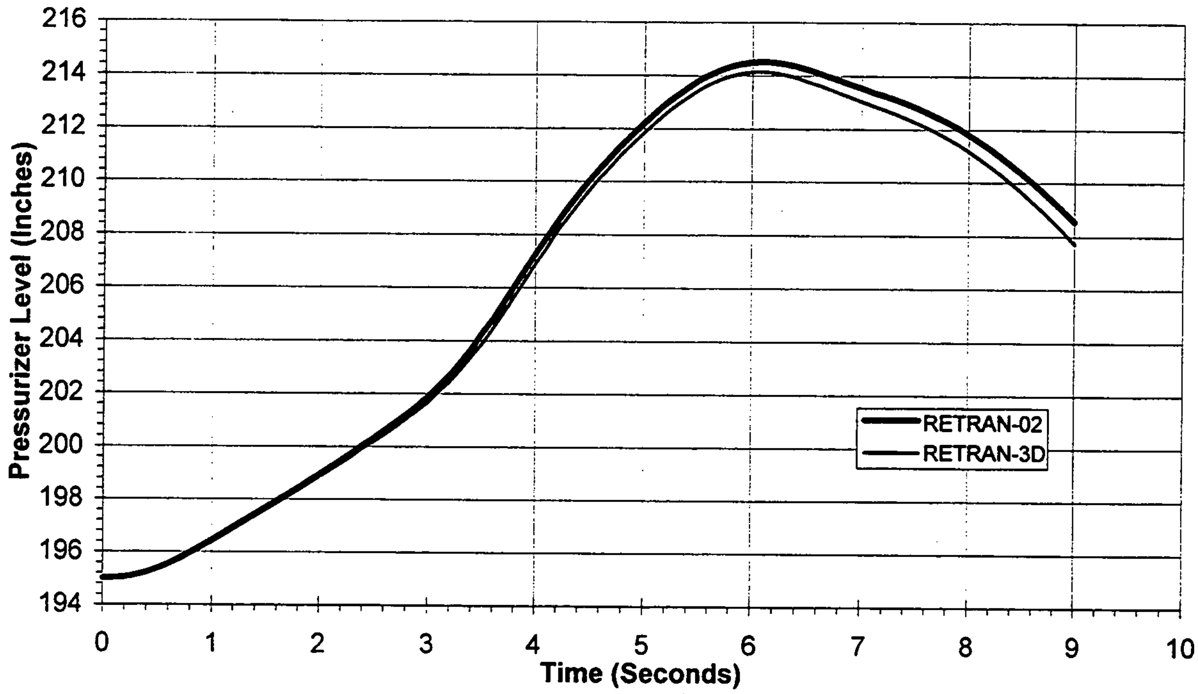


Figure 10-4
LOCKED ROTOR
LOCKED ROTOR FROM 4 RCP OPERATION

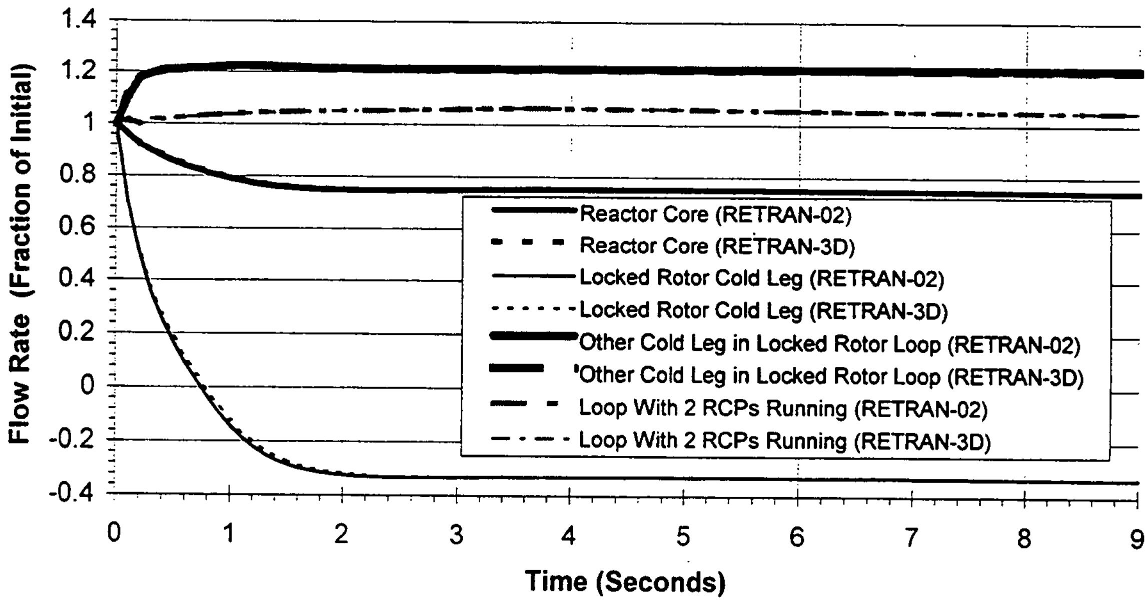


Figure 10-5
LOCKED ROTOR
LOCKED ROTOR FROM 4 RCP OPERATION

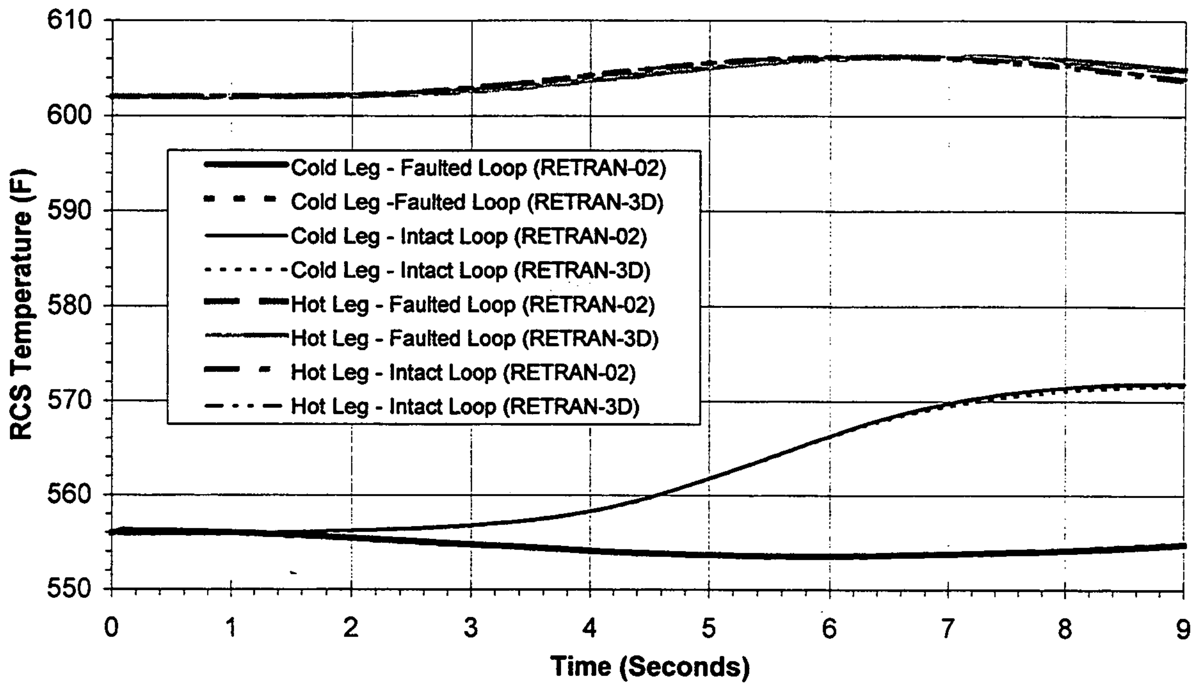


Figure 10-6
LOCKED ROTOR
LOCKED ROTOR FROM 3 RCP OPERATION

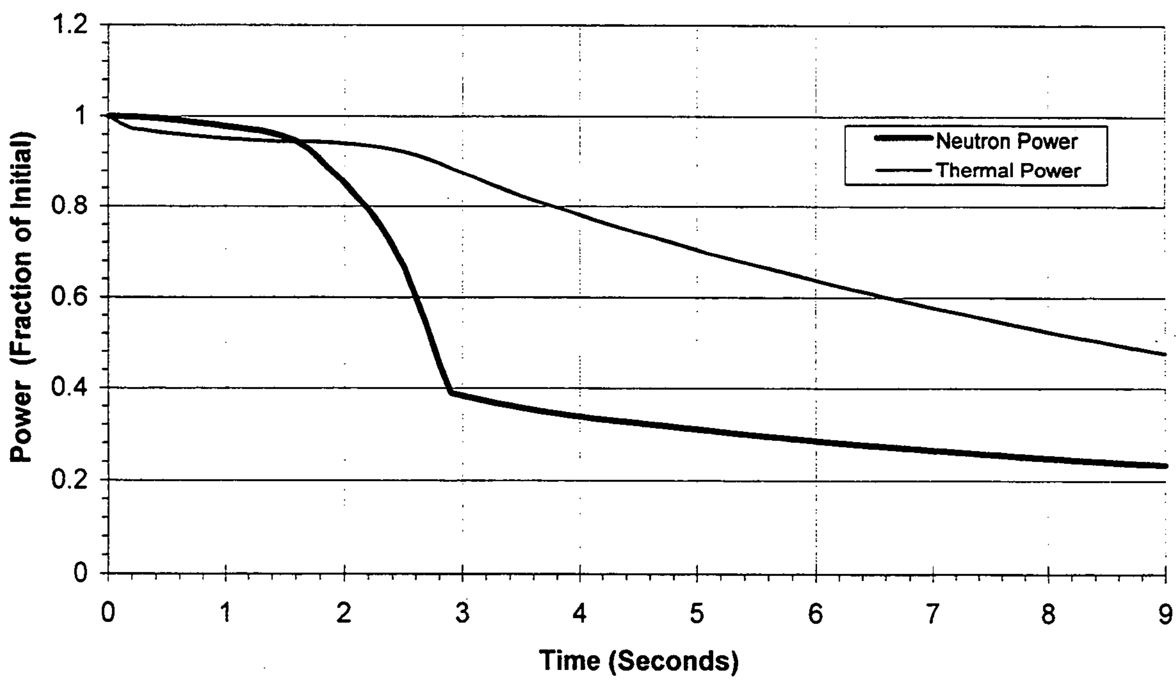


Figure 10-7
LOCKED ROTOR
LOCKED ROTOR FROM 3 RCP OPERATION

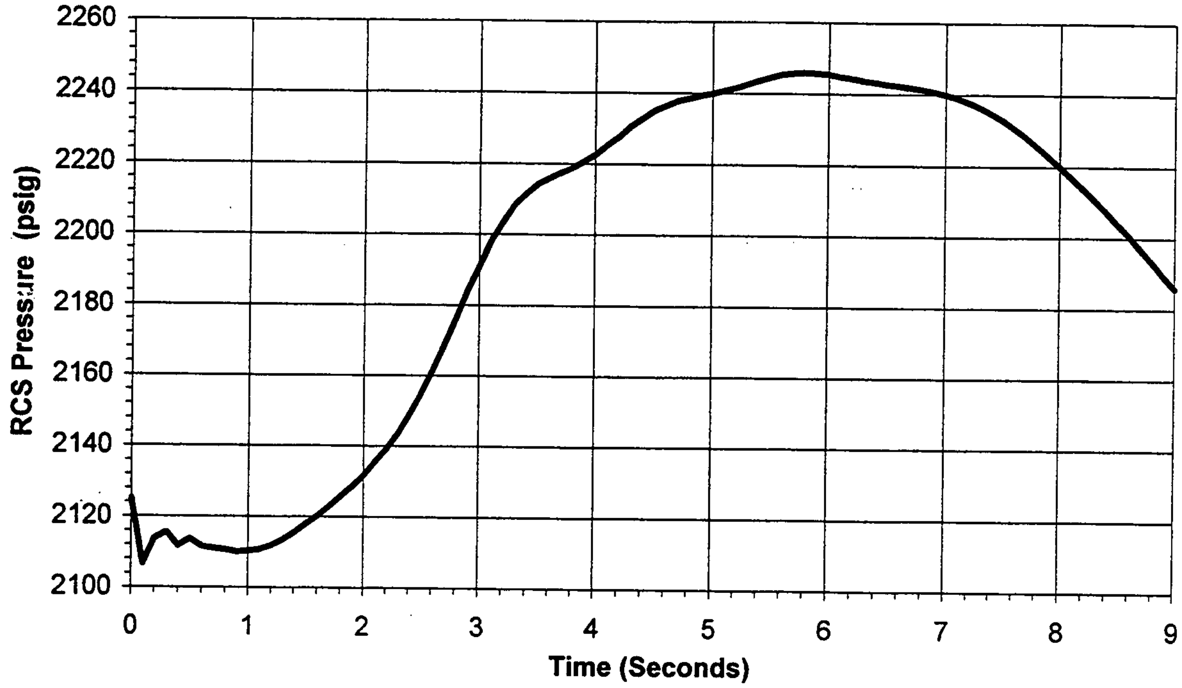


Figure 10-8
LOCKED ROTOR
LOCKED ROTOR FROM 3 RCP OPERATION

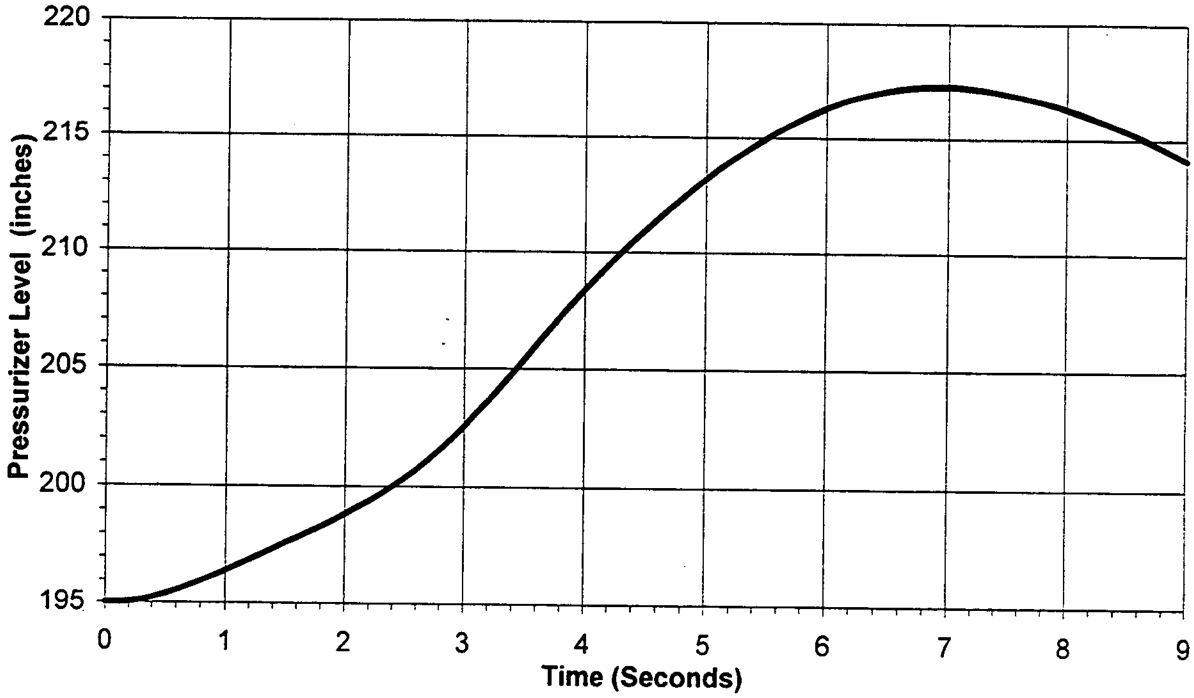


Figure 10-9
LOCKED ROTOR
LOCKED ROTOR FROM 3 RCP OPERATION

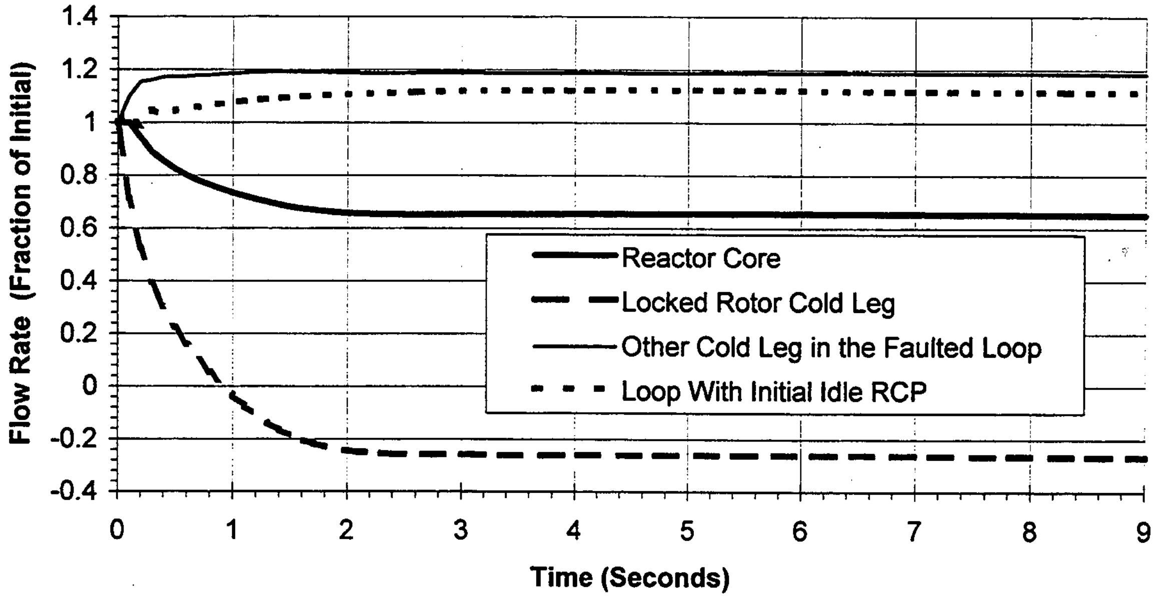


Figure 10-10
LOCKED ROTOR
LOCKED ROTOR FROM 3 RCP OPERATION

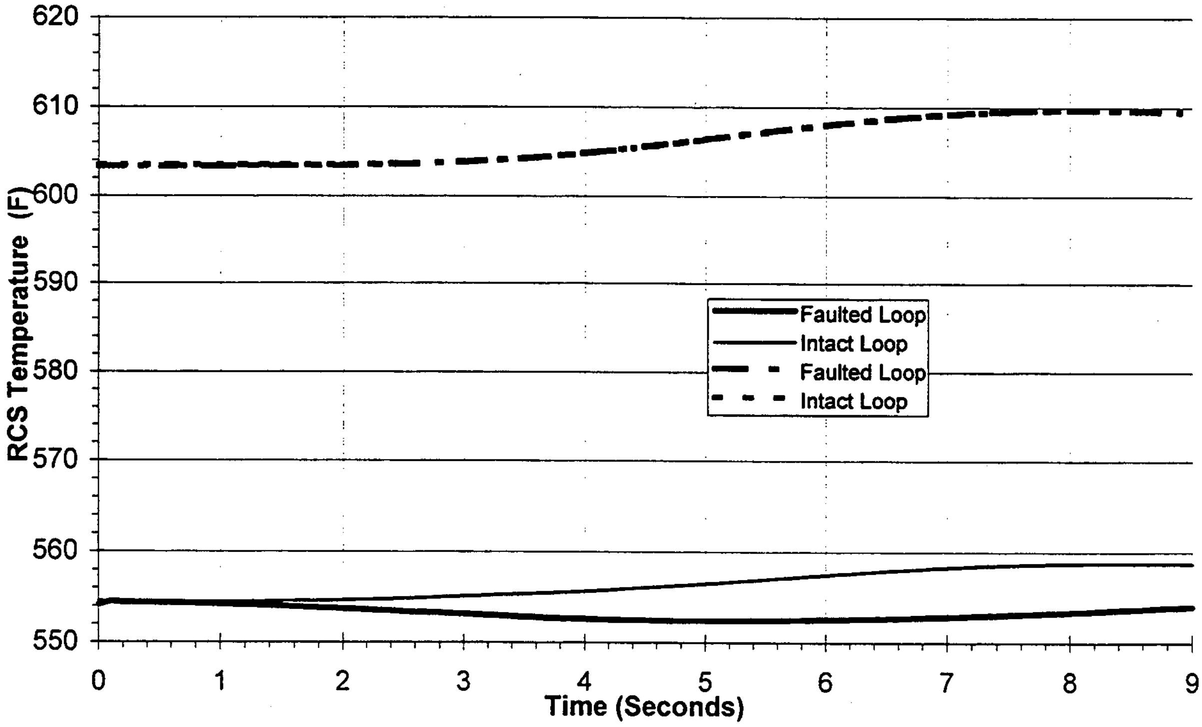


Figure 10-11

LOCKED ROTOR

LOCKED ROTOR FROM 4 RCP OPERATION

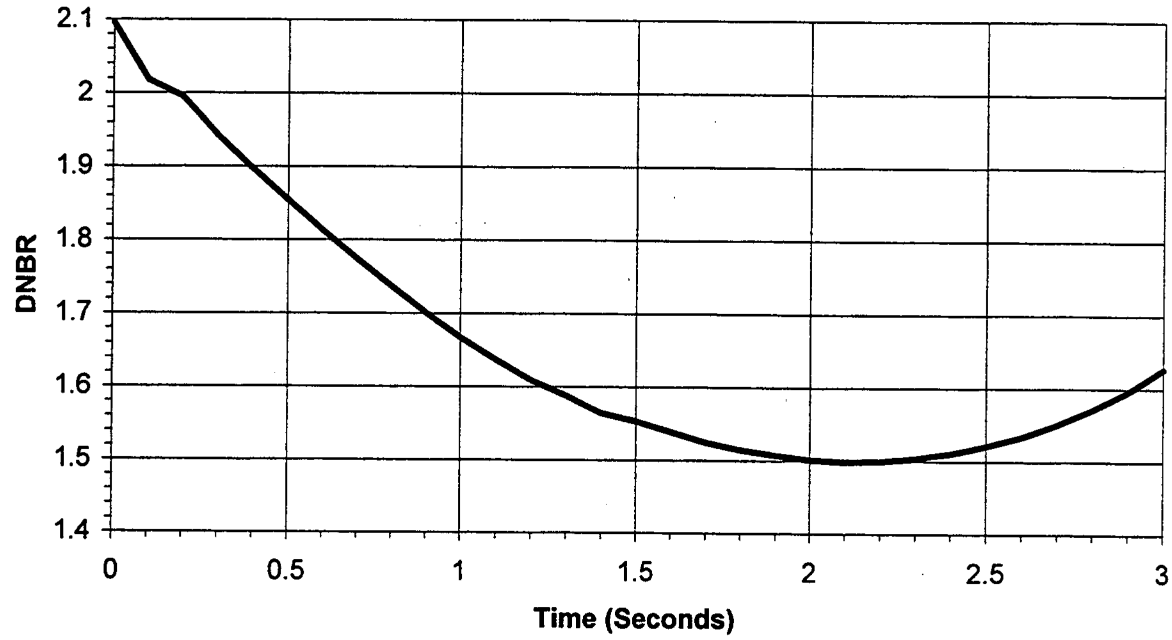


Figure 10-12

LOCKED ROTOR

LOCKED ROTOR FROM 3 RCP OPERATION

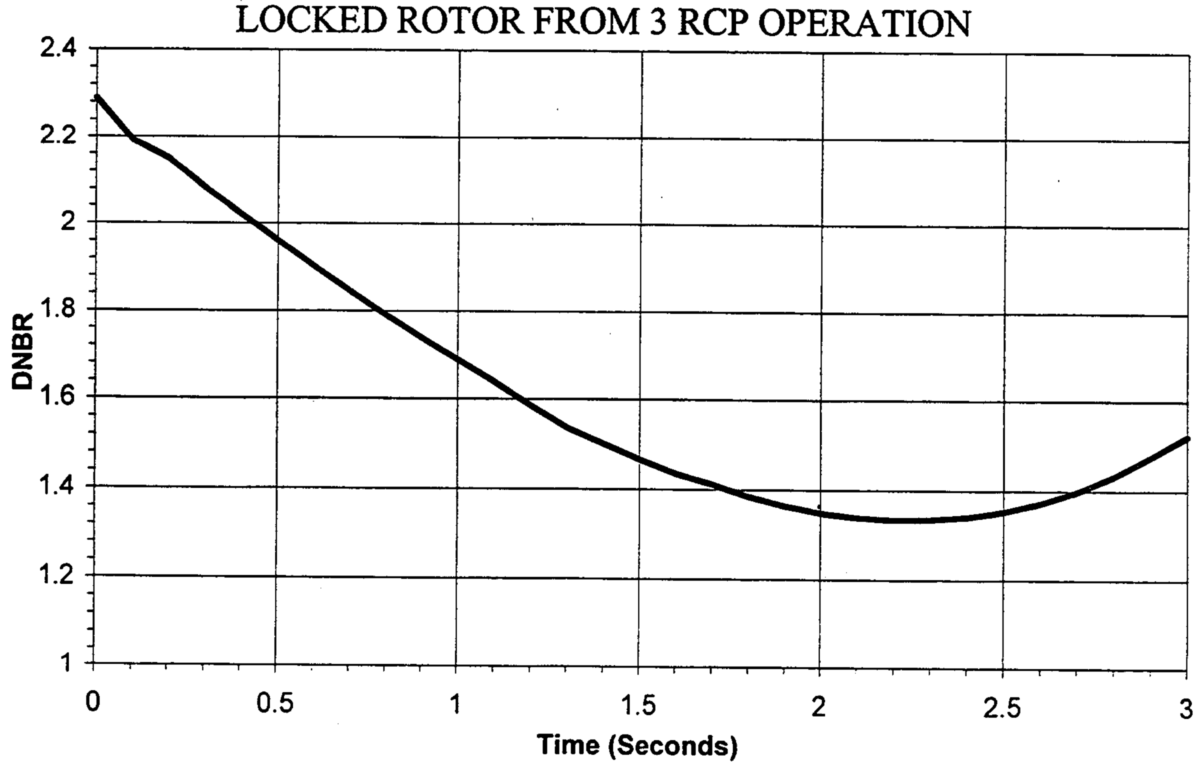


Figure 10-13

LOCKED ROTOR

LOCKED ROTOR FROM 4 RCP OPERATION

Figure 10-14

LOCKED ROTOR

LOCKED ROTOR FROM 3 RCP OPERATION

11.0 CONTROL ROD MISALIGNMENT ACCIDENT

11.1 Dropped Control Rod

Control rods are normally grouped into patterns which maintain a symmetric core power distribution. A mechanical or electrical failure can cause a control rod to drop partially or fully into the core. The resulting transient causes a rapid reduction in power and moderator temperature which is followed by an increase in power due to the negative moderator temperature coefficient. If control rods are withdrawn by the Integrated Control System (ICS) during the transient response, they will add to the increase in power. The magnitude of the power increase may exceed the initial power level. This elevated power level, with consideration for the asymmetric power distribution, has the potential for the DNBR and centerline fuel melt (CFM) limits to be exceeded.

The general response of the analog ICS to a dropped rod event is as follows, assuming no credit for the asymmetric rod indication generating either an ICS runback signal or a control rod withdrawal inhibit signal. This conservative assumption is in addition to the normal assumption that control systems either function as designed or do not respond, whichever results in the worst transient response. The plant is assumed to be initially at 100% full power conditions with the ICS in automatic. The initiating event is a control rod dropping into the core. The negative reactivity inserted into the core causes an immediate drop in core power and is assumed to produce a significant quadrant tilt. The difference between the reactor demand and indicated core power signals (neutron error) causes the Group 7 control rods to be withdrawn. A neutron cross limit signal is generated when the neutron error signal reaches 5%, causing the unit to go into tracking mode. In tracking mode, the ULD follows generated megawatts. Turbine control will try to maintain turbine header pressure at its nominal setpoint of 885 psig. The neutron cross limit will also impact the feedwater demand signal to keep core power and feedwater flow coordinated. As Group 7 is withdrawn, actual core power may increase above the initial power level resulting in an increase in steam pressure. The turbine control response will be to open the turbine control valves to maintain header pressure. If the assumed quadrant tilt is large enough, the unit will remain in track for the duration of the transient. If no quadrant tilt is assumed, the unit will come out of track when the neutron cross limit clears and will hold at the then current demand set value.

The analog ICS has been replaced by an advanced digital ICS. The same modeling philosophy presented for the analog ICS will be used in analyzing the plant response to a dropped rod event with the digital ICS.

The transient response is analyzed with the RETRAN-02 code (Reference 11-1). The DNB analysis is performed with the VIPRE-01 code (Reference 11-2). The core power distribution is analyzed with the SIMULATE-3P code (Reference 11-3). The acceptance criteria for this analysis are to ensure that the minimum DNBR remains above the DNBR limit, the CFM limits are not exceeded, and that the pressure in the Reactor Coolant System (RCS) remains below 110% of design pressure. The minimum DNBR is determined using the statistical core design (SCD) methodology. Based on the analysis results, peak RCS pressure is not a concern during this event. The initial conditions and boundary conditions chosen for this analysis are therefore those that will result in the lowest DNBR and the highest linear heat rates.

11.1.1 Nodalization

This transient is analyzed using the Oconee two-loop RETRAN model (Reference 11-4). This permits the evaluation of cases with both three and four-pump operation. A junction is added to the base model to connect the steam lines since an asymmetric steam generator response will occur during cases with three-pump operation.

11.1.2 Initial Conditions

<u>Core Power Level</u>
A high initial power level for both three and four RCP operation maximizes the primary system heat flux. The uncertainty for this parameter is incorporated in the SCD methodology.

<u>RCS Pressure</u>
Low initial RCS pressure is conservative for DNBR. The SCD accounts for instrument uncertainty in the pressure indication, but does not account for a controller deadband bias. Nominal pressure less the controller deadband bias is therefore assumed for the initial RCS pressure.

Pressurizer Level

Low initial level increases the volume of the pressurizer steam space which minimizes the pressure increase resulting from the power increase and subsequent insurge.

RCS Temperature

The nominal average temperature is assumed, with the uncertainty for this parameter incorporated in the SCD methodology.

RCS Flow

Low initial flow is conservative for DNB analyses. The uncertainty associated with this parameter is incorporated in the SCD methodology.

Core Bypass Flow

A high core bypass flow is assumed to minimize the coolant flow along the fuel rods.

Fuel Temperature

A smaller increase in fuel temperature results during this transient when starting from a lower initial fuel temperature. With a negative Doppler coefficient, this results in a smaller negative reactivity insertion which will result in a higher peak power. Therefore, a low initial fuel temperature is assumed.

Steam Generator Mass

High initial steam generator mass is assumed to minimize the increase in average moderator temperature. With a negative moderator temperature coefficient, this results in a smaller negative reactivity insertion which will result in a higher peak power.

Steam Generator Tube Plugging

Low steam generator tube plugging is assumed to maximize primary-to-secondary heat transfer which will limit RCS heatup.

11.1.3 Boundary Conditions

Dropped Rod

A range of dropped rod worths and core locations are analyzed to bound all possible single dropped rods at full power for four pumps in operation, and at 80% power for three pumps in operation.

Pressurizer Inter-Region Heat Transfer Coefficient

A high inter-region heat transfer coefficient is assumed to maximize the heat transfer rate at the liquid-steam interface of the pressurizer. This will minimize RCS pressurization as power increases.

Single Failure

The excore power range flux detector in the quadrant with the highest neutron flux response is assumed to fail.

11.1.4 Physics Parameters

Moderator Temperature Coefficient

At BOC a most positive/least negative moderator temperature coefficient is assumed to minimize the negative reactivity feedback during the power increase. At EOC a most negative moderator temperature coefficient is assumed to maximize the positive reactivity feedback during the cooldown.

Doppler Temperature Coefficient

At BOC a least negative Doppler temperature coefficient is assumed to minimize the negative reactivity feedback during the power increase. At EOC a most negative Doppler temperature coefficient is assumed to maximize the positive reactivity feedback during the cooldown.

Beta-Effective and Neutron Lifetime

Maximum and minimum values at both BOC and EOC are assumed in sensitivity analyses to determine the most conservative assumption.

Scram Curve and Worth

A conservatively slow BOC and EOC scram curve along with a minimum scram worth are assumed.

11.1.5 Control, Protection, and Safeguards Systems

Reactor Trip

A reactor trip signal may not occur as a result of this event. The reactor trip functions credited are the high flux trip and the high RCS pressure trip. The effect of reactor vessel downcomer temperature decrease and rod motion on the excore flux signal is considered. A conservative trip delay time is assumed.

Reactor Control

The reactor control subsystem of the ICS is assumed to be in automatic. A decrease in indicated power results in the ICS withdrawing rods to maintain the initial power level.

Excore Flux Instrumentation

The excore flux instrumentation is simulated to conservatively model two process effects characteristic of a dropped rod transient. The largest effect is the indicated quadrant tilt caused by the asymmetric power distribution resulting from the dropped rod. The second effect is attenuation of the flux signal on the excore detectors as the reactor vessel downcomer temperature decreases. The flux signal is compared to the ICS reactor demand signal to obtain a neutron error signal. This signal is input to the ICS rod control subsystem and used to determine the neutron cross-limit which is input to the ICS main feedwater subsystem.

RCS Pressure Control

Pressurizer spray is assumed to be operable to minimize the pressure increase resulting from the power increase. Pressurizer heaters are assumed to be inoperable.

Pressurizer Level Control

RCS makeup and letdown have a negligible impact on the results of this analysis.

Main Feedwater System

The main feedwater (MFW) subsystem of the ICS is assumed to be in automatic control. The ICS will adjust the MFW flow to the steam generators according to changes in the demand signal.

Turbine Control

The turbine control subsystem of the ICS is assumed to be in automatic. The ICS will attempt to maintain steam line pressure by opening or closing the turbine control valves. This action will tend to minimize temperature decreases, and maximize primary-to-secondary heat transfer and limit RCS heatup.

11.1.6 VIPRE-01 Analysis

The forcing functions necessary to perform the DNB analysis (core average heat flux, core inlet flow and temperature, core exit pressure) are obtained from the RETRAN-02 analysis results and input to VIPRE-01. The VIPRE-01[ ]channel model (Reference 11-4) is then used to determine the time of the minimum DNBR statepoint for the transient conditions analyzed. At these statepoint conditions a set of maximum allowable radial peak (MARP) curves is developed for determining if the DNBR limit is exceeded.

11.1.7 Results

The peak power levels for a dropped rod transient as predicted by RETRAN are 116.6% for the four-pump initial condition, and 93.6% for the three-pump initial condition. The results of the DNBR analysis have demonstrated that the power peaking will remain below the DNBR limit. The results of the CFM analysis have demonstrated that the maximum linear heat rate is less than the CFM limit.

11.1.8 Reload Cycle-Specific Evaluation

Physics parameters that are checked for each reload core are:

- Moderator temperature coefficient

- Doppler temperature coefficient
- Minimum scram worth curve
- Maximum withdrawable Group 7 control rod worth curve

11.2 Statically Misaligned Rod

The statically misaligned rod event considers the situation where a control rod is misaligned from the remainder of its bank. A rod misalignment may produce an increase in core peaking which decreases the margin to DNB. Steady-state three-dimensional power peaking analyses are performed with SIMULATE-3P to confirm that the asymmetric power distributions resulting from the rod misalignment will not result in DNB. There is no system transient associated with the analysis of the statically misaligned rod case. The reactor is assumed to remain at its initial power level.

The statically misaligned rod evaluation is performed at nominal hot full power conditions. Axial shapes allowed by the power dependent axial offset limits are considered in the evaluation. Two specific cases are analyzed which characterize the worst case misalignments. The first case considers the full insertion of any one rod within Group 7 positioned anywhere within the full power rod insertion limits. The second case considers the misalignment of a single Group 7 rod at its fully withdrawn position, with the remainder of Group 7 positioned at the full power rod insertion limit. A rod position uncertainty of 2% is considered.

The results of the generic evaluation of the statically misaligned rod event show that this event is bounded by the dropped rod event. Therefore, power distributions from the statically misaligned rod accident are not analyzed for each reload core.

11.3 References

11-1 RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI-NP-1850-CCM, Revision 4, EPRI, November 1988

11-2 VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM-A, Revision 3, EPRI, August 1989

11-3 SIMULATE-3: Advanced Three-Dimensional Two-Group Reactor Analysis Code, Studsvik/SOA-92/01, Studsvik of America, April 1992

11-4 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000, Duke Power Company, July 1987

12.0 TURBINE TRIP

12.1 Overview

12.1.1 Description

A turbine trip can result from any one of the following conditions: generator trip, low condenser vacuum, loss of lubrication oil, turbine thrust bearing failure, turbine overspeed, reactor trip, or a manual trip. If the reactor does not trip on turbine trip, the mismatch between the power generated in the primary system and the heat removed by the secondary system results in an increase in the primary system and secondary system temperatures and pressures. The temperatures and pressures will continue to increase until the reactor trips on high Reactor Coolant System (RCS) pressure.

12.1.2 Acceptance Criteria

The heatup and pressurization that result following the turbine trip event leads to concerns regarding peak primary system pressure and core cooling capability (or DNB). The acceptance criteria for the turbine trip event are that the peak primary system pressure shall not exceed 110% of the design pressure, and the DNBR limit shall not be exceeded. The DNB acceptance criterion is not challenged during a turbine trip transient.

12.1.3 Analytical Approach

The turbine trip event is analyzed with the RETRAN-02 code (Reference 12-1) for the full power four-pump operating condition. The peak primary system pressure response with three RCPs in operation is not analyzed because it is bounded by the four RCP operation case. Initial and boundary conditions are assumed to maximize the primary system pressure response. These parameters are input to the Oconee RETRAN-02 model (Reference 12-2) for the system thermal-hydraulic analysis.

12.2 Simulation Code and Model

The turbine trip analysis employs RETRAN-02 and the Oconee RETRAN two-loop base model (Reference 12-2) to simulate the plant response during the turbine trip event.

12.3 Peak Primary System Pressure Analysis

12.3.1 Initial Conditions

The initial conditions are chosen to maximize heat input to the primary system and to minimize heat transfer to the secondary system, thereby maximizing the heat-up and pressurization of the primary system.

Power Level

A high initial power level plus uncertainty is assumed to maximize the heat addition to the primary system and thus maximize the primary system pressure response.

RCS Pressure

A high initial pressure maximizes the actual pressure at the time of the reactor trip. A low initial pressure, however, maximizes the time from the start of the transient until the high RCS pressure reactor trip setpoint is reached. Delaying reactor trip will maximize heat input to the primary system. A high initial pressure plus uncertainty is assumed because it yields the most limiting peak primary pressure response.

Pressurizer Level

A high initial pressurizer level plus uncertainty is assumed to minimize the size of the steam bubble. A smaller steam bubble will lead to a more rapid pressurization than a larger steam bubble.

RCS Temperature

A high initial average temperature plus uncertainty is assumed to maximize the primary system stored energy, and thus maximize the transient primary system pressure response.

RCS Flow

A low initial flow less uncertainty is assumed to minimize the primary-to-secondary heat transfer so as to maximize the primary system pressurization.

Core Bypass Flow

A low core bypass flow is assumed to maximize the heat addition to the primary system and thus maximize the primary system pressure response.

Fuel Temperature

A bounding high initial fuel temperature is assumed to maximize the initial stored energy in the reactor core and thus maximize the transient primary system pressure response.

Steam Generator Mass

A low initial mass is assumed to minimize the secondary heat sink, thus maximizing the primary system heatup and pressure response.

Steam Generator Tube Plugging

A bounding high tube plugging level is assumed to minimize the primary system mass and the primary-to-secondary heat transfer. This will maximize the primary system pressure response.

12.3.2 Boundary Conditions

Turbine Stop Valves

A conservatively fast turbine stop valve closure stroke time is assumed.

Pressurizer Safety Valves

To maximize the primary system pressure response the pressurizer safety valves are modeled with conservative assumptions for drift, blowdown, and relief capacity.

Pressurizer Inter-Region Heat Transfer Coefficient

A conservatively low pressurizer inter-region heat transfer coefficient is assumed. This will result in a larger pressurization of the steam bubble thus maximizing the primary system pressure response

Main Steam Safety Valves

In order to maximize the primary system pressure response the main steam safety valves (MSSVs) are modeled with conservative assumptions for drift, blowdown, and relief capacity to minimize steam relief. A higher secondary side pressure will reduce primary-to-secondary heat transfer, thereby maximizing the primary system pressure response.

Decay Heat

High post-trip decay heat maximizes the primary and secondary system pressure responses. Therefore, bounding high EOC decay heat values predicted by the 1979 ANS Standard 5.1 decay heat curve is assumed.

Single Failure

No credible single failure has been identified which adversely impacts the transient.

12.3.3 Physics Parameters

Moderator Temperature Coefficient

A least negative moderator temperature coefficient is assumed to minimize the negative reactivity feedback prior to reactor trip, thereby maximizing the integrated core power, and maximizing the system pressurization.

Doppler Temperature Coefficient

Prior to the reactor trip the fuel temperature increases slightly; therefore the least negative Doppler temperature coefficient is assumed to minimize the negative reactivity feedback, thereby maximizing the integrated core power and maximizing the system pressurization.

Beta-Effective and Neutron Lifetime

To minimize negative reactivity feedback prior to the reactor trip, it is conservative to use the largest βeff. Also, a large βeff will slow the post-trip neutron power decrease. Similarly, a maximum value for the prompt neutron lifetime is used, although this parameter has been found to have little impact on the results.

Normalized Scram Curve

A conservatively slow scram curve along with a minimum scram worth is used.

12.3.4 Control, Protection, and Safeguards Systems

Reactor Trip

The reactor trips on the high RCS pressure trip. A conservative trip delay time is assumed. The reactor trip on turbine trip above 28% full power is defeated. This is an additional conservative assumption that is not necessary but is assumed. Future analyses may elect to delete this assumption.

RCS Pressure Control

To maximize the primary system pressure response the pressurizer spray and the PORV are assumed inoperable, and the pressurizer heaters are assumed operable.

Pressurizer Level Control

Due to the short duration of the transient, modeling of makeup/letdown will not significantly affect the results of the analysis, and therefore is not modeled.

Main Feedwater System

Continued post-trip main feedwater flow tends to reduce the secondary system pressure and temperature which results in more primary-to-secondary heat transfer leading to a lower primary system pressure response. Therefore, main feedwater is isolated on turbine trip to maximize the primary system pressure.

Emergency Feedwater System

No credit is taken for emergency feedwater flow. The limiting conditions for the transient occur within seconds of the turbine trip before the Emergency Feedwater System can actuate.

Turbine Bypass System

The TBS is assumed to be inoperable to maximize the secondary system pressure and temperature. This will result in reduced primary-to-secondary heat transfer, thus maximizing the primary system pressure response.

12.3.5 Results

The peak primary system pressure case is analyzed for 40 seconds. The sequence of events is shown in Table 12.1. The turbine trip occurs at 0.1 seconds. Following the turbine trip, steam flow to the turbine stops abruptly due to the immediate closure of the turbine stop valves. The mismatch between power generated in the primary system and heat removed by the secondary system results in an increase in the secondary system temperature and pressure. As a result the primary system pressure and temperature increase. The neutron power (Figure 12-1) initially increases slightly due to the positive reactivity feedback associated with the increasing RCS pressure, which exceeds the negative reactivity feedback associated with the increasing RCS temperature. The RCS pressure (Figure 12-2) and the hot and cold leg temperatures (Figure 12-3) increase almost instantly due to the reduction in primary-to-secondary heat transfer. The pressurizer level (Figure 12-4) increases due to the expansion of the coolant.

The high RCS pressure trip setpoint is reached at 3.49 seconds after turbine trip, with the actual reactor trip occurring at 3.99 seconds including the RPS delay time. The RCS pressure continues to increase following the reactor trip and reaches a maximum actual pressure of 2503 psig at 7.5 seconds after the turbine trips. The PSVs do not lift. A few seconds after the reactor trips on high RCS pressure the RCS pressure begins to decrease. This results from decreasing reactor power and decreasing main steam pressure (Figure 12-5) resulting from the lifting of the main steam safety valves. The peak primary pressure occurs at the bottom of the reactor vessel (Figure 12-6). At 7.4 seconds after turbine trip, the pressure at the bottom of the reactor vessel reaches a maximum value of 2590.4 psig. This is within the acceptance criterion of 2750 psig.

12.3.6 RETRAN-3D Comparison

The RETRAN-02 and RETRAN-3D results comparisons are shown in Figures 12-1 to 12-6. The results are in good agreement with the exception of the primary system pressure. A difference of 13 psi (approximately 4%) between the RETRAN-02 and RETRAN-3D primary pressure predictions develops in the first 8 seconds of the transient. Pressure is very sensitive to rapid increases in pressurizer level, and in this case the deviation in the pressurizer level is less than 2 inches. This level deviation results from a small difference in primary-to-secondary heat transfer beginning at around 3 seconds when the main steam safety valves begin to lift. This is attributed

to small differences in the predicted critical flow through the main steam safety valves between RETRAN-02 and RETRAN-3D. The critical flow at a junction is a function of the pressure and enthalpy of the fluid. These properties are calculated in slightly different ways in RETRAN-02 and RETRAN-3D. Both calculation methods are appropriate approximations based on known local fluid properties. These small differences are insignificant considering the much greater effect of the numerous conservatisms included in the turbine trip analysis, and the large margin to the acceptance criterion.

12.4 Reload Cycle-Specific Evaluation

The key parameters that are checked for each reload core are:

- Moderator temperature coefficient
- Doppler temperature coefficient
- Minimum scram worth curve

12.5 References

12-1 RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI-NP-1850-CCM, Revision 4, EPRI, November 1988

12-2 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000, Duke Power Company, July 1987

12-3 RETRAN-3D - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-7450, Revision 3, EPRI, September 1998

Table 12-1

Sequence of Events

Time (sec)	Event Description
0.1	Turbine trip
	MFW assumed lost
3.06 - 4.20	MSSVs lift
4.09	Reactor trips on high RCS pressure
7.50	Peak RCS pressure of 2590.4 psig occurs at bottom of reactor vessel
17.10	MSSV begin to reseat
40.00	End of simulation

Figure 12-1
TURBINE TRIP

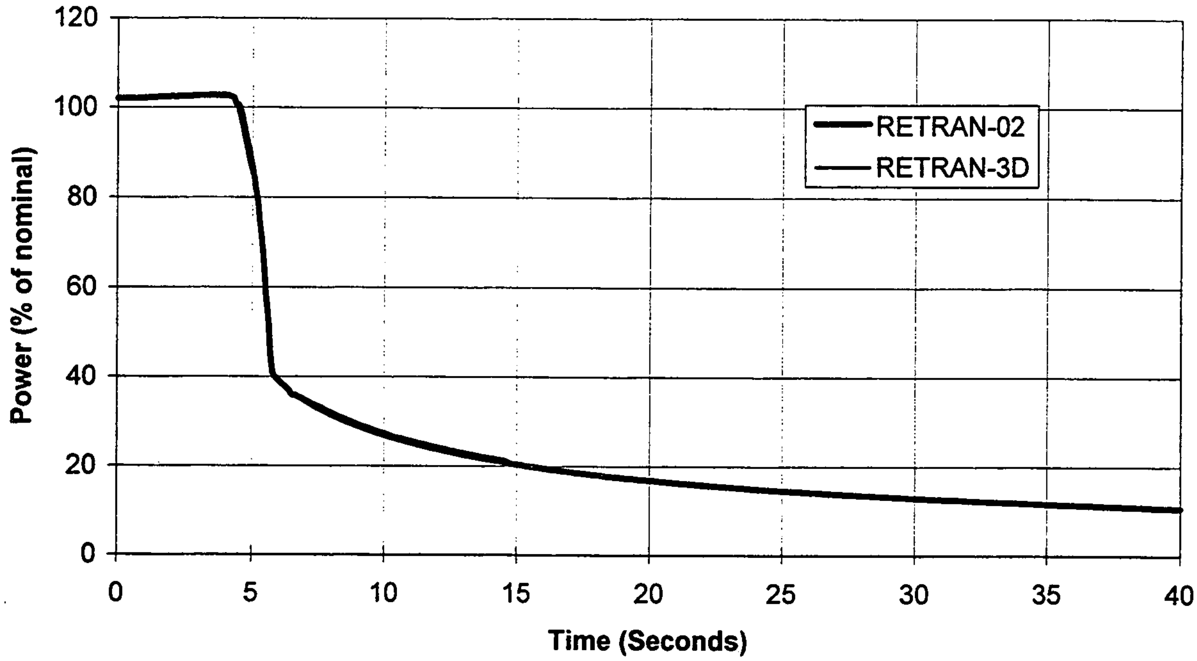


Figure 12-2
TURBINE TRIP

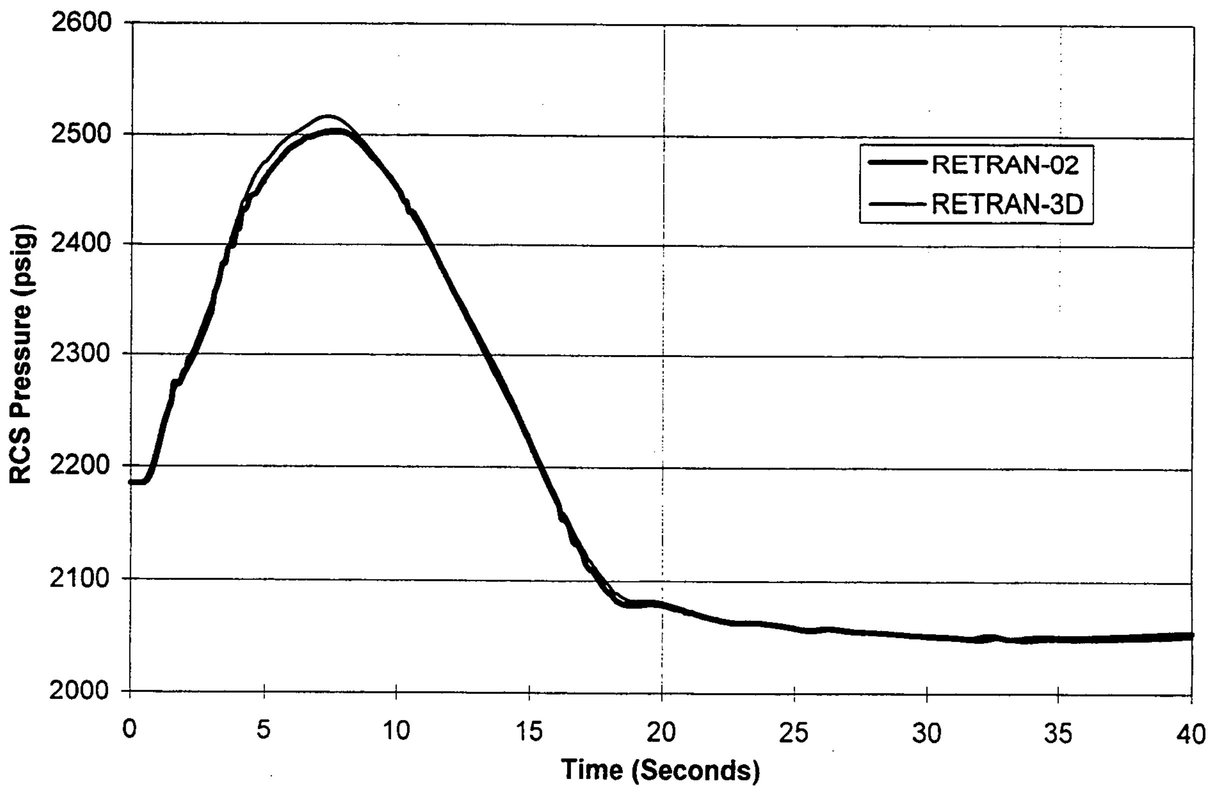


Figure 12-3
TURBINE TRIP

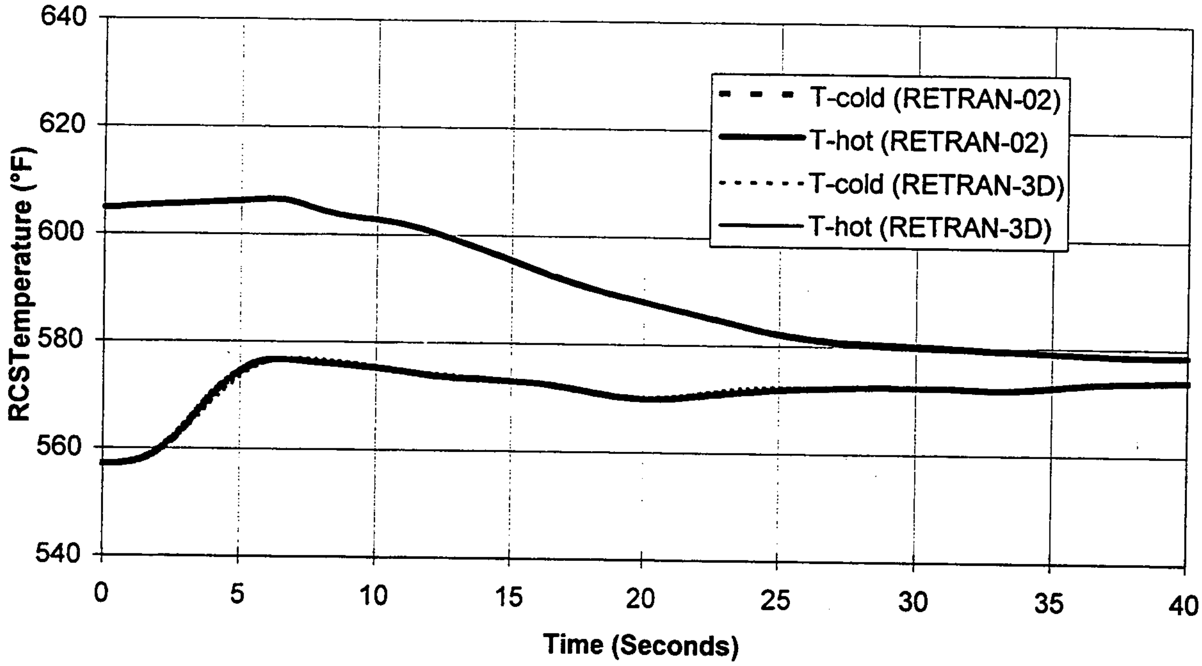


Figure 12-4
TURBINE TRIP

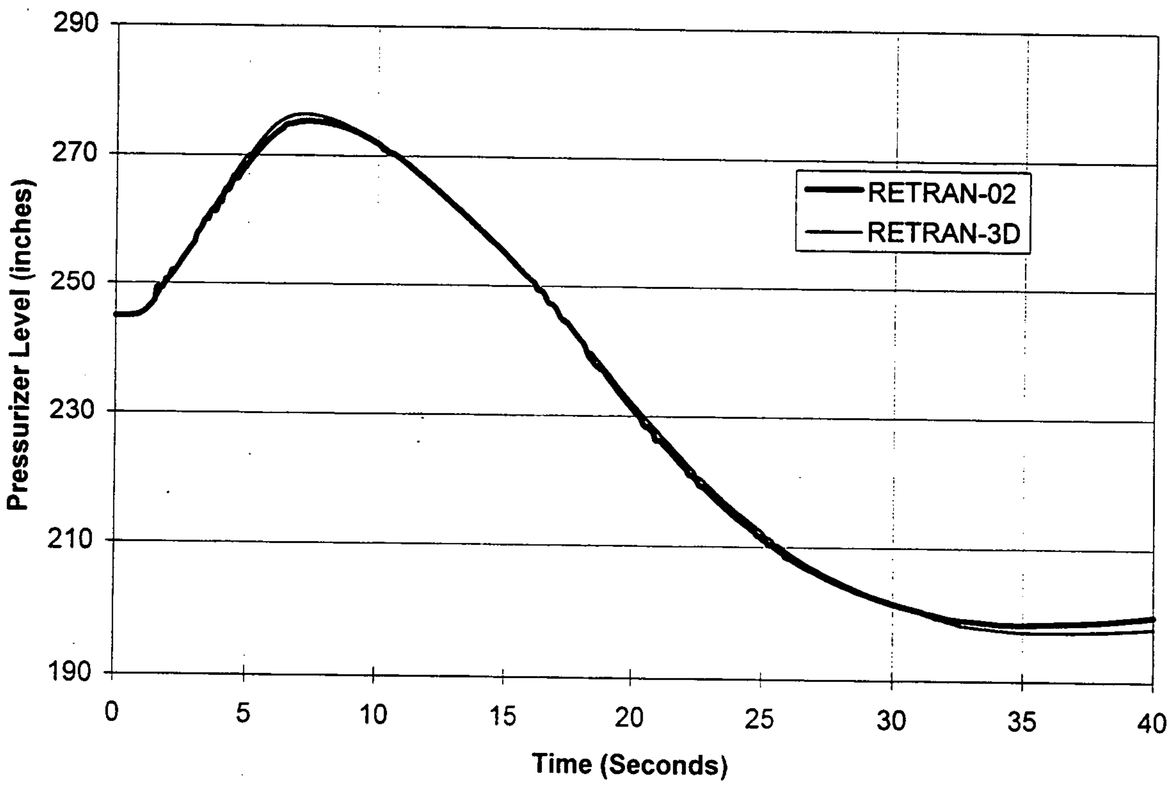


Figure 12-5
TURBINE TRIP

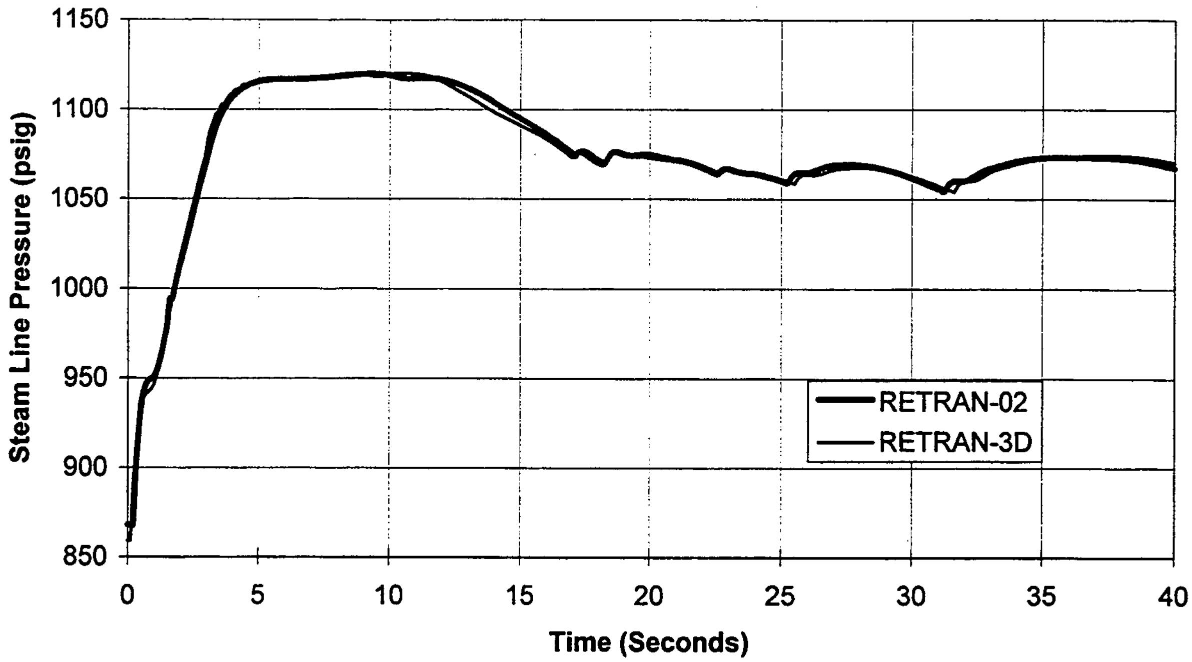
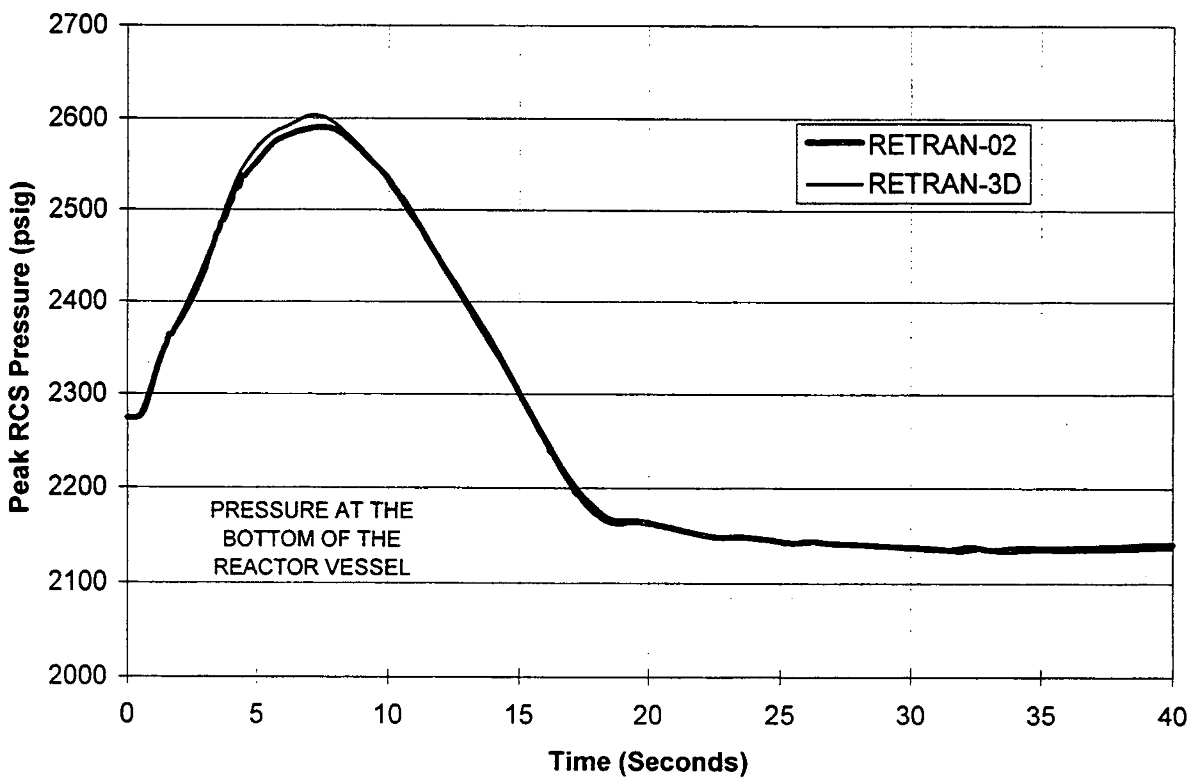


Figure 12-6
TURBINE TRIP



13.0 STEAM GENERATOR TUBE RUPTURE

The steam generator tube rupture (SGTR) event analyzed is a double-ended rupture of a single tube. This transient is analyzed to provide input data for a separate analysis of the fission product release to the environment. The key results are the primary-to-secondary leak rate through the ruptured tube and the secondary steaming rates.

The acceptance criterion for this event is that the offsite radiological doses will be less than 100% of the limits established in 10CFR100.

13.1 Nodalization

This asymmetric transient is analyzed using the two-loop Oconee RETRAN model (Reference 13-1). A junction is added to the base model to connect the steam lines together, which better represents the actual plant design prior to turbine trip. This base model addition is necessary since an asymmetric steam generator response will occur prior to turbine trip during this event.

[

]

13.2 Initial Conditions

Core Power Level

High initial core power with a positive uncertainty is assumed to maximize the primary system heat load.

RCS Pressure

Nominal pressure with a positive uncertainty maximizes the pressure difference across the ruptured tube, thus maximizing break flow and the offsite dose release.

Pressurizer Level

Nominal level with a positive uncertainty maximizes the post-trip energy content of the RCS that must be removed during the cooldown.

RCS Temperature

Nominal RCS average temperature with negative uncertainty is assumed to minimize the primary fluid temperature at the break location, thus maximizing the mass flow rate through the ruptured tube.

RCS Flow

Nominal RCS flow with a negative uncertainty is assumed to minimize frictional and form losses between the reactor coolant pump discharge and the break location. For a given initial RCS pressure, this will maximize the pressure at the break location and thus break flow.

Core Bypass Flow

Core bypass flow is not an important parameter for this transient.

Fuel Temperature

High initial fuel temperature is assumed to maximize the stored energy in the RCS which must be removed during the post-trip cooldown.

Steam Generator Mass

Low initial steam generator mass reduces the initial secondary inventory available to mix with and dilute the primary-to-secondary leakage, which is conservative for the offsite dose calculation.

Steam Generator Tube Plugging

A high steam generator tube plugging level is assumed to minimize the initial steam generator steam pressure and maximize the pressure differential across the ruptured tube. High tube plugging can also reduce the post-trip cool down rate.

13.3 Boundary Conditions

Break Model

The break is assumed to be a double-ended rupture of a single steam generator tube at the bottom tube sheet surface. This location maximizes the mass flow through the break.

RCP Operation

Reactor coolant pumps (RCPs) are assumed to be available and operate manually during this event. It is assumed that operators have the ability to trip RCPs during the unit cooldown to assist in reducing the RCS heat load.

Ambient Heat Loss From the RCS

A non-conducting heat exchanger is used to model ambient heat loss from the RCS following reactor trip.

Pressurizer Inter-Region Heat Transfer Coefficient

No credit is taken for liquid-steam interface heat transfer in the pressurizer until pressurizer level recovers to its post-trip setpoint. During pressurizer insurges, this will maximize RCS pressurization and hence primary-to-secondary leakage.

Atmospheric Dump Valves

The atmospheric dump valves are modeled to minimize their steaming capacity, which will slow the post-trip cooldown.

Main Steam Safety Valves

The main steam safety valves are modeled with conservative lift, blowdown, and relief capacity assumptions which minimize secondary pressure and maximize atmospheric steam releases.

Decay Heat

High decay heat is assumed to mixímize the post-trip steaming rates. Bounding high EOC decay heat values predicted by the 1979 ANS Standard 5.1 decay heat curve are assumed.

Ruptured Steam Generator Level Control

After operators isolate the ruptured steam generator, break flow will gradually fill the ruptured steam generator. Upon reaching a high level setpoint which is below the elevation at which water will spill into the main steam lines, operators will control the level by steaming to the atmosphere and/or draining.

Single Failure

The single failure identified for maximizing offsite dose is the failure of the emergency feedwater (EFW) control valve on the intact steam generator to open following reactor trip. This results in the ruptured steam generator providing all post-trip heat removal until operator action corrects the problem.

Manual Operator Actions

- Immediate action to initiate flow from the High Pressure Injection (HPI) System.

- Identify the failed-closed position of the EFW control valve and restore EFW to the intact steam generator. An operator action delay time of 23 minutes after reactor trip is assumed.

- Identification of the ruptured steam generator is determined by the EFW flow imbalance between the intact steam generator and the ruptured steam generator. An operator action delay time of 10 minutes after flow has been restored to the intact steam generator is assumed.

- An operator action delay time is assumed from the time the ruptured steam generator is identified to the time a cooldown of the RCS to 532°F begins. Operator action delay times of 5 minutes for control room actions, and 40 minutes for local action are assumed.

- Isolate the ruptured steam generator after reaching 532°F. This includes isolation of steam loads and isolation of EFW to this steam generator. An operator action delay time of 10 minutes after reaching 532°F is assumed.

- RCS subcooled margin is minimized after identification of the tube rupture. This is accomplished by using pressurizer spray to depressurize the RCS. An operator action delay time of 12 minutes following identification of the tube rupture is assumed.

- One RCP per loop is tripped off after the RCS has cooled down to 532°F. A 10 minute operator action delay time is assumed for this after reaching 532°F.

- A shift changeover delay of one hour is assumed. The RCS is held at a stable condition during this time.

- Cooldown of the RCS to 450°F occurs after shift changeover is completed. An operator action delay time of 5 minutes is assumed.

- Cooldown of the RCS is halted upon reaching 450°F . The RCS boron concentration is determined at this time. An operator action delay time of 90 minutes is assumed.

- Boration of the RCS is performed to reach the cold shutdown boron concentration requirement. An operator action delay time of 30 minutes is assumed.

- Cooldown to decay heat removal conditions resumes after the cold shutdown boron concentration has been achieved. An operator action delay time of 5 minutes is assumed.

- During the cooldown the ruptured steam generator is periodically steamed to the atmosphere and/or drained to prevent water from entering the steam lines. A conservatively low steam generator level setpoint corresponding to an elevation below that of the steam line is assumed for this action.

- A delay time is assumed to align the decay heat removal system. An operator action delay time of 45 minutes is assumed.

13.4 Physics Parameters

Credit for pre-trip negative reactivity addition from HPI System boron injection using the generalized transport model and a conservatively low value of differential boron worth is credited in the methodology. The analysis presented in this report does not include the pre-trip effect of boron injection.

13.5 Control, Protection, and Safeguards Systems

Reactor Control

Reactor power does not change prior to reactor trip, thus the reactor control subsystem of the Integrated Control System (ICS) is assumed to be in manual.

Reactor Trip

The Reactor Protective System is assumed to trip the reactor 20 minutes after the tube rupture occurs.

RCS Pressure Control

Pressurizer heaters are assumed to be operable during this event. Pressurizer spray is also assumed to be operable since it is the limiting means of minimizing the RCS subcooled margin after identification of the tube rupture.

Pressurizer Level Control

ECCS injection is manually throttled after the tube rupture occurs to control pressurizer level to a high setpoint.

ECCS Injection

Maximum ECCS injection flow is assumed to be available coincident with the tube rupture. This maximizes primary-to-secondary leakage and lengthens the time to reactor trip. [

]

Main Feedwater System

The main feedwater subsystem of the ICS is assumed to be in automatic control prior to reactor trip. This will throttle main feedwater to the ruptured steam generator due to the break flow entering it. Main feedwater flow is assumed to be terminated after reactor trip. This minimizes the secondary inventory available to mix with and dilute primary-to-secondary leakage. This also reduces the ability to cool the unit down.

Turbine Control

The turbine control subsystem of the ICS is assumed to be in automatic to prevent steam generator pressure from increasing before the reactor is tripped. This will maximize primary-to-secondary leakage.

Emergency Feedwater System

EFW initiation occurs on the loss of MFW with a long delay. A single failure of the EFW control valve on the intact steam generator to open results in the ruptured steam generator providing all post-trip heat removal until operator action corrects the problem. Minimum flow rates are assumed to minimize primary-to-secondary heat transfer.

13.6 Results

The thermal-hydraulic response resulting from this event is provided as input to a separate analysis which determines the fission product release to the environment.

13.7 Reload Cycle-Specific Evaluation

The reload physics parameter that must be checked is a minimum boron worth.

13.8 References

13-1 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000, Duke Power Company, July 1987

13.8 References

13-1 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000, Duke Power Company, July 1987

14.0 ROD EJECTION

14.1 Overview

14.1.1 Description

The rod ejection accident is initiated by a failure of a control rod drive mechanism housing, which allows a control rod to be rapidly ejected from the reactor by the Reactor Coolant System pressure. If the reactivity worth of the ejected control rod is large enough, the reactor will become prompt critical. The resulting power excursion will be limited by the fuel temperature feedback and the accident will be terminated when the Reactor Protective System trips the reactor on high neutron flux and the remaining control rods fall into the core. If a control rod ejection should occur, the nuclear design of the reactor core and limits on control rod insertion will limit any potential fuel damage to acceptable levels.

14.1.2 Acceptance Criteria

The acceptance criteria for the rod ejection accident analysis are:

- The accident will not further damage the RCS and,
- The offsite dose will be less than 100% of the 10CFR100 limits.

The first criterion of no further damage to the RCS is interpreted to mean that the peak primary pressure and the peak pellet radial average enthalpy both remain below a specified limit. The peak primary pressure limit is to remain within Service Limit C as defined by the ASME Code (Reference 14-1), which is 120% of the 2500 psig design pressure, or 3000 psig. The peak enthalpy limit is such that the radially averaged fuel pellet enthalpy shall not exceed 280 cal/gm at any location. To evaluate the second criterion of offsite dose being with the 10CFR100 limits, the extent of fuel failures are quantified with the assumption that any fuel pin that exceeds the DNB limit is considered failed. The fuel failure results are used in the offsite dose calculations to verify that the offsite dose criterion is satisfied.

14.1.3 Analytical Approach

The complexity of the core and system response to a rod ejection event requires the application of a sequence of computer codes. The rapid core power excursion is simulated with a three-dimensional transient neutronic and thermal-hydraulic model using the ARROTTA code (Reference 14-2) or SIMULATE-3K (Reference 14-3). A pin-to-assembly factor from SIMULATE-3P (Reference 14-4) is used to expand ARROTTA assembly peaks to pin peaks. The resulting transient core power distribution results are then input to VIPRE-01 (Reference 14-5) core thermal-hydraulic models. The VIPRE models calculate the fuel temperatures, the allowable power peaking to avoid exceeding the DNBR limit, and the core coolant expansion rate. The allowable power peaking is then used along with a post-ejected condition fuel pin census to determine the percentage of pins exceeding the DNB limit. The coolant expansion rate is input to a RETRAN-02 (Reference 14-6) model of the Reactor Coolant System to determine the peak pressure resulting from the core power excursion.

14.2 Simulation Codes and Models

14.2.1 ARROTTA

The ARROTTA code described in Section 2.6 is used to calculate core power response and three-dimensional power distribution. Modifications are made to the model to ensure conservative results. Each of these are discussed in Section 14.3.

14.2.2 SIMULATE-3K

The SIMULATE-3K code described in Section 2.7 is also used to calculate core power response and three-dimensional power distribution. SIMULATE-3K includes a prediction of individual pin powers. Modifications are made to the model to ensure conservative results. These changes produce a rod ejection model which is functionally equivalent to the ARROTTA model.

14.2.3 SIMULATE-3P

The SIMULATE-3P code described in Section 2.5 is used to calculate a pin-to-assembly factor to apply to the ARROTTA averaged assembly results.

14.2.4 VIPRE-01

The VIPRE-01 code described in Section 2.3 is used in the calculations of peak fuel enthalpy, DNBR, and the coolant expansion rates for various initial and boundary conditions postulated for this transient. Any changes to the Oconee VIPRE model (Reference 14-9) are described in the following subsections.

14.2.4.1 Peak Pellet Enthalpy

To show that the peak fuel enthalpy acceptance criterion is met, a [ ] channel VIPRE model with fuel conduction is used to calculate the maximum hot spot fuel temperature during the transient. Given the ARROTTA predicted [ ], VIPRE calculates the transient maximum hot spot average fuel temperature. This fuel temperature is then used in the calculation of the maximum radial average fuel enthalpy. Details regarding the [ ] channel VIPRE model and initial and boundary conditions follow.

Model Description

A [ ] channel model with fuel conduction is constructed to simulate the peak fuel pin in the hot assembly during the transient. Sensitivity studies have shown that a [ ] channel model and the [ ] channel model yield nearly identical fuel temperature results. The single channel model consists of a [ ].

Power Distributions

During the transient, the hot assembly axial power distributions change mainly due to the ejected control rod and to the insertion of control rods as the reactor trips. [ ] For hot-

zero-power (HZP) conditions, the [

]

Fuel Conduction Model

The VIPRE fuel conduction model is used. A [ ] results in conservative fuel temperatures and is used in the fuel temperature calculations. During the transient, core average power will increase as will the hot fuel assembly power. Above a certain power level, [

]

Heat Transfer Correlations

Heat transfer correlations used for the four major segments of the boiling curve are as shown below.

Single-phase forced convection: [
Saturated nucleate boiling regime:
Transition boiling regime:
Film boiling regime:]

The critical heat flux correlation used to define the peak of the boiling curve is the BWC (Reference 14-7) correlation (the same correlation will be used that is used to predict the DNBR). The minimum DNBR value for the peak of the curve is set to 1.24, which is the correlation limit of 1.18 plus 5% margin.

Flow Correlations

For the rod ejection analysis, the subcooled void, the bulk void, and the two-phase friction multiplier are modeled by using the [ ] correlations, respectively.

Conservative Factors

An engineering hot channel factor (Fq) of [ ] is applied to the hot rod. This factor accounts for the increase in rod power due to manufacturing tolerances such as differences in the number of U-235 grams per rod, loading tolerances of U-235 per stack, and variations on the powder lot mean enrichment.

The hot subchannel flow area is reduced to account for variations in the as-built dimensions of the subchannel. The following flow area reductions are assumed.

Unit channel: [ ] flow area reduction
Thimble channel: [ ] flow area reduction

A core inlet flow maldistribution penalty is assumed depending upon the pump configuration. This inlet flow reduction is applied to the hot assembly and the [ ] channel model and is as follows:

4 reactor coolant pumps: 5% inlet flow reduction
3 reactor coolant pumps: [ ] inlet flow reduction
2 reactor coolant pumps: [ ] inlet flow reduction

[ ]

Direct Moderator Heating

The amount of heat generated in the coolant is 2.7% of the total power.

Fuel Enthalpy Calculation

VIPRE-01 does not perform fuel enthalpy calculations. Thus, the fuel enthalpy for a given fuel temperature during the transient is calculated separately from VIPRE based on the equation

obtained from MATPRO (Reference 14-8).

$$\text{FENTHL} = \text{FENTHL (T)} - \text{FENTHL (Tref)}$$

with

$$\text{FENTHL} = \frac{K_1\Theta}{\exp(\Theta/T)-1} + \frac{K_2T^2}{2} + \frac{YK_3}{2}\exp(-E_D/RT)$$

where

FENTHL = fuel enthalpy (J/kg)

T = temperature (°K)

Y = oxygen to metal ratio = 2.0

R = 8.3143 (J/mol-°K)

Θ = Einstein Temperature (°K)

= 535.285 for UO_2

K_1 = 296.7 (J/kg-°K)

K_2 = 0.0243 (J/kg-°K^2)

K_3 = 8.745 E+7 (J/kg)

E_D = 1.577 E+5 (J/mol)

FENTHL (Tref) = fuel enthalpy at any desired reference temperature

The above fuel enthalpy correlation is only valid for a fuel temperature greater than about 300 °K (80.3 °F). Thus, the reference temperature is 300 °K. To convert J/kg to cal/g, FENTHL is multiplied by 0.23889/1000.0.

14.2.4.2 DNBR Evaluation

Utilizing the ARROTTA or SIMULATE-3K transient [ ] the Oconee [ ] channel VIPRE model (Reference 14-9) or the [ ] -channel model described in Section 14.2.4.1 are used to determine acceptable power peaking limits. For a given axial power profile, the maximum pin radial peak can be determined such that DNB would not occur during the transient. These DNB limits are referred to as maximum allowable radial peaks (MARP) limits. The CHF correlation used for the results

presented is the BWC correlation and the DNBR limit is 1.24, which is the correlation design limit of 1.18 plus 5 % margin. A fuel pin census is then performed to determine the number of fuel pins in the core that exceed the power peaking limit.

The number of pins exceeding the DNB limit during the transient is determined by first combining the ARROTTA[

]. The radial peak for each pin is then compared to the radial peak DNB limit based on the axial peak and location of the peak. The total number of pins exceeding the DNB peaking limit is then calculated for use in the offsite dose analysis.

Due to the severity of the power excursions observed for this transient, a non-statistical core design (SCD) analysis is performed for the DNB calculations. Thus, various conservative factors included in the SCD must be included in this analysis, as discussed below.

Model Description

The model used for the DNBR evaluation is the previously approved[ ]channel model or the [ ]channel model.

Power Distributions

A spectrum of axial power distributions is analyzed for each rod ejection case. The magnitude of the axial peaks are varied to bound the range of peaks observed from the ARROTTA or SIMULATE-3K analyses. For each axial peak value and location described above, the pin radial peak is varied until the DNB limit is obtained. This pin radial peak is held constant throughout the transient.

Fuel Conduction Model

A[ ]is assumed to generate conservative DNB results. Since the MARP limits are applied to every pin in the core but generated only for the hot assembly, two different assumptions are made[ ]resulting in two different sets of MARP limits for each rod ejection case. The first set is for[

]

[ ]

Heat Transfer Correlations

For the DNBR calculations, only the single-phase forced convection and nucleate boiling heat transfer modes are applicable. The [ ] is used for the single-phase forced convection mode. The Thom plus [ ] is used for the nucleate boiling regime. The critical heat flux correlation used to define the peak of the boiling curve is the BWC correlation. The minimum DNBR value for which transition boiling occurs is set to 1.24.

Flow Correlations

The [

] for the two-phase friction multiplier.

Other Thermal-Hydraulic Correlations

In addition to those correlations discussed in Section 14.2.4.1, turbulent mixing is also calculated by VIPRE for the flow and energy solutions in the DNBR calculations. The single phase mixing correlation used is shown below and is used for both single and two-phase mixing.

$$w' = A S \overline{G}$$

where: A =

S = gap width, feet

$\overline{G}$ = average mass velocity in the channels connected by gap K, lbm/sec-ft^2

Conservative Factors

The conservative factors used in the DNBR calculations are identical to those used in Section 14.2.4.1.

14.2.4.3 Coolant Expansion Rate

If the peak fuel enthalpy criterion is met, there is no fuel dispersal into the coolant. Therefore, the Reactor Coolant System expansion rate may be calculated using conventional heat transfer from the fuel and prompt heat generation in the coolant. This rate must be calculated with the consideration of the spatial power distribution before and during the transient since this rate, at any location in the reactor core, depends on the initial amount of subcooling and the heat transfer rate. A[ ]channel VIPRE model (refer to Section 2.3.2) is constructed for this purpose. Using the ARROTTA or SIMULATE-3K[

]the Oconee RETRAN plant transient model for simulating the resulting pressure response.

Model Description

A[ ]channel model with fuel conduction is constructed to calculate the reactor coolant expansion rate during the transient. ARROTTA results show that the[

.]

Axial Power Distributions

ARROTTA or SIMULATE-3K generate assembly axial power distributions for each assembly during the transient.[

.]

Radial Power Distributions

The transient assembly radial power distributions generated by ARROTTA or SIMULATE-3K are used for the analysis. [

]

Fuel Conduction Model

A [ ] is used to provide for a conservative calculation of the channel local fluid conditions. [

]

Heat Transfer Correlations

For conservatism, only the first two segments of the boiling curve are used in the coolant expansion rate calculation. This forces VIPRE to use the [ ] for post-DNB conditions. This results in a conservatively large heat flux post-DNB and thus a conservatively high coolant expansion rate.

Single-phase forced convection: [

Saturated nucleate boiling regime:]

The critical heat flux correlation used to define the peak of the boiling curve is the BWC correlation for the results presented (the same correlation used to predict the DNBR). The minimum DNBR value for the peak of the curve is set to 1.24, which is the correlation limit of 1.18 plus 5% margin.

Flow Correlations

For the coolant expansion rate calculations, the subcooled void, the bulk void, and the two-phase friction multiplier are modeled by using the [

].

Other Thermal-Hydraulic Correlations

[ ].

Conservative Factors

Only the core inlet flow maldistribution penalties discussed in Section 14.2.4.1 are assumed in the coolant expansion rate calculation. [

].

Calculation of the Reactor Coolant Expansion Rate

[

]

14.2.5 System Thermal-Hydraulic Analysis

The RCS response to a rod ejection accident is primarily a rapid pressurization due to the increase in heat transfer associated with the power excursion. The [

] boundary condition. The RETRAN-02 model is the single-loop base model described in detail in Reference 14-9.

14.3 Nuclear Analysis

The response of the reactor core to the rapid reactivity insertion from a control rod ejection is simulated with two independent models, ARROTTA and SIMULATE-3K. Model geometry is typically one node per fuel assembly in the radial direction. The radial and axial nodalization depends on the fuel assembly design, such as whether or not axial blanket fuel is being modeled. For the analysis presented, a typical axial nodalization of 23 equal length fuel nodes in the axial direction is used. Beyond the fuel nodes, both models explicitly calculate neutron leakage from the core by use of reflector nodes in the radial direction beyond the fuel region and in the axial direction above and below the fuel column stack.

The SIMULATE-3K three dimensional neutron kinetics model used in this analysis is functionally equivalent to the ARROTTA model. Thus the analyses documented in this report provide a benchmark of the SIMULATE-3K model relative to ARROTTA, with ARROTTA having been previously reviewed and approved by the NRC for rod ejection analysis.

ARROTTA and SIMULATE-3K are used to calculate the core power level and nodal power distribution versus time during the rod ejection transient. SIMULATE-3K includes pin power reconstruction capability [

] This information is used by VIPRE to determine the fuel enthalpy, the percentage of the fuel pins exceeding the DNB limit, and the coolant expansion rate.

14.3.1 ARROTTA Analysis

The control rod ejection transient is analyzed for the following six initial conditions.

- BOC at maximum allowable core power level with 4 RCPs operating
- BOC at maximum allowable core power level with 3 RCPs operating
- BOC at HZP (two RCPs in the analysis presented, three in the future)

- EOC at maximum allowable core power level with 4 RCPs operating
- EOC at maximum allowable core power level with 3 RCPs operating
- EOC at HZP (two RCPs in the analysis presented, three in the future)

Because of the modifications to the ARROTTA model which are described below, the analyses performed are expected to bound all future Oconee reload core designs.

Ejected Rod Location and Velocity

The core location of the ejected control rod is chosen specifically for each analysis to produce the most conservative results. Figure 14-1 shows the core configuration and location of the ejected control rod for the various analyses. For both HZP transients the control rod in core location L-10 is ejected from a fully inserted position in 0.15 seconds at constant velocity. The time required for rod ejection is consistent with the original FSAR analysis. It assumes that the pressure barrier has failed in such a way that it no longer offers any restriction to the ejection and that there is no viscous drag force limiting the rate of ejection. The analyses performed with 3 RCPs operating eject the control rod from core location H-12. The initial position of the ejected rod, including uncertainty, is 38% withdrawn which corresponds to the maximum insertion limit allowed for this condition. The time required to eject the rod is determined by proportionally reducing the 0.15 seconds required for full length ejection by the fraction of control rod inserted. Thus the conservative assumptions previously stated are preserved. For the transients performed with 4 RCPs operating, the control rod in location H-12 is ejected from an initial position of 58% withdrawn. As before this corresponds to the maximum insertion limit allowed for this condition and dictates a proportional reduction in ejection time.

Ejected Rod Worth

The worth of the ejected rod is the key parameter which drives the core power excursion in these transients. Adjustments are made to the worth of the ejected control rod to ensure a conservative analysis that is expected to bound future reload core designs. The ejected control rod worth is [ ] for fuel compositions in the appropriate core locations.

Initial Power Distribution

[ ]

Beta-Effective

The core response to this transient is sensitive to the effective delayed neutron fraction (β_{eff}). Since β_{eff} is dependent on enrichment and burnup, there will be some cycle-to-cycle variation caused by changing feed enrichments and discharge batch size. ARROTTA input includes β_{eff} for six delayed groups for each unique fuel composition in the core. [ ]

Moderator Temperature Coefficient

The initial condition moderator temperature coefficient [ ]. While this coefficient has little effect on the peak power response of this transient, this conservative adjustment is made to

ensure consistency with other postulated transients which are limited by the moderator temperature coefficient.

Doppler Temperature Coefficient

The Doppler temperature coefficient is important to this transient because the negative reactivity feedback from increased fuel temperature is the initial effect that limits the power excursion and begins to shut down the reactor. [

]

Reactor Trip and Single Failure

The reactor trip signal is generated for all six transients when three of four excore detectors exceed the high flux trip setpoint used during cycle operation. This conservative modeling assumes that the detector which would indicate the highest flux level has failed and that a two out of the remaining three logic is required to generate a trip signal. The excore signals are synthesized from a conservative combination of power densities of several assemblies in the proximity of each excore detector. A conservative trip delay time is assumed.

Scram Curve and Worth

The reactor scram assumes that the ejected rod, part length axial power shaping rods, and the remaining rod with the highest worth do not fall into the core. The remaining scrammed rod worth is reduced to ensure that only the minimum net shutdown margin is achieved. Additional conservatism is applied by limiting the rate at which the scrammed rods are allowed to fall into the core such that the reactivity inserted is bounded by the limiting curves.

The total effect of all these conservatisms is to create an ARROTTA rod ejection model that will provide results which bound all future reload cycles. Table 14-1 identifies the conservative core parameters used in each analysis.

14.3.2 SIMULATE-3K Analysis

The SIMULATE-3K analyses utilize the same methodology as previously described for ARROTTA. The conservatisms required by the rod ejection analysis and described above for the ARROTTA analysis are implemented using SIMULATE-3K user features.

14.3.3 Results

ARROTTA results from each of the six cases are summarized in Table 14-2. Results from the SIMULATE-3K cases are provided in Table 14-3. Core power versus time for each model is shown in Figures 14-2 through 14-7.

For the 4 and 3 RCP transients, which begin at 102% and 82% power, respectively, core power increases rapidly as the control rod is ejected. The ejected rod worth in these transients is not sufficient to achieve a prompt critical state. Power increases until Doppler feedback from increasing fuel temperature begins to turn the excursion around. Core power level continues to decrease as the fuel temperature approaches an equilibrium value. A reactor trip signal on high flux occurs very early in these transients but the conservative trip delay time prevents rod motion until after the peak core power occurs. Additional conservatisms applied to the rate of rod insertion and scram worth minimizes the effect of the reactor trip until the rods approach the bottom of the reactor core.

The transients initiated from HZP differ from the at power initial conditions in that the ejected rod worth is large enough to achieve a prompt critical core. Power increase continues long after the control rod is fully ejected until the fuel heats up enough for Doppler feedback to turn the excursion around. Conservatisms on trip delay time, rate of rod insertion, and scram worth minimize the impact of the reactor trip.

Core power level and [ ] are key inputs to the thermal hydraulic analyses discussed below. Figures 14-8 and 14-9 provide full core maps of the ARROTTA assembly averaged power distribution at the time of peak core power during the two most limiting rod ejection transients. This data is input to the VIPRE analysis described below to determine the number of pin failures due to DNB.

14.3.4 Fuel Pin Census for DNBR Evaluation

DNBR calculations are performed with VIPRE for each of the six rod ejection cases. The results are expressed as a family of curves of maximum allowed radial peak (MARP) versus assembly axial peak and location. When the radial pin power of a fuel pin exceeds the appropriate MARP during a transient, DNB is assumed to occur and that pin is assumed to fail. A utility code compares the pin power distributions generated by the core neutronics calculations to the MARP limits generated during the VIPRE analysis and accumulates the number of failed fuel pins by assembly and total core. Table 14-4 summarizes the results of the fuel pin census for each case modeled. Figures 14-10 and 14-11 provide core maps showing distribution of pin failures for the two most limiting cases analyzed. It is noted that HZP cases do not result in as many fuel failures as the at power cases, and therefore the fuel pin census need not be checked in future reload cores unless any of the other key parameters fail the reload check.

14.4 Thermal-Hydraulic Analysis

14.4.1 VIPRE-01 Analysis and Results

The four key initial conditions assumed for each VIPRE calculation are the core inlet flow, core exit pressure, core inlet temperature and core average power. These conditions will vary depending on the power/pump combination. RETRAN-02 is used to generate these initial conditions by initializing RETRAN to a conservative value for each parameter. This allows VIPRE and RETRAN to be coupled via the initial conditions.[

]The transient core average power forcing functions are shown in Figures 14-2 through 14-7.

During the rod ejection accident RCS pressure increases due to the coolant expansion as a result of the reactor power excursion.[

]

The mechanical failure that results in the ejected rod also causes a hole to open in the reactor vessel head. In order for the hole to permit the ejection of a control rod, it must be at least 1.5 inches in diameter. This hole will provide a pressure relief path for the RCS and is credited in the RETRAN peak pressure analysis.

14.4.1.1 Peak Fuel Enthalpy

The fuel temperatures and enthalpies are calculated for the various power/pump/burnup combinations. The maximum fuel temperature and radial averaged enthalpy during the transient are dependent on the initial radial pin power. The results tabulated below assume an initial pin power well above what could be expected in future cores.

Case	Maximum Centerline Fuel Temperature (F)	Maximum Fuel Average Fuel Temperature (F)	Maximum Clad Surface Temperature (F)	Maximum Fuel Average Enthalpy (cal/gm)
HFP, BOL	4776.4	3217.4	785.0	132.8
82 %FP, BOL	4645.3	3143.0	1077.3	129.0
HZP, BOL	1871.8	1495.9	707.8	55.1
HFP, EOL	4347.5	2739.5	772.2	109.7
82 %FP, EOL	4577.3	2921.9	785.2	118.3
HZP, EOL	1954.3	1578.0	733.4	58.5

The above results show that during the transient, the maximum fuel average enthalpy is well below the acceptance criterion of 280 cal/gm. The HZP cases presented were analyzed with 2 RCPs when 3 RCPs could have been credited. Future analyses will credit 3 RCPs. It is noted that EOL and HZP conditions do not yield pellet enthalpies as high as the BOL HFP and 82 % FP cases.

14.4.1.2 DNBR Evaluation

The DNBR calculations are performed for the various power/pump/burnup combinations. The DNBR results are expressed as a family of curves of maximum allowed radial power (MARP)

versus assembly axial peak and location. Representative MARP curves are presented in Figures 14-12 through 14-14 for BOC conditions, with the HZP cases assuming 2 RCPs in operation when 3 RCPs could have been credited. EOC conditions yield similar, though higher, MARP curves which translates into fewer pin failures. When the radial power peak of the fuel pin exceeds the MARP during the transient, DNB is assumed to occur and the cladding fails. The fuel pin census results resulting from the use of these MARP curves are given in Table 14-4.

14.4.1.3 Coolant Expansion Rate

The BOC 3 RCP rod ejection transient results in the highest coolant expansion rate. Figure 14-15 shows the instantaneous core coolant expansion rate in ft^3/sec as a function of transient time. The initial expansion rate corresponds to the full power initial condition and the resulting decrease in coolant density due to sensible heating in the core. The result shows that a peak expansion rate of [ ] sec, the expansion rate has nearly decreased back to its initial value as power decreases due to reactor trip.

14.4.2 RETRAN Peak Primary Pressure Analysis

14.4.2.1 Initial Conditions

The RETRAN model pressure response to the rod ejection transient is primarily a function of the coolant expansion rate, which is input as a boundary condition. Most parameters such as initial primary temperature have little impact on the pressure response due to the [ ] which results in minimizing temperature transport effects. However, since the results of these initializations are used to generate the initial conditions for the VIPRE analyses, the transient is evaluated as if temperature transport effects do play an important role in the pressurization. Each initial condition is discussed below.

Only the maximum power levels at 4 and 3 RCP conditions are analyzed for the peak primary pressure.

Power Level

The pressure response is insensitive to the initial core power level. However, for input to the VIPRE analyses, a high core power level is assumed that includes uncertainty.

RCS Temperature

A high initial RCS average temperature will not significantly affect the results of this transient. However, a high initial RCS average temperature, including uncertainty, will maximize the core inlet temperature and therefore be conservative as an initial condition to the VIPRE analysis.

RCS Pressure

Sensitivity cases indicate that a low initial RCS pressure, including uncertainty, is conservative in that VIPRE will predict a higher coolant expansion rate and thus a greater peak primary pressure. For 3 RCP operation, one loop will be higher in pressure than the other loop. Since low RCS pressure is conservative, the highest loop pressure will be initialized to the target pressure.

Pressurizer Level

A high initial pressurizer level decreases the volume of the steam bubble thereby increasing the pressurization effect. Nominal pressurizer level plus uncertainty is assumed.

RCS Flow

A high RCS flow is conservative to maximize the loop ΔP. However, since the RETRAN and VIPRE analyses are coupled via the initial conditions and a low RCS flow will yield a greater coolant expansion rate, a low RCS flow, including uncertainties, is assumed.

Core Bypass Flow

To obtain a conservatively low core inlet flow for input to VIPRE, a high bypass flow is assumed.

SG Tube Plugging

High tube plugging will minimize the primary-to-secondary heat transfer and thus maximize the primary system pressurization. Furthermore, high tube plugging will minimize the RCS inventory. Therefore, a high tube plugging level is assumed.

SG Level

Secondary side conditions have little impact on the primary side pressurization due to the short duration of the transient.

Pressurizer Inter-Region Heat Transfer Coefficient

A conservatively low inter-region heat transfer coefficient is assumed to maximize the pressurization due to pressurizer insurges.

14.4.2.2 Boundary Conditions

Reactor Coolant Volume Expansion Rate

Reactor Power

Although reactor trip occurs within the first second of the rod ejection transient, reactor power is maintained at its initial value for the duration of the RETRAN simulation. This is necessary to maintain the initial (steady-state) coolant expansion rate. This technique ensures an accurate coolant volume expansion boundary condition.

Small Break LOCA

The mechanical failure that results in the ejected rod also causes a hole to develop in the reactor vessel head. Based on control rod drive drawings, the minimum hole size that will accept the control rod drive shaft is 1.5 inches in diameter. Even though the small break LOCA is present

at the start of the transient, it is not assumed to occur until 0.1 sec to allow time for the rod to completely eject from the vessel.

RCS Pressure Control

In order to conservatively bound the pressure response, the pressurizer PORVs and spray are defeated.

Pressurizer Safety Valves

The pressurizer code safety valves function as overpressure mitigation equipment. Conservative assumptions are made in regard to drift, blowdown, and relief capacity.

Main Feedwater System

Main feedwater is ramped to zero flow immediately following turbine trip to simulate Integrated Control System actions and steam generator pressurization effects on MFW flow.

Turbine Control

The turbine is assumed to trip immediately upon reactor trip.

Main Steam Safety Valves

The main steam safety valves are drifted high to minimize the secondary side pressure relief.

Single Failure

The single failure is described in Section 14.3.1.

14.4.2.3 Results

The Reactor Coolant System pressure response to the rod ejection is shown in Figure 14-16. The pressure plotted represents the pressure at the bottom of the reactor vessel where the highest system pressure occurs. Figure 14-16 shows that a peak system pressure of 2885 psig is reached in 2.3 seconds. The peak pressure is within the acceptance criterion of 3000 psig.

14.5 Reload Cycle-Specific Evaluation

The failed fuel pin census results from Section 14.3.4 are used in the determination of the offsite dose consequences. In addition to the calculated pin census results, a maximum allowable number of failed pins is determined in the offsite dose calculation. Each reload core design then verifies that this maximum allowable limit is not exceeded. A cycle- specific check will be made for those key physics parameters which most significantly determine the response of the rod ejection transient. These key physics parameters are listed in Table 14-5.

14.6 References

14-1 ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components", ASME

14-2 ARROTTA: Advanced Rapid Reactor Operational Transient Analysis, EPRI, August 1993

14-3 SIMULATE-3 Kinetics Theory and Model Description, SOA-96/26, Studsvik of America, April 1996

14-4 SIMULATE-3: Advanced Three-Dimensional Two-Group Reactor Analysis Code, STUDSVIK/SOA-92/01, Studsvik of America, April 1992

14-5 VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM, Revision 3, EPRI, August 1989

14-6 RETRAN-02: A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Revision 4, EPRI, November 1988

14-7 BWC Correlation of Critical Heat Flux, BAW-10143P-A, B&W, April 1985

14-8 Donald L. Hagrmann, et al., MATPRO-Version 11 (Revision 2): A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior, NUREG/CR-0479, August 1981

14-9 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000, Duke Power Company, July 1987

Table 14-1

Rod Ejection Accident Parameters

Parameter	BOC			EOC		
	4 RCP	3 RCP	HZP	4 RCP	3 RCP	HZP
Initial Core Power, %FP	102	82	1E-7	102	82	1E-7
Initial Core Avg TMOD, °F	581	581	540	581	581	540
Reactor Pressure, psia	2200	2200	2200	2200	2200	2200
Core Flow, gpm X 10^{+5}	3.71	2.73	1.73	3.71	2.73	1.73
Delayed Neutron Fraction	0.0058	0.0058	0.0058	0.0049	0.0049	0.0049
MTC, pcm / °F	-3.00	-2.20	+7.00 *	-25.0	-25.0	-15.0
DTC, pcm / °F	-1.25	-1.30	-1.65	-1.35	-1.38	-1.75
Ejected Rod Worth, pcm	200	400	800	200	400	800

* At ARO conditions

Table 14-2

ARROTTA Rod Ejection Results

Parameter	BOC			EOC		
	4 RCP	3 RCP	HZP	4 RCP	3 RCP	HZP
Initial Ejt Rod Position, %WD	58	38	0	58	38	0
Begin Rod Ejection, sec	0	0	0	0	0	0
End Rod Ejection, sec	0.063	0.093	0.150	0.063	0.093	0.150
Maximum Core Power, %FP	144	195	2098	148	223	1918
Time of Max Core Power, sec	0.076	0.106	0.270	0.079	0.112	0.262
Peak Assembly Power	2.38	3.01	4.28	2.45	3.12	4.51
Peak Nodal Power	3.33	4.37	6.66	3.44	4.77	9.99
Trip Signal Generation, sec	0.054	0.082	0.248	0.057	0.083	0.245
Begin Scram Rod Motion, sec	0.454	0.482	0.648	0.457	0.483	0.645
End Scram Rod Motion, sec	2.854	2.882	3.048	2.857	2.883	3.045

Table 14-3

SIMULATE-3K Rod Ejection Results

Parameter	BOC			EOC		
	4 RCP	3 RCP	HZP	4 RCP	3 RCP	HZP
Initial Ejt Rod Position, %WD	58	38	0	58	38	0
Begin Rod Ejection, sec	0	0	0	0	0	0
End Rod Ejection, sec	0.063	0.093	0.150	0.063	0.093	0.150
Maximum Core Power, %FP	140	194	1841	137	214	1752
Time of Max Core Power, sec	0.076	0.109	0.288	0.081	0.117	0.277
Peak Assembly Power	2.29	2.93	4.17	2.25	3.00	4.33
Peak Nodal Power	3.14	4.20	6.40	3.09	4.81	10.5
Trip Signal Generation, sec	0.054	0.082	0.248	0.057	0.083	0.245
Begin Scram Rod Motion, sec	0.454	0.482	0.648	0.457	0.483	0.645
End Scram Rod Motion, sec	2.854	2.882	3.048	2.857	2.883	3.045

Table 14-4

Total Pins Achieving DNB During Rod Ejection Accident (ARROTTA)

Transient	BOC	EOC
4 RCP	40.6%	27.6%
3 RCP	39.2%	36.3%
HZP	<1%	2.1%

Table 14-5

Rod Ejection Accident

Reload Cycle Key Parameter Checklist

Parameter		BOC			EOC		
		4 RCP	3 RCP	HZP	4 RCP	3 RCP	HZP
Max Ejected Rod Worth, $	≤	0.345	0.690	1.38	0.408	0.816	1.63
MTC, pcm / °F	≤	-3.00	-2.20	+7.00 *	-25.0	-25.0	-15.0
DTC, pcm / °F	≤	-1.25	-1.30	-1.65	-1.35	-1.38	-1.75
F Δh (at full power)	≤	1.80	1.80	1.80	1.80	1.80	1.80
Fq (at peak power)	≤	4.17	6.30	8.01	4.03	6.20	13.8
Fuel Failures, %	≤	***	***	N/A**	***	***	N/A**

* At ARO conditions

** The HZP cases are non-limiting in terms of fuel failure and do not need to be checked unless any of the other key parameters are violated

*** The percentage of fuel failures must be less than the percentage assumed in the dose analysis.

Figure 14-1

ROD EJECTION ACCIDENT

Oconee Core Configuration

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A															
B							F		F						
C				F		F		F		F		F			
D			F		F		F		F		F		F		
E				F		F				F		F			
F			F		F		F		F		F		F		
G		F		F		F				F		F		F	
H			F									3RCP 4RCP	F		
K		F		F		F				F		F		F	
L			F		F		F		F	HZP	F		F		
M				F		F				F		F			
N			F		F		F		F		F		F		
O				F		F		F		F		F			
P							F		F						
R															

F = Fresh fuel location

4RCP = Ejected rod location for transients with 4 RCPs

3RCP = Ejected rod location for transients with 3 RCPs

HZP = Ejected rod location for transients at HZP

Figure 14-2

ROD EJECTION ACCIDENT
BOC 4 RCP Core Power Versus Time

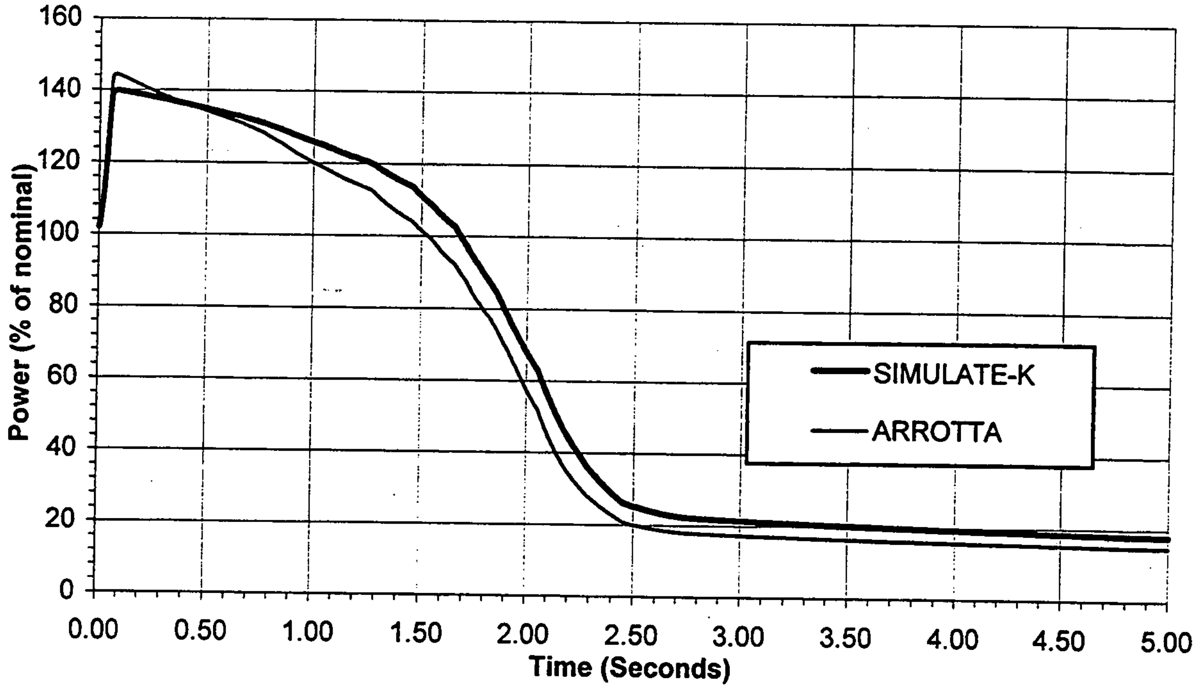


Figure 14-3

ROD EJECTION ACCIDENT
BOC 3 RCP Core Power Versus Time

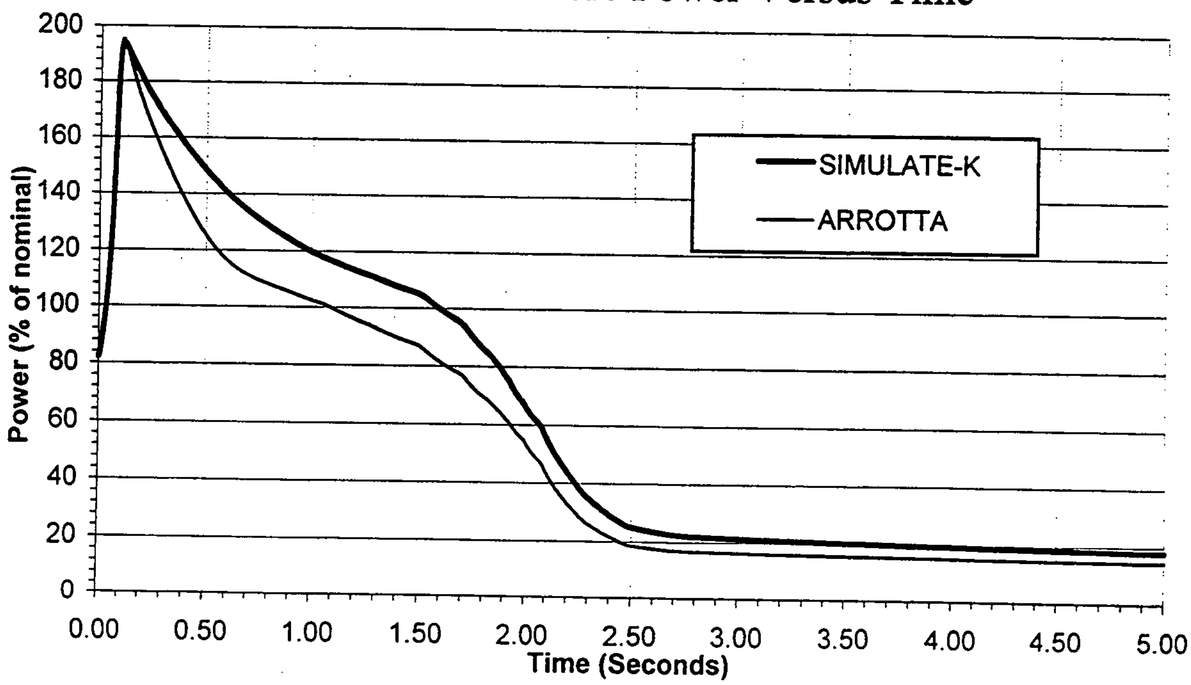


Figure 14-4

ROD EJECTION ACCIDENT
BOC HZP Core Power Versus Time

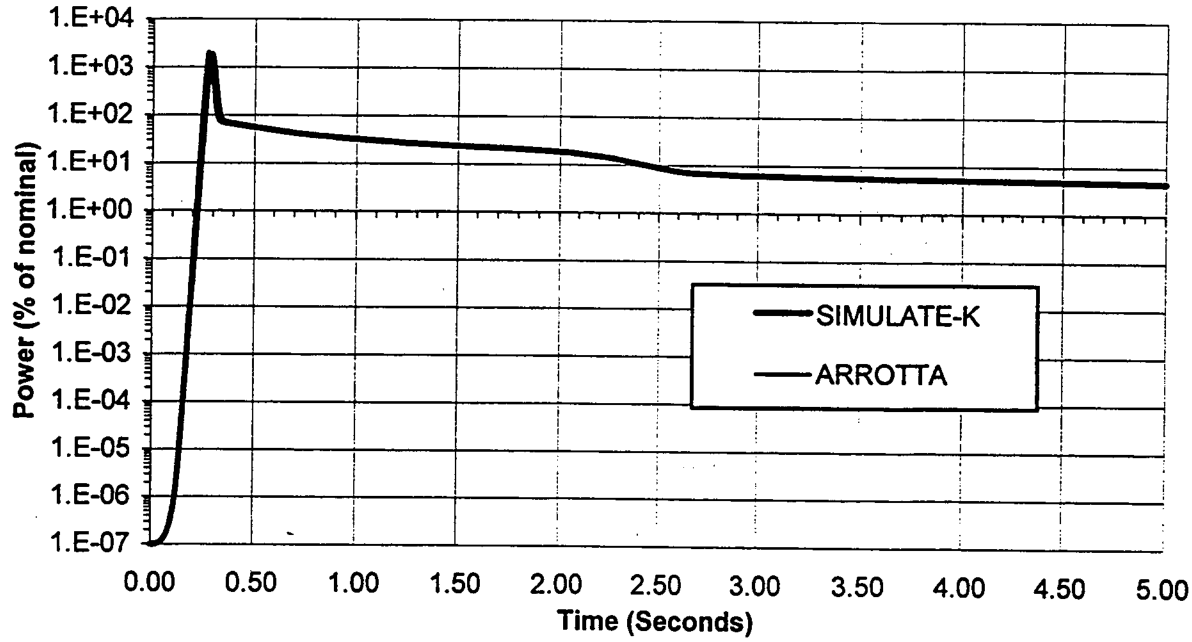


Figure 14-5

ROD EJECTION ACCIDENT
EOC 4 RCP Core Power Versus Time

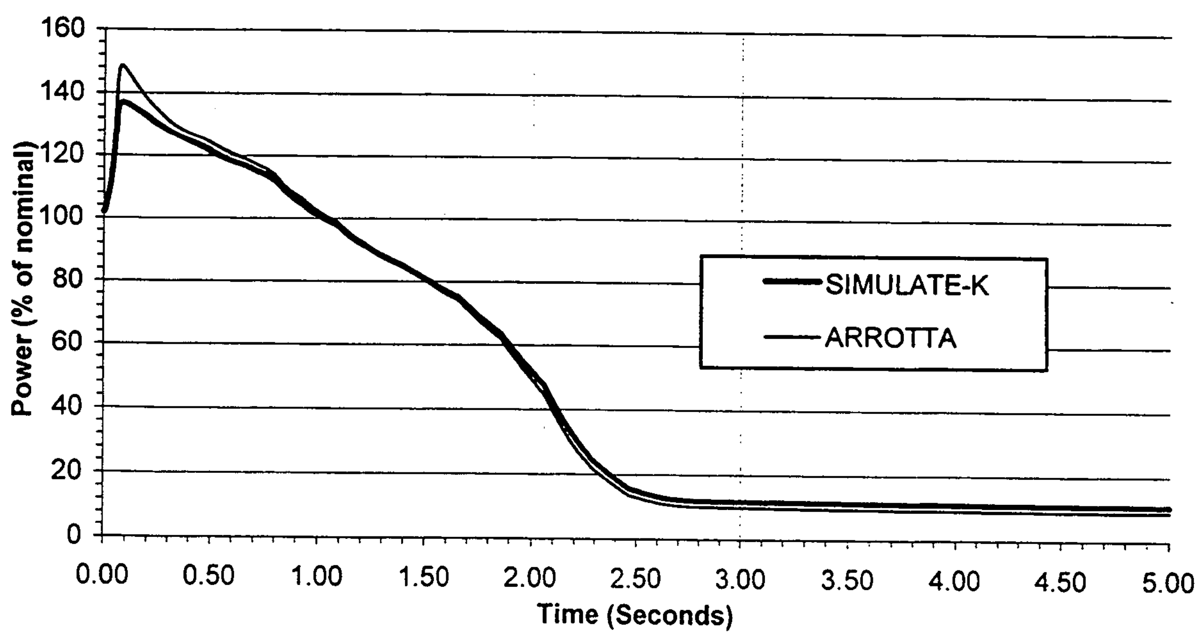


Figure 14-6

ROD EJECTION ACCIDENT
EOC 3 RCP Core Power Versus Time

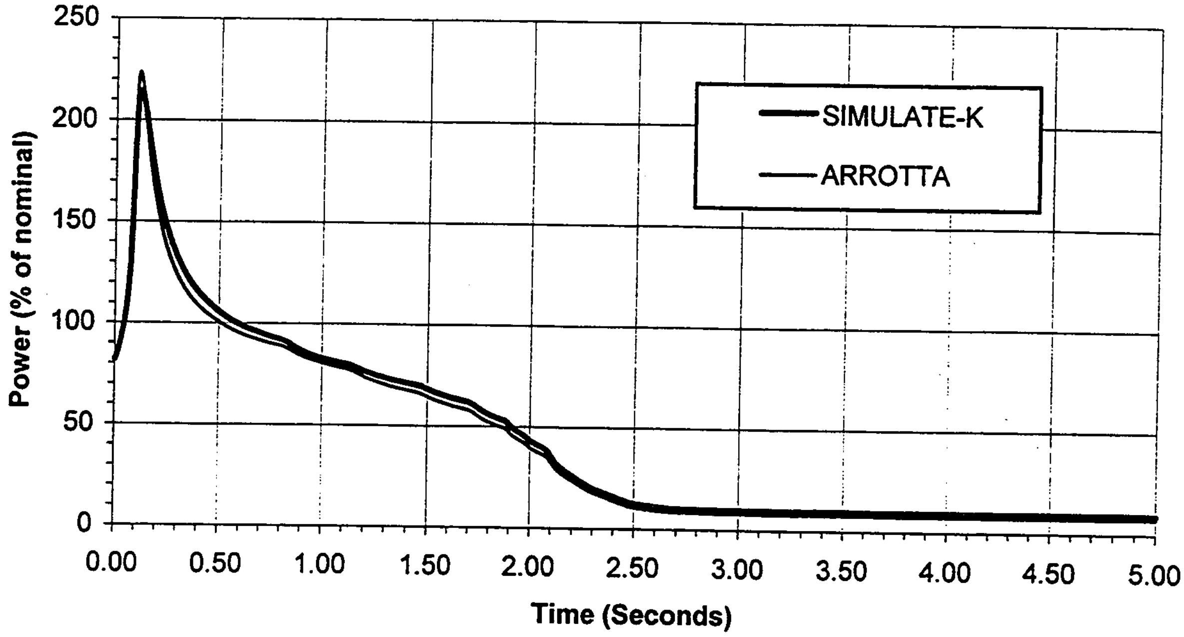


Figure 14-7

ROD EJECTION ACCIDENT
EOC HZP Core Power Versus Time

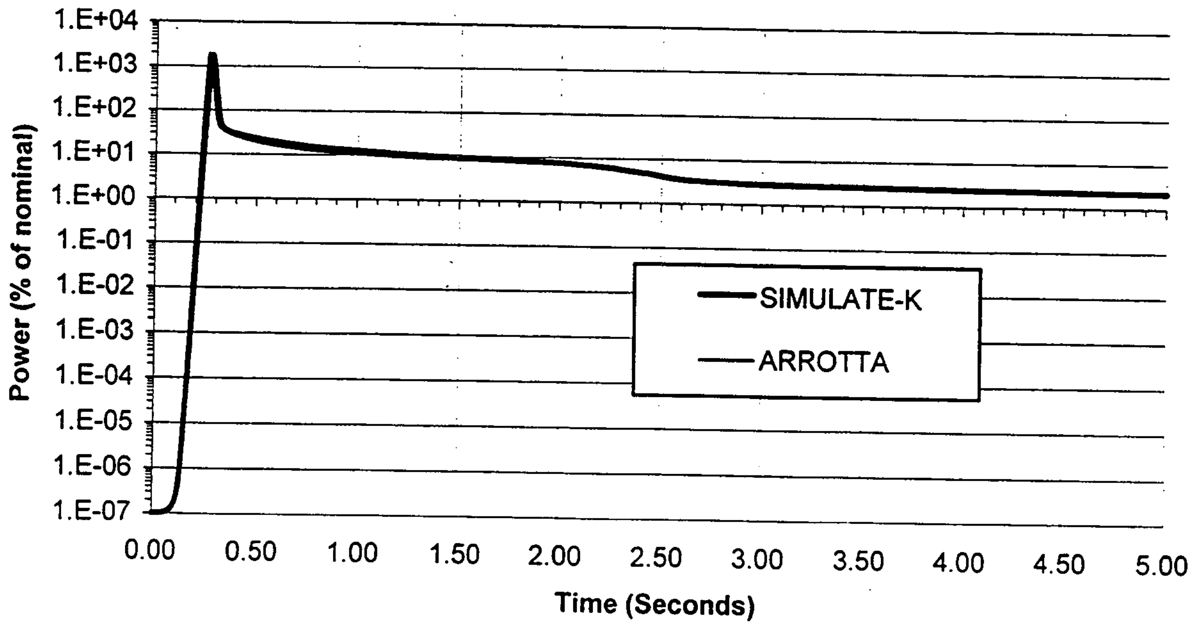


Figure 14-8

ROD EJECTION ACCIDENT

Assembly Power Distribution at Maximum Core Power

BOC with 4 RCP

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						0.181	0.270	0.288	0.278	0.192					
B				0.160	0.315	0.698	1.099	0.920	1.138	0.748	0.353	0.185			
C			0.207	0.623	0.744	1.112	1.391	1.389	1.451	1.212	0.857	0.753	0.264		
D		0.153	0.612	0.679	0.994	0.943	1.371	1.282	1.450	1.063	1.195	0.849	0.827	0.222	
E		0.302	0.720	0.975	0.998	1.222	1.448	1.439	1.569	1.441	1.289	1.344	1.052	0.462	
F	0.170	0.659	1.057	0.906	1.190	0.924	1.298	1.251	1.453	1.170	1.670	1.372	1.665	1.066	0.281
G	0.250	1.025	1.302	1.290	1.370	1.249	1.051	1.048	1.233	1.683	2.093	2.151	2.196	1.708	0.424
H	0.263	0.844	1.278	1.190	1.315	1.144	0.978	0.752	1.176	1.590	2.107	2.384	2.233	1.441	0.452
K	0.250	1.025	1.302	1.290	1.370	1.249	1.051	1.048	1.233	1.683	2.093	2.150	2.195	1.707	0.424
L	0.170	0.659	1.057	0.906	1.190	0.924	1.298	1.251	1.453	1.170	1.669	1.372	1.665	1.066	0.281
M		0.303	0.721	0.976	0.998	1.222	1.447	1.439	1.569	1.441	1.289	1.343	1.051	0.461	
N		0.157	0.614	0.679	0.993	0.943	1.370	1.282	1.450	1.063	1.196	0.849	0.824	0.217	
O			0.207	0.621	0.743	1.111	1.391	1.389	1.452	1.212	0.858	0.754	0.264		
P				0.156	0.314	0.698	1.099	0.920	1.138	0.748	0.355	0.190			
R						0.181	0.269	0.288	0.278	0.192					

Figure 14-9

ROD EJECTION ACCIDENT

Assembly Power Distribution at Maximum Core Power

BOC with 3 RCP

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						0.163	0.244	0.265	0.262	0.184					
B				0.135	0.275	0.619	0.978	0.835	1.057	0.717	0.350	0.187			
C			0.165	0.505	0.636	0.975	1.222	1.224	1.342	1.172	0.855	0.752	0.274		
D		0.124	0.486	0.504	0.841	0.833	1.196	0.998	1.355	1.067	1.228	0.810	0.893	0.254	
E		0.248	0.588	0.804	0.860	1.089	1.316	1.342	1.555	1.516	1.420	1.513	1.227	0.557	
F	0.140	0.540	0.864	0.758	1.027	0.838	1.223	1.242	1.524	1.310	1.945	1.646	2.037	1.326	0.355
G	0.206	0.829	1.045	1.039	1.165	1.125	1.011	1.081	1.360	1.953	2.522	2.657	2.744	2.147	0.539
H	0.216	0.682	1.004	0.833	1.101	1.032	0.946	0.791	1.322	1.873	2.571	3.007	2.811	1.824	0.576
K	0.206	0.830	1.045	1.039	1.165	1.125	1.011	1.081	1.360	1.953	2.522	2.657	2.743	2.147	0.539
L	0.141	0.540	0.864	0.758	1.027	0.838	1.222	1.242	1.524	1.310	1.945	1.646	2.037	1.325	0.355
M		0.249	0.589	0.804	0.860	1.089	1.316	1.342	1.555	1.516	1.419	1.512	1.226	0.556	
N		0.127	0.487	0.504	0.840	0.833	1.196	0.998	1.355	1.067	1.228	0.810	0.891	0.248	
O			0.165	0.504	0.635	0.974	1.221	1.224	1.342	1.172	0.856	0.754	0.274		
P				0.132	0.274	0.619	0.978	0.835	1.057	0.717	0.351	0.191			
R						0.163	0.244	0.265	0.262	0.185					

Figure 14-10

ROD EJECTION ACCIDENT

Accumulated Pins in DNB

BOC with 4 RCP

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						0	0	0	0	0					
B				0	0	0	92	0	104	0	0	0			
C			0	0	0	42	208	208	208	126	0	0	0		
D		0	0	0	0	0	208	208	208	0	61	0	0	0	
E		0	0	0	0	167	208	208	208	208	191	208	31	0	
F	0	0	13	0	140	0	208	141	208	3	208	208	208	85	0
G	0	66	208	208	208	202	0	0	137	208	208	208	208	206	0
H	0	0	208	208	208	23	0	0	94	208	208	208	208	179	0
K	0	66	208	208	208	202	0	0	139	208	208	208	208	206	0
L	0	0	13	0	140	0	208	140	208	4	208	208	208	85	0
M		0	0	0	0	167	208	208	208	208	193	208	31	0	
N		0	0	0	0	0	208	208	208	0	61	0	0	0	
O			0	0	0	41	208	208	208	126	0	0	0		
P				0	0	0	91	0	104	0	0	0			
R						0	0	0	0	0					

Figure 14-11

ROD EJECTION ACCIDENT

Accumulated Pins in DNB

BOC with 3 RCP

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A						0	0	0	0	0					
B				0	0	0	53	0	73	0	0	0			
C			0	0	0	2	208	208	208	88	0	0	0		
D		0	0	0	0	0	208	115	208	0	92	0	0	0	
E		0	0	0	0	75	208	200	208	208	208	208	119	0	
F	0	0	0	0	33	0	205	132	208	208	208	208	208	128	0
G	0	12	194	157	201	157	0	0	199	208	208	208	208	208	0
H	0	0	162	80	161	2	0	0	185	208	208	208	208	208	0
K	0	12	195	157	201	157	0	0	199	208	208	208	208	208	0
L	0	0	0	0	30	0	205	129	208	208	208	208	208	128	0
M		0	0	0	0	76	208	198	208	208	208	208	118	0	
N		0	0	0	0	0	208	115	208	0	92	0	0	0	
O			0	0	0	2	208	208	208	88	0	0	0		
P				0	0	0	53	0	73	0	0	0			
R						0	0	0	0	0					

Figure 14-12

ROD EJECTION ACCIDENT
BOC 4 RCP

Figure 14-13

ROD EJECTION ACCIDENT
BOC 3 RCP

Figure 14-14

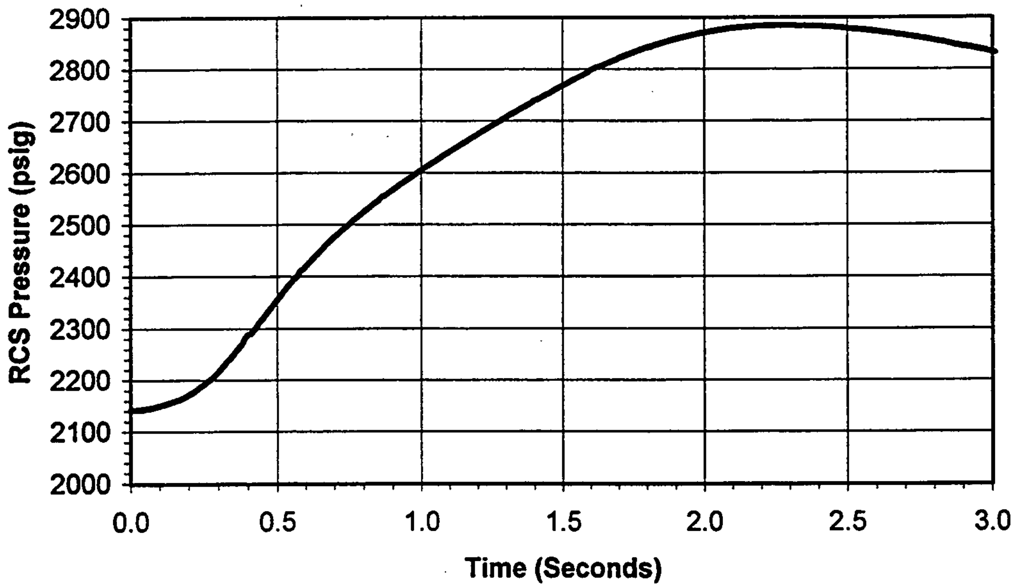
ROD EJECTION ACCIDENT
BOC HZP

Figure 14-15

ROD EJECTION ACCIDENT
BOC 3 RCP

Figure 14-16

ROD EJECTION ACCIDENT
BOC 3 RCP



15.0 STEAM LINE BREAK

15.1 Overview

15.1.1 Description

The steam line break accident initiates with a double-ended rupture of one of the two main steam lines. Since the two steam lines are connected in the steam chest between the turbine stop valves and the control valves, the break initially results in a rapid blowdown of both steam generators. The steam generator depressurization initiates a rapid Reactor Coolant System (RCS) cooldown leading to a reactor trip on low RCS pressure or variable low RCS pressure within the first few seconds of the accident. The reactor trip causes the turbine stop valves to close, isolating the affected steam generator from the unaffected steam generator. Main feedwater flow to each steam generator will be controlled by the Integrated Control System (ICS) by maintaining a minimum post trip steam generator level. If main feedwater is available and controlling steam generator level to the ICS setpoint, emergency feedwater will not be actuated. If main feedwater is lost, or if the ICS fails to control feedwater flow to the affected steam generator, emergency feedwater is likely to be actuated. The affected steam generator continues to depressurize, while the pressure in the isolated steam generator repressurizes and is controlled by the turbine bypass valves and possibly the main steam safety valves. Auxiliary steam loads may also depressurize the isolated steam generator. The cooldown of the RCS continues, resulting in reverse heat transfer in the isolated steam generator. The cooldown of the RCS caused by the continued addition of main and/or emergency feedwater to the depressurized steam generator may lead to a loss of shutdown margin and a return-to-power. Any return-to-power is eventually shut down by the boron injected from the High Pressure Injection (HPI) System and core flood tanks (CFTs).

15.1.2 Acceptance Criteria

The acceptance criteria for the steam line break accident are as follows:

- The core will remain intact for effective core cooling, assuming minimum tripped rod worth with a stuck rod.
- Doses will be within 100% of 10CFR100 limits.

The steam line break analysis is performed assuming a stuck control rod, a single failure in the Engineered Safety Features or the Emergency Feedwater System, and with consideration of both offsite power maintained and offsite power lost. Fuel failure will be assumed for any fuel pin that exceeds the DNBR limit.

15.1.3 Analytical Approach

The steam line break transient requires a limiting set of physics parameters to be determined for use as initial and boundary conditions. These parameters are input to the Oconee RETRAN-02 model (References 15-1 and 15-2) for the system thermal-hydraulic analysis. The with offsite power RETRAN-02 analysis generates the transient core thermal-hydraulic boundary conditions (core heat flux, core inlet flow, core inlet temperature and core exit pressure). The steam line break with offsite power is a severe overcooling transient which results in a return-to-power condition. The affected loop cold leg temperatures are much colder than the unaffected loop and cause asymmetric core inlet temperature conditions. To simulate this asymmetric condition properly, a [ ] channel VIPRE-01 (Reference 15-3) model is used (Figure 15-1). A statepoint DNBR calculation is performed since the return-to-power during the steam line break accident is slow and a statepoint analysis provides conservative DNBR results. The RETRAN-02 thermal-hydraulic statepoint is analyzed using the SIMULATE-3P (Reference 15-4) code to determine a detailed core power distribution including a stuck rod. The detailed core power distribution and the statepoint conditions are then analyzed with the VIPRE-01 code to determine the minimum DNBR.

For the without offsite power case, the core thermal-hydraulic boundary conditions from the RETRAN-02 analysis are input to the Oconee VIPRE-01 [ ] channel model (Reference 15-2) to determine the DNBR statepoint. The VIPRE-01 model is then utilized to calculate a set of maximum allowable radial peaking (MARP) limits such that DNB will not occur. The MARP limits are compared against the SIMULATE-3P core power distribution to determine the number of fuel pins exceeding the DNB limit and therefore assumed to fail.

15.2 Simulation Codes and Models

15.2.1 RETRAN-02

The RETRAN-02 Oconee base model described in Section 2.2.1 of Reference 15-2 is utilized for the steam line break analysis except as described below. The steam line break model has been previously submitted (Reference 15-5) and approved by the NRC for the Oconee steam line break accident mass and energy release modeling.

15.2.1.1 Nodalization of Reactor Vessel

[ ]

15.2.1.2 Transport Delay Model

Results indicate that reverse heat transfer in the unaffected steam generator causes significant voiding in the RCS loop associated with this generator. The voiding is severe enough to cause degraded reactor coolant pump (RCP) performance which leads to brief periods of reverse flow in the isolated loop. Since reverse flow can cause anomalous predictions when the transport delay model is used, this model is deleted from the primary system piping volumes.

15.2.1.3 Condensate/Feedwater System Model

A Condensate/Feedwater System model is added to the RETRAN base deck to accurately predict the feedwater flow boundary condition during the steam line break accident. The Condensate/Feedwater System model contains fill tables to simulate the condensate booster pumps and the D heater drain pumps. Homologous pump curves are included to accurately model the main feedwater pumps. Non-conducting heat exchangers are used to model all of the feedwater heaters.

15.2.1.4 Steam Generator Model

Low steam generator tube plugging will maximize the transient primary-to-secondary heat transfer. The assumption of low steam generator tube plugging also maximizes the RCS volume, which slightly increases the overall heat capacity of the RCS. Sensitivity studies have been performed and have determined that the impact of the tube plugging on the heat transfer area is the dominant effect. Based upon plant data, a lower bound of 1% tube plugging is modeled.

The vertical junction option is used for the aspirator junctions to smooth the enthalpy and mass flow rate predictions through the aspirator port during the accident. This is necessary due to the reverse flow predicted through these junctions during the accident. The inertia for these junctions is also increased in order to minimize the rate of change in flow through the aspirator ports. The choking option (Extended Henry and Moody) is turned on at the steam generator exit

nozzle junction, which is reasonable given the high steam velocities. The choking option is turned off at the feedwater nozzle junction, which will result in more feedwater entering the faulted steam generator. The isoenthalpic expansion choked flow option is utilized for Junctions 126, 134, 225, 226 and 234. This avoids junction enthalpy errors when the enthalpy decreases below 170 Btu/lbm (Moody limit). In addition, dynamic slip is modeled in Junctions 136 and 137. This is done in an attempt to minimize the liquid carried into the steam line.

15.2.1.5 Steam Generator Water Carryout Control

Water carryout during blowdown of the affected steam generator can have the effect of reducing the rate of overcooling, since water that does not boil in the tube bundle region will not absorb the heat of vaporization. A secondary concern with water carryout in a steam line break analysis is that the break flow with two-phase conditions will be considerably less on a volumetric basis than single-phase steam flow, and will thus slow the rate of steam generator depressurization. This in turn slows the decrease in steam generator saturation temperature and the primary-to-secondary heat transfer rate, which is non-conservative. However, with uncontrolled main and/or emergency feedwater flow, steam generator overfill will eventually occur. Water carryout at that time is realistic. Possible unrealistic or non-conservative water carryout is addressed in the model.

15.2.1.6 <u>Break Model</u>

The break is modeled by dividing the ruptured main steam line into two volumes with a connecting junction, and by adding the two break junctions. The full cross-sectional area of the 34" main steam line is 6.3 ft^2. Thus, the double-ended break of the 34" main steam line results in a total initial break flow area of 12.6 ft^2.

[ ]

15.2.2 VIPRE-01

The VIPRE-01 code is used for the steam line break core thermal-hydraulic analyses. VIPRE-01 thermal-hydraulic boundary conditions (core exit pressure, core inlet temperature, core inlet flow, and heat flux) are obtained from the RETRAN-02 system transient simulation. Since the

[ ]

[ ] The modeling of conservative factors, direct moderator heating, flow correlations, and other correlations is identical to that described in Reference 15-2. The subcooled and bulk void correlations are different than those in Reference 15-2, and are described in Section 15.3.1.2.3. The critical heat flux (CHF) correlations used to evaluate the DNBR are the Westinghouse W-3S (Reference 15-3, Appendix D) for the Mk-B10 or Mk-B11 fuel types, and the BWU (References 15-7 and 15-8) correlations for Mark-B11 fuel.

For the without offsite power analysis, the [ ] channel VIPRE-01 model described in Reference 15-2 is used to calculate the transient local coolant properties and DNBR. The BWC (Reference 15-6) and BWU CHF correlations are used to perform the DNBR calculations for the Mk-B10T and Mk-B11 fuel assembly types, respectively. The VIPRE-01 analysis employs the SCD methodology for the offsite power lost case.

15.2.3 SIMULATE-3P

SIMULATE-3P is used to generate safety analysis physics parameters and three-dimensional core pin power distributions. The system transient response during a steam line break accident is sensitive to core temperature feedback. The moderator reactivity versus temperature and the Doppler reactivity versus fuel temperature curves are selected such that the most limiting conditions, which occur at end-of-cycle (EOC), are predicted.

The asymmetric conditions for the with offsite power analysis require non-uniform core inlet temperatures to be input to SIMULATE-3P. The maximum worth stuck rod is conservatively assumed to be in the cold half of the core which will increase the local reactivity and power. A 10% reduction in the worth of the remaining control rods is also assumed. These assumptions result in a conservative reactivity calculation and power distribution at the limiting RETRAN statepoint. The SIMULATE-3P reactivity prediction is used to verify that the RETRAN kinetics model is conservative. The SIMULATE-3P pin power distribution at the limiting RETRAN statepoint is then input to VIPRE for the DNBR analysis.

For the without offsite power analysis, the stuck rod is conservatively assumed to be in the colder half of the core since this will increase the local reactivity and power. SIMULATE-3P is used to calculate the pin power distribution which is used to compare to the MARP limits generated in the VIPRE analysis.

15.3 Transient Analysis

The steam line break analysis presented herein is divided into two sections. The first section assumes that offsite power is available, and is concerned with the potential for a post-trip return-to-power and DNB. The second section assumes that offsite power is lost coincident with the opening of the break, and is concerned with the flow coastdown and primary system depressurization effects on DNB.

15.3.1 With Offsite Power

15.3.1.1 RETRAN-02 Analysis

15.3.1.1.1 Initial Conditions

The initial conditions for the steam line break analysis with offsite power are selected to maximize the RCS cooldown and depressurization, and thereby maximize the potential for a post-trip return-to-power and DNB. Since the SCD methodology does not cover the range of RCS pressures expected for the cases that assume offsite power is available, a deterministic approach will be utilized in the selection of the initial conditions.

Power Level

Full rated power plus uncertainty is assumed. High initial power level maximizes the initial steam generator inventory and feedwater flow rate, both of which will maximize the primary-to-secondary heat transfer once the break occurs. A steam line break accident from hot zero power (HZP) is not analyzed. At HZP, feedwater is aligned through the startup feedwater control valves, which results in a much lower feedwater flow rate than at full power. Sensitivity studies have been performed and have determined that the steam line break from HZP is bounded.

RCS Pressure

A low initial pressure minimizes the time to reactor trip. An earlier trip reduces the integrated energy deposition into the RCS, leading to lower RCS temperatures. A lower initial pressure is also conservative with respect to DNB. However, a lower initial pressure results in an earlier actuation of the Engineered Safeguards Systems (HPI and CFTs) which inject boron into the

RCS and shut down the reactor if a return-to-power occurs. Sensitivity studies have been performed and have determined that a low initial RCS pressure is the most limiting assumption.

<u>Pressurizer Level</u>

A low initial pressurizer level minimizes the volume of relatively hot water that drains into the RCS upon pressurizer outsurge, thereby maximizing the RCS cooldown and any return-to-power. Thus, nominal pressurizer level less uncertainty is assumed.

<u>RCS Temperature</u>

The ICS controls the average coolant temperature at a constant value whenever power is greater than 15%. For the steam line break accident, a lower initial average coolant temperature will result in a greater cooldown of the primary system. This will result in more positive reactivity addition due to the negative moderator temperature coefficient, and thus maximize any return-to-power. Thus, nominal RCS average temperature less uncertainty is assumed.

<u>RCS Flow</u>

Since this transient is being evaluated for minimum DNBR, a low initial RCS flow is used.

<u>Core Bypass Flow</u>

High core bypass flow is assumed which minimizes core flow and is conservative for DNB.

<u>Fuel Temperature</u>

A low initial fuel temperature is used to minimize the stored energy in the fuel. A conservatively low EOC fuel temperature is assumed.

<u>Steam Generator Mass</u>

A conservatively high steam generator mass is assumed to maximize the overcooling.

15.3.1.1.2 Boundary Conditions

The key boundary conditions for the steam line break with offsite power are as follows:

Break Opening Time

A break opening time of 0.1 seconds is assumed. Based on sensitivity studies performed, shorter break opening times do not significantly alter the initial secondary side depressurization.

Reactor Coolant Pump Modeling

The RETRAN two-phase flow degradation model is used for the RCPs since significant voiding is predicted in the unaffected loop. The RCPs in the unaffected loop are tripped at 100 seconds to avoid a code error associated with pressure oscillations in this loop due to two-phase pump performance. Tripping the RCPs in the unaffected loop has a conservative impact on the cooldown of the RCS since there is reverse heat transfer taking place in the steam generator.

Turbine Stop Valves

A slow turbine stop valve stroke time (1.0 second) is assumed to isolate the unaffected steam generator from the affected steam generator. This maximizes the overcooling.

Main Steam Safety Valves

The main steam safety valves are modeled using conservative assumptions for drift, blowdown and valve capacity that maximize relief flow and minimize the secondary pressure response in the unaffected steam generator. A lower pressure will minimize the reverse primary-to-secondary heat transfer in this steam generator, and maximize the RCS cooldown.

Extraction Steam

To maximize the cooldown of the RCS, it is conservative to model the steam loads on the isolated steam generator. A conservatively high extraction steam flow rate is assumed.

Decay Heat

To maximize the RCS cooldown, a low decay heat power level assuming a multiplier of 0.9 is applied to the 1979 ANS Standard 5.1 decay heat power.

Single Failure

The analysis examines a single failure of the EFW control valve to the affected steam generator or a single failure of the Engineered Safeguards that results in only one train of HPI.

15.3.1.1.3 Physics Parameters

Moderator Temperature Feedback

A table of reactivity as a function of moderator density is input to account for moderator reactivity effects. The consequences of a steam line break accident are more severe at EOC due to the more negative moderator temperature coefficient. The most negative EOC moderator temperature feedback curve is used in the analysis.

Doppler Temperature Feedback

A table of reactivity as a function of fuel temperature is input to model Doppler reactivity effects. The most negative Doppler curve is used in the analysis. However, the most negative Doppler curve will also result in the largest negative feedback during any return-to-power. Since the boron injected by the HPI system and CFTs limits the return-to-power (rather than the negative reactivity due to Doppler feedback) it is conservative to assume a most negative Doppler curve.

Reactivity Weighting

Beta-effective and Neutron Lifetime

A small value of β_{eff} and prompt neutron lifetime are chosen to maximize the power decrease on reactor trip. Small values of these parameters will also enhance any return-to-power. EOC decay constants and delayed neutron precursor fractions are also assumed.

Scram Curve and Worth

The control rods are inserted when the reactor trips. For this analysis, a top-peaked scram curve and a lower bound on the rod insertion time are assumed. These assumptions minimize the post-trip energy addition to the RCS, leading to a greater cooldown. A scram worth is selected which maintains a reactivity margin between the RETRAN-02 and SIMULATE-3P reactivity predictions at the limiting RETRAN statepoint.

Boron Reactivity

[ ]

Differential Boron Worth

A differential boron worth is used to model the reactivity addition from the boron injected by the HPI pumps and the CFTs. A low differential boron worth (% $\Delta k/k$/ppm) is conservative in that it will minimize the negative reactivity added by these systems.

15.3.1.1.4 Control, Protection, and Safeguards Systems

Reactor Control

Following the steam line break, the combined effect of decreasing turbine header pressure and T-ave would result in an increase in reactor demand to the high limit. Since a reactor trip will occur within the first few seconds of the accident, it is reasonable to make the simplifying assumption that the control rods are in manual control.

Reactor Trip

An early reactor trip is conservative in that it minimizes the integrated energy transferred into the RCS, leading to a more severe cooldown. Thus, the variable low pressure trip and low RCS pressure trip setpoints are adjusted to ensure an early reactor trip occurs. A lower bound on the delay time for both trip functions is used.

RCS Pressure Control

No credit is taken for pressurizer heater operation. Due to the rapid depressurization of the RCS, the pressurizer sprays, PORV, and safety valves are not actuated.

Pressurizer Level Control

No credit is taken for the automatic operation of makeup and letdown to attempt to maintain pressurizer level. The makeup and letdown flows are assumed to isolate simultaneously and to be balanced prior to isolation. Not taking credit for the makeup and letdown is conservative for the evaluation of minimum DNBR.

Emergency Core Cooling System

Minimum HPI flow is conservative for the steam line break, since the injected borated water from this system helps to prevent or terminate any return-to-power as well as repressurize the RCS. The HPI system is simulated using fill tables that model the A and B HPI pumps injecting through the A train and the C HPI pump injecting through the B train. Sensitivity studies have examined the effect of a failure in the 4160V switchgear or the failure of the EFW control valve to the affected steam generator. For the cases that assume an EFW control valve single failure, three HPI pump minimum flow is credited. Since reverse flow is established in the unaffected loop, the A and B pump flow is injected in the unaffected loop. This maximizes the flowpath the injected boron must take to reach the core inlet, which delays the boron negative reactivity addition. For cases that assume the failure of one of the three available 4160V switchgear, the A train of HPI is assumed to be lost. This results in the C pump injecting through the B train for the first 10 minutes. The C pump injected flow occurs in the unaffected loop to maximize the delay in the boron negative reactivity addition. The presence of any unborated water initially in the HPI piping is modeled. A conservative minimum boron concentration is also assumed.

Similarly, the boron concentration in the CFTs is assumed to be a conservative minimum value. Lower bounds on the initial CFT inventory, pressure and temperature are also assumed. These assumptions will delay CFT injection, minimize the available inventory of borated water and maximize the RCS cooldown.

Main Feedwater System

Since a reactor trip occurs within the first few seconds of the accident, changes in feedwater control over this period of time will have a negligible impact on the accident. Following reactor trip, the ICS rapidly decreases feedwater demand to zero, and then feedwater flow is restored when steam generator level drops below the minimum level control setpoint. With the ICS in manual MFW flow will continue and, assuming no credit for ICS control or operator action, steam generator overfill will occur. The limiting assumption with respect to maximizing the overcooling and reactivity addition has been determined by analysis to be the case with the ICS controlling MFW to the minimum steam generator level setpoint including uncertainty.

Emergency Feedwater System

The three emergency feedwater (EFW) pumps automatically start upon a loss of both main feedwater pumps, and the two motor-driven pumps also start on a low steam generator level. Low main feedwater pump discharge pressure (ATWS Mitigation System Actuation Circuit) can also result in actuation of all three EFW pumps. If the ICS is functioning to throttle MFW flow by controlling on steam generator level, EFW is not modeled. For the cases that assume the ICS does not throttle MFW flow, EFW is actuated when the low MFW pump discharge pressure setpoint plus uncertainty is satisfied. Maximum EFW flow is assumed to maximize the

cooldown. Nominally, the EFW flow is controlled to maintain a minimum steam generator level. The analysis assumes the EFW level control setpoint is higher and includes uncertainty. For the cases assuming a single failure in the EFW System, the EFW control valve to the affected steam generator is assumed to fail full-open. A conservatively low temperature is assumed for the EFW.

Main Steam Line Break Detection and Main Feedwater Isolation Instrumentation

This instrumentation is not credited in the analysis.

Turbine Control

The steam line break causes a rapid decrease in steam generator pressure. Thus, the ICS will attempt to close the turbine control valves in order to restore turbine header pressure to its setpoint. Since steam flow to the turbine is maximized if the turbine control valves remain open, it is conservative to assume that turbine control is in manual.

Turbine Bypass System

The Turbine Bypass System is assumed operable to limit the post-trip pressure in the unaffected steam generator, thereby minimizing the secondary-to-primary heat transfer from the unaffected steam generator to the RCS. This is conservative for maximizing the RCS cooldown.

15.3.1.1.5 With Offsite Power Results

The steam line break with offsite power analysis assumes the ICS controls the post-trip SG level to an uncertainty adjusted setpoint of 100 inches. The single failure is assumed to be a train of Engineered Safety Features that results in only one train of HPI for the first 10 minutes. Table 15-1 gives the sequence of events for this case.

The steam line break initially causes the pressure to decrease in both steam generators (Figure 15-3). Break flowrates (Figure 15-4) for both steam generators rapidly increase. After the turbine stop valves close, break flow from the unaffected steam generator stops. Beyond this point, break flow from the affected steam generator decreases with decreasing pressure, and the unaffected steam generator repressurizes and opens the turbine bypass valves and the first bank of main steam safety valves for a short period of time. The unaffected steam generator gradually

depressurizes due to reverse heat transfer and extraction steam loads. Both steam generators are nearly fully depressurized by the end of the simulation.

The cooldown in the affected loop is initially much more severe than in the unaffected loop, as shown in the cold leg and hot leg temperature responses (Figure 15-5). The cold leg temperature in the unaffected loop increases once the turbine stop valves close. A fairly large ΔT initially develops in the affected loop. The ΔT in the affected loop decreases over the course of the simulation as the RCS is cooled. The unaffected loop ΔT remains fairly small until this loop begins to void and the flow degrades. At about 30 seconds the unaffected loop cold leg temperature exceeds the hot leg temperature. This is the result of reverse heat transfer in the unaffected loop and the beginning of flow degradation in the unaffected loop. At approximately 140 seconds, the unaffected loop hot leg temperature exceeds the unaffected loop cold leg temperature. This is a result of the RCP trip in the unaffected loop at 100 seconds. After the RCPs coast down, the flow reverses in the unaffected loop. Due to flow stagnation and the injection of cold HPI inventory, the unaffected loop cold leg temperature falls below the affected loop cold leg temperature at approximately 170 seconds. The bulk of the RCS has cooled to approximately 270 °F by the end of the simulation.

The total, moderator, Doppler, boron and control rod reactivities are presented in Figure 15-6. The negative reactivity insertion at the beginning of the transient is due to the reactor trip and control rod insertion. The cooldown causes positive reactivity insertion due to the negative moderator and Doppler coefficients. The core returns to a critical condition at approximately 140 seconds. Injected boron from the HPI system and the CFTs reaches the core at approximately 160 seconds. The negative reactivity inserted by the boron returns the core to a subcritical condition by approximately 200 seconds. Subcriticality is maintained for the remainder of the simulation.

The reactor power (Figure 15-7) decreases rapidly on reactor trip. The thermal power generally follows the neutron power response. The fluctuations in the heat flux are caused by flow surges in the core which result from flow degradation due to two-phase conditions in the unaffected loop. A peak return-to-power of 13.09 %FP heat flux occurs at approximately 160 seconds. RCS pressure (Figure 15-8) rapidly decreases until the affected loop and reactor vessel head

begin to saturate at approximately 4 seconds. After this time, RCS pressure continues to decrease for the remainder of the simulation.

Core inlet mass flow (Figure 15-9) initially increases with time following the steam line break. Since the reactor coolant pumps provide essentially constant volumetric flow, the decreasing RCS temperatures initially result in an increase in mass flow. However, as the unaffected loop begins to void and RCP performance degrades as predicted by the RETRAN two-phase pump degradation model, core inlet flow decreases to approximately half of the initial flow. After the RCPs in the unaffected loop are tripped at 100 seconds, the flow oscillations diminish.

15.3.1.2 VIPRE-01 Analysis

15.3.1.2.1 Initial and Boundary Conditions

The RETRAN-02 analyses provide the limiting statepoint core exit pressure, core inlet temperature, core inlet flow rate, and core average heat flux [ ] These boundary conditions are input to VIPRE-01 as steady state boundary conditions.

15.3.1.2.2 Axial and Radial Power Distributions

<u>Axial Power Distributions</u>

[ ]

<u>Radial Power Distributions</u>

The maximum pin radial power peak in the hot assembly is calculated explicitly by SIMULATE. Also, utilizing the hot assembly pin radial power distributions as described in Reference 15-2, the hot assembly pin radial power distributions for the return-to-power situation can be derived. For [ ] For

[ ]

15.3.1.2.3 Flow Correlations

For the steam line break with offsite power case, subcooled and bulk voids are modeled with the [ ] Sensitivity studies have shown that using this combination of void correlations results in an acceptable prediction of DNBR.

15.3.1.2.4 Conservative Factors

Conservative factors described in Reference 15.2 are applied to the [ ] channel VIPRE-01 model. These conservative factors are the hot channel area reduction factors (2% for the hot unit subchannel and 3% for the hot instrumentation subchannel), the engineering hot channel factor (F_q) of 1.013, and the core inlet flow maldistribution factor. Based on the vessel model flow test and Oconee core pressure drop measurement, the core inlet flow maldistribution is conservatively modeled as a reduction in the hot assembly flow. Since in the with offsite power RETRAN-02 analysis two RCPs are assumed to trip, the hot assembly flow reduction factor for the VIPRE-01 DNB analysis is therefore [ ] as described in Section 9.3.2.3.

15.3.1.2.5 Critical Heat Flux Correlation

The W-3S CHF correlation is used for the with offsite power steam line break DNBR analysis. The historical range of applicability for the W-3S correlation is (Reference 15-3):

Pressure (psia)	1000 to 2300
Mass flux (10^6 lbm/hr-ft^2)	1.0 to 5.0
Quality (equilibrium)	-0.15 to 0.15

The W-3S CHF correlation has been approved by the NRC for analysis with system pressures as low as 500 psia and mass flux as low as 0.5 x 10^6 lbm/hr-ft^2 (References 15-9 and 15-10).

15.3.1.2.6 Results

Using the limiting statepoint from the RETRAN-02 analysis discussed in Section 15.3.1.1.5, together with the power distributions discussed in Section 15.3.1.2.2, the VIPRE-01[ ] channel model is used to calculate the core local fluid properties and MDNBR. For the with offsite power analysis the MDNBR predicted by the W-3S CHF correlation is 3.28, which is much greater than 1.45. Therefore, the acceptance criterion discussed in Section 15.1.2, is met.

15.3.1.3 SIMULATE-3P Analysis

The limiting RETRAN statepoint conditions for the steam line break analysis with offsite power are input to SIMULATE-3P. The SIMULATE analysis demonstrates that a reactivity margin is maintained between the RETRAN prediction and the SIMULATE prediction. Therefore, the RETRAN reactivity prediction is conservative. The SIMULATE core power distribution at the limiting RETRAN statepoint (Figure 15-10) is input to VIPRE for the DNBR analysis.

15.3.2 Without Offsite Power

15.3.2.1 <u>RETRAN-02 Analysis</u>

15.3.2.1.1 Initial Conditions

The initial conditions for the steam line break analysis without offsite power are selected to maximize the RCS depressurization and maximize the post-trip core power response. The steam line break analysis without offsite power is very similar to a loss of coolant flow analysis (Chapter 9.0). Thus, sensitivity study results from the loss of coolant flow analysis are utilized to select appropriate initial conditions. The transient RCS conditions for the steam line break without offsite power are within the ranges covered by the statistical core design (SCD) approach. Therefore, the analysis will utilize the SCD approach.

Power Level

Nominal full power will be assumed since the uncertainty in power is accounted for in the SCD limit.

RCS Pressure

Low initial pressure is generally conservative for DNB calculations. The SCD limit accounts for the uncertainty in indicated pressure.

Pressurizer Level

Sensitivity studies have concluded that initial pressurizer level is not an important parameter with respect to DNB for the steam line break with offsite power lost analysis.

RCS Temperature

Nominal RCS average temperature will be assumed. The indication uncertainty and ICS deadband associated with T-ave are accounted for in the SCD limit.

RCS Flow

A low initial flow rate is conservative with respect to DNB calculations. The uncertainty in RCS flow is accounted for in the SCD limit.

Core Bypass Flow

A high core bypass flow is assumed to minimize the coolant flow along the fuel rods.

Fuel Temperature

A high initial fuel temperature is conservative with respect to DNB calculations for loss of flow analyses. Since BOC kinetics parameters are assumed, a maximum BOC fuel temperature is assumed.

Steam Generator Mass

A conservatively high steam generator mass is assumed to maximize the overcooling.

15.3.2.1.2 Boundary Conditions

For a steam line break with coincident loss of offsite power, the reactor will trip and the RCPs will begin to coast down. For this scenario the accident resembles a loss of flow accident with a coincident depressurization. For a loss of flow accident, the minimum DNBR statepoint is expected within the first few seconds of the RCP coastdown. Therefore, detailed modeling of many boundary conditions that would not occur until after the limiting statepoint are unnecessary. The boundary conditions for the steam line break with offsite power lost which differ from the with offsite power case are as follows:

Loss of Offsite Power

The loss of offsite power occurs coincident with the break. The control rods are assumed to lose power coincident with the loss of offsite power. Upon losing power, control rod insertion is delayed to account for gripper coil release delay. The loss of offsite power also initiates a coastdown of the RCPs.

Decay Heat

A decay heat multiplier curve is applied to the 1979 ANS Standard 5.1 decay heat power to ensure that the RETRAN prediction of decay heat is conservatively maximized. Maximum decay heat is conservative for loss of flow DNB analyses.

Single Failure

No single failure could be identified which affects the results.

15.3.2.1.3 Physics Parameters

Moderator Temperature Coefficient

Reactivity insertion curves as a function of temperature are used to model moderator temperature feedback. BOC least negative values are conservative. This assumption minimizes the negative feedback associated with any core moderator heatup that occurs with a loss of flow.

Doppler Temperature Coefficient

Reactivity insertion curves as a function of temperature are used to model Doppler temperature feedback. BOC least negative values are conservative. This assumption minimizes the negative feedback associated with any fuel heatup that occurs with a loss of flow.

Beta-effective and Neutron Lifetime

A large β_{eff} and prompt neutron lifetime are chosen to slow the core power decrease on control rod insertion. BOC decay constants and delayed neutron precursor fractions are also utilized.

Scram Curve and Worth

The control rods are inserted when offsite power is lost. For this analysis, a bottom-peaked scram curve and an upper bound on the rod insertion time are assumed. These assumptions maximize the post-trip energy addition, which is conservative for the DNB prediction. A minimum trippable worth (not to exceed a 1% Δ/k subcritical margin), including an allowance for the most reactive rod stuck out of the core, is utilized in the analysis.

15.3.2.1.4 Control, Protection, and Safeguards Systems

Main Feedwater System

On a loss of offsite power, the hotwell pumps and condensate booster pumps will trip, resulting in a trip of the main feedwater pumps on low suction pressure. With the suction head diminishing, the MFW pumps will rapidly coastdown. A maximum coastdown time is assumed for the MFW pumps.

Emergency Feedwater System

The Emergency Feedwater System cannot start and deliver flow in the short duration of this analysis and is not modeled.

Steam Line Break Detection and Mitigation Circuitry

The steam line break detection and mitigation circuitry is not credited in the analysis.

15.3.2.1.5 Results

The steam line break without offsite power case assumes offsite power is lost coincident with the opening of the steam line break. Thus, an RCS flow coastdown also begins with the opening of the break. Table 15-2 gives the sequence of events for this case.

The steam line break initially causes the pressure to decrease in both steam generators (Figure 15-11). Once the turbine stop valves close, the unaffected steam generator repressurizes and opens the turbine bypass valves. The affected steam generator has depressurized to about 400 psig by the end of the simulation. The break flow response is similar to what has been discussed for the with offsite power analysis. The cooldown in the affected loop is much more severe than in the unaffected loop, as shown in the cold leg temperature response (Figure 15-12). The affected loop hot leg temperature is slightly higher than the unaffected loop hot leg temperature due to the outsurge of hot liquid from the pressurizer. The slight increase in hot leg temperatures from 2 to 5 seconds can be attributed to the RCS flow coastdown.

The RCS volumetric flow decreases for the duration of the simulation (Figure 15-13). This is the result of the loss of offsite power. The loss of offsite power also results in control rod insertion, which drives the core kinetics response (Figure 15-14). Due to control rod insertion, the core average fuel temperature begins to decrease. However, due to the relatively slow changes in the moderator and fuel temperatures, and given that the time period of interest for DNB is within the first 1-2 seconds of the flow coastdown, the moderator and Doppler feedback for the offsite power lost analysis are generally negligible.

The reactor power decreases rapidly on reactor trip (Figure 15-15). The core thermal power also decreases after reactor trip, but does not decrease as fast as neutron power. RCS pressure (Figure 15-16) initially decreases due to the effects of the steam line break and control rod insertion. As flow and primary-to-secondary heat transfer begin to degrade, the RCS pressure increases briefly between 2 to 5 seconds. The brief RCS pressure increase is also a result of the closure of the turbine stop valves. After this time, RCS pressure decreases for the remainder of the simulation.

15.3.2.2 VIPRE-01 Analysis

15.3.2.2.1 Initial and Boundary Conditions

The RETRAN analyses provide the transient core exit pressure, core inlet temperature, core inlet flow rate, and core average heat flux for both core halves of the split reactor vessel model. For the without offsite power analysis, both core halves have identical transient boundary conditions for the duration of the analysis. These boundary conditions are input to VIPRE as transient forcing functions.

15.3.2.2.2 Axial and Radial Power Distributions

For the SCD statepoint analysis, the axial power distribution is a chopped cosine shape with an axial peak of [ ] peaked at X/L = [ ], and the radial power distribution is the base model radial power distribution with a hot pin radial power of [ ] (Reference 15-3). For the maximum allowable radial peak (MARP) analyses, a set of axial power shapes are analyzed. The magnitude and elevation of the axial shape is varied to cover the full range of shapes resulting from the nuclear design analysis.

15.3.2.2.3 Conservative Factors

Since the SCD methodology is utilized for predicting the DNBR, the SCD limit accounts for most of the uncertainties in key parameters. Based on the vessel model flow tests and Oconee core pressure drop measurement, the core inlet flow maldistribution is conservatively modeled as a reduction in the hot assembly flow. The hot assembly flow reduction factor for four-pump operation is 5%.

15.3.2.2.4 Critical Heat Flux Correlation

The BWC critical heat flux (CHF) correlation is used for the steam line break transient DNBR analysis for the results presented. The range of applicability for the BWC CHF correlation is:

Pressure (psia)	1600 to 2600
Mass flux (Mlbm/hr-sqft)	0.43 to 3.8
Quality	-0.20 to 0.26

The BWC CHF correlation SCD limit for the steam line break transient is determined utilizing the minimum DNBR statepoint boundary conditions described in Section 15.3.2.2.5.

15.3.2.2.5 Results

The transient VIPRE DNBR results are shown in Figure 15-17, with a minimum DNBR of 1.51 at 1.90 seconds. This statepoint is used to determine the SCD limit for the steam line break transient. The MARP results are shown in Figure 15-18.

15.3.2.3 Fuel Pin Census

The MARPs are used for the fuel pin census. When the radial power peak of the fuel pin exceeds the MARP limit during the transient, DNB and cladding failure are assumed to occur. The fuel pin census is performed to determine the number of failed fuel pins during the steam line break accident. The results of the fuel pin census indicate that no peaks exceed the MARP limits, and therefore no cladding failure occurs for the steam line break accident. Based on this result the core will remain intact for effective core cooling.

15.3.3 Without Offsite Power (Using The RETRAN Point and 1-D Kinetics Models)

15.3.3.1 RETRAN-02 Analysis

The large steam line break accident without offsite power case, as described in Section 15.3.2, is simulated using both the RETRAN point kinetics and the RETRAN 1-D kinetics models in a consistent manner in order to demonstrate the 1-D kinetics methodology and to demonstrate the ability to modify reactivity feedback effects and control rod reactivity via cross section adjustments.

For both cases the initial and boundary conditions are as specified in Sections 15.3.2.1.1 and 15.3.2.1.2, respectively, with the following exceptions. In the point kinetics case the minimum control rod worth is allowed to be inserted (as opposed to limiting the inserted worth to not exceed a 1% $\Delta k/k$ subcritical margin). In addition, core power fraction and reactivity weighting are changed to be bottom peaked to be consistent with the power shape that was used to generate

the scram curve used in this model. In the 1-D kinetics case adjustments are made to cross sections to yield the same initial physics parameters (i.e. least negative moderator and Doppler temperature coefficients and the minimum scram worth) assumed in the point kinetics analysis case. The cross sections are generated using a bottom peaked core, thus yielding a conservative bottom peaked scram curve consistent with the point kinetics case. In addition, the cross sections include the effect of the most reactive rod stuck out of the core. A bottom-peaked scram curve and an upper bound on the rod insertion time maximizes the post-trip energy addition, which is conservative for the DNB prediction.

15.3.3.2 <u>Results</u>

In general the system thermal-hydraulic response of the 1-D kinetics case is very similar to the system response of the point kinetics case. The small difference in the neutron power shape (which is influenced by the scram curve shape) is attributed to the spatial effect that is captured by the 1-D kinetics model and not by the point kinetics case. The neutron power decreases rapidly on reactor trip (Figure 15-19). The core thermal power also decreases after reactor trip, but does not decrease as fast as neutron power.

The loss of offsite power also results in control rod insertion, which drives the core kinetics response (Figure 15-20). The core reactivity response of the 1-D kinetics case is very similar to the core reactivity response of the point kinetics case except for the scram curve shape as explained above. Due to control rod insertion, the core average fuel temperature begins to decrease. However, due to the relatively slow changes in the moderator and fuel temperatures, and given that the time period of interest for DNB is within the first 1-2 seconds of the flow coastdown, the moderator and Doppler feedback for the without offsite power analysis are generally negligible.

The comparison of the point and 1-D kinetics analyses of the steam line break without offsite power illustrates that the methodology for developing cross sections for the RETRAN 1-D model compares well with the point kinetics model. The spatial effect of control rod insertion as simulated with the 1-D model is well-predicted, while the components of the total reactivity effect are maintained with the cross section adjustments.

15.4 Reload Cycle-Specific Evaluation

To verify that the steam line break analysis is being performed conservatively, a reactivity margin which includes the maximum worth stuck rod and a 10% reduction in scram worth will be maintained between the RETRAN model and the SIMULATE-3P model at the limiting RETRAN statepoint for the with offsite power analysis. Each reload cycle also confirms the following core physics parameters are bounded.

- Moderator temperature coefficient (without offsite power)
- Doppler temperature coefficient (with and without offsite power)
- Minimum scram worth curve (without offsite power)
- Differential boron worth (with offsite power)

15.5 References

15-1 RETRAN-02: A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Revision 4, EPRI, November 1988

15-2 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000, Revision 1, Duke Power Company, July 1987

15-3 VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM-A, Revision 3, EPRI, August 1989

15-4 Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004, Duke Power Company, November 1992

15-5 Mass and Energy Release and Containment Response Methodology, DPC-NE-3003, Duke Power Company, August 1993

15-6 BWC Correlation of Critical Heat Flux, BAW-10143P-A, April 1985

15-7 BAW-10199-PA, The BWU Critical Heat Flux Correlations, Addendum 1, September 1996

15-8 Letter, D. E. LaBarge (NRC) to W. R. McCollum (Duke), SER on topical report DPC-NE-3000-PA, Revision 2, October 14, 1998

15-9 Letter, A. S. Thadani (NRC) to W. J. Johnson (Westinghouse), SER on WCAP-9226-P, Reactor Core Response to Excessive Secondary Steam Releases", January 31, 1989

15-10 Letter, T. A. Reed (NRC) to H. B. Tucker (Duke), SER on topical report DPC-NE-3001, November 15, 1991

Table 15-1

Sequence of Events

Steam Line Break - With Offsite Power

Event	Time (sec)
Break opens	0.0
Reactor trip on variable low pressure-temperature	0.7
Control rod insertion begins	0.8
Third CBP starts	1.5
Turbine stop valves closed Control rods fully inserted	1.8
MSSV opens on unaffected SG	6.9
HPI actuates	20.8
MSSV closes on unaffected SG	26.9
Boron injection from HPI begins	103.4
CFT injection begins	131.5
Boron from CFT B starts	152.0
Boron from CFT A starts	157.2
Peak return-to-power occurs	160.0
Simulation ends	600.0

Table 15-2

Sequence of Events

Steam Line Break - Without Offsite Power

Event	Time (sec)
Break Opens, LOOP Occurs	0.0
RCPs Begin Coastdown	
Condensate Booster and D Heater Drain Pumps Begin Coastdown	
MFW Pumps Begin Coastdown	
Control Rod Insertion Begins	0.14
Turbine Stop Valves Closed	1.76
Control Rods Fully Inserted	2.54
MFW Pumps Stop	5.0
Condensate Booster and D Heater Drain Pumps Stop	10.0
Simulation Ends	10.1

Table 15-3

Sequence of Events

Steam Line Break - Without Offsite Power (1-D Kinetics)

Event	Time (sec)
Break Opens, LOOP Occurs	0.0
RCPs Begin Coastdown	
Condensate Booster and D Heater Drain Pumps Begin Coastdown	
MFW Pumps Begin Coastdown	
Control Rod Insertion Begins	0.14
Turbine Stop Valves Closed	1.72
Control Rods Fully Inserted	2.54
MFW Pumps Stop	5.0
Condensate Booster and D Heater Drain Pumps Stop	10.0
Simulation Ends	10.1

Figure 15-1
Large Steam Line Break
[ ] Channel VIPRE-01 Model

Figure 15-2
Large Steam Line Break
Split Core Reactor Vessel Model

Figure 15-3

LARGE STEAM LINE BREAK
WITH OFFSITE POWER

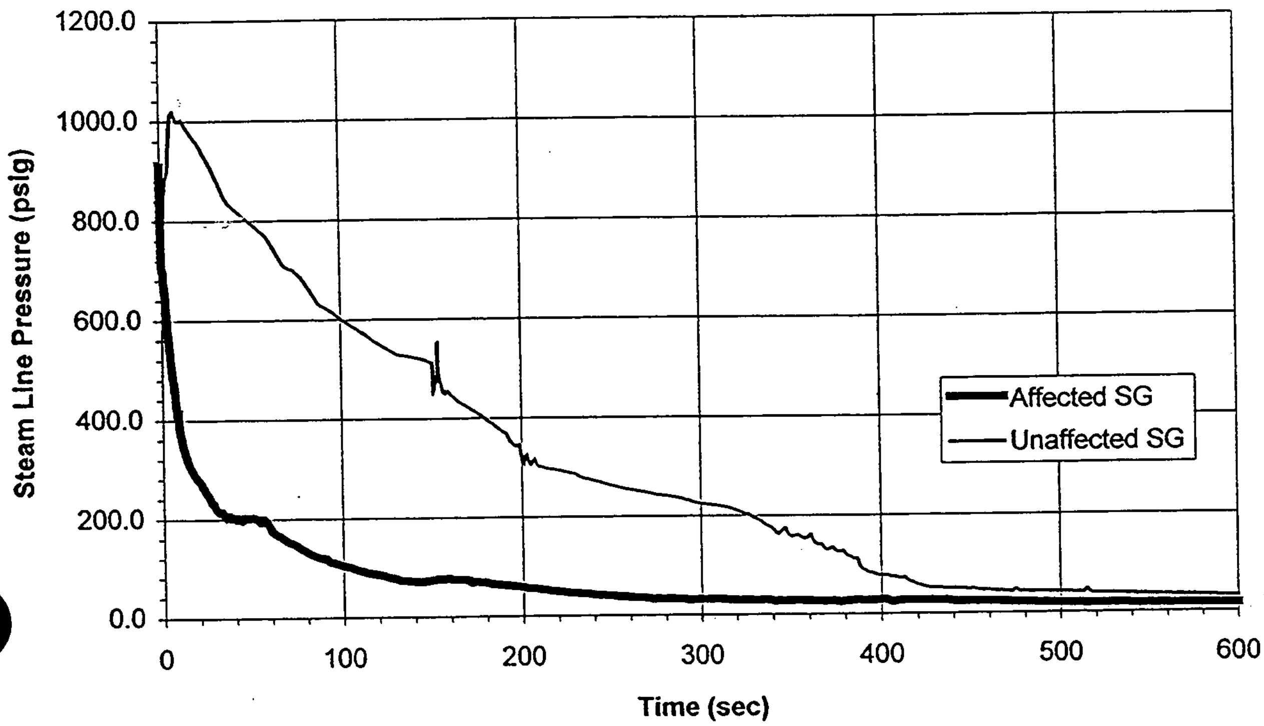


Figure 15-4

LARGE STEAM LINE BREAK
WITH OFFSITE POWER

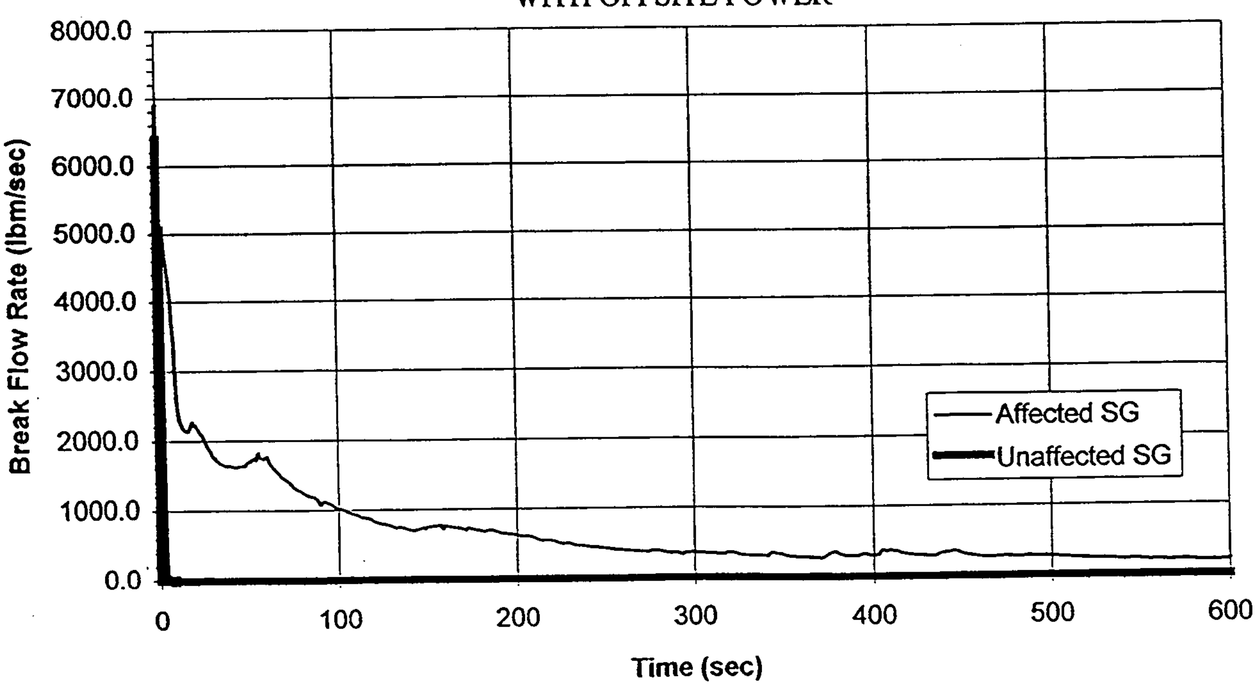


Figure 15-5

LARGE STEAM LINE BREAK

WITH OFFSITE POWER

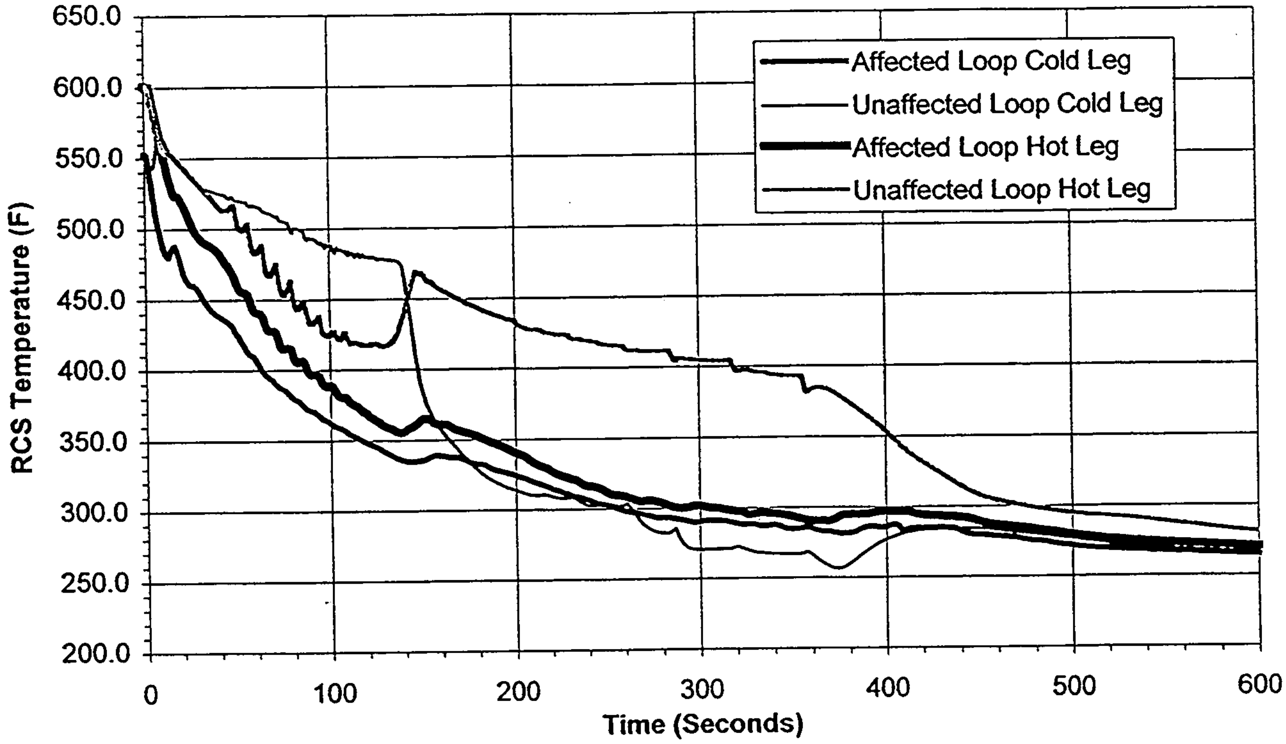


Figure 15-6

LARGE STEAM LINE BREAK

WITH OFFSITE POWER

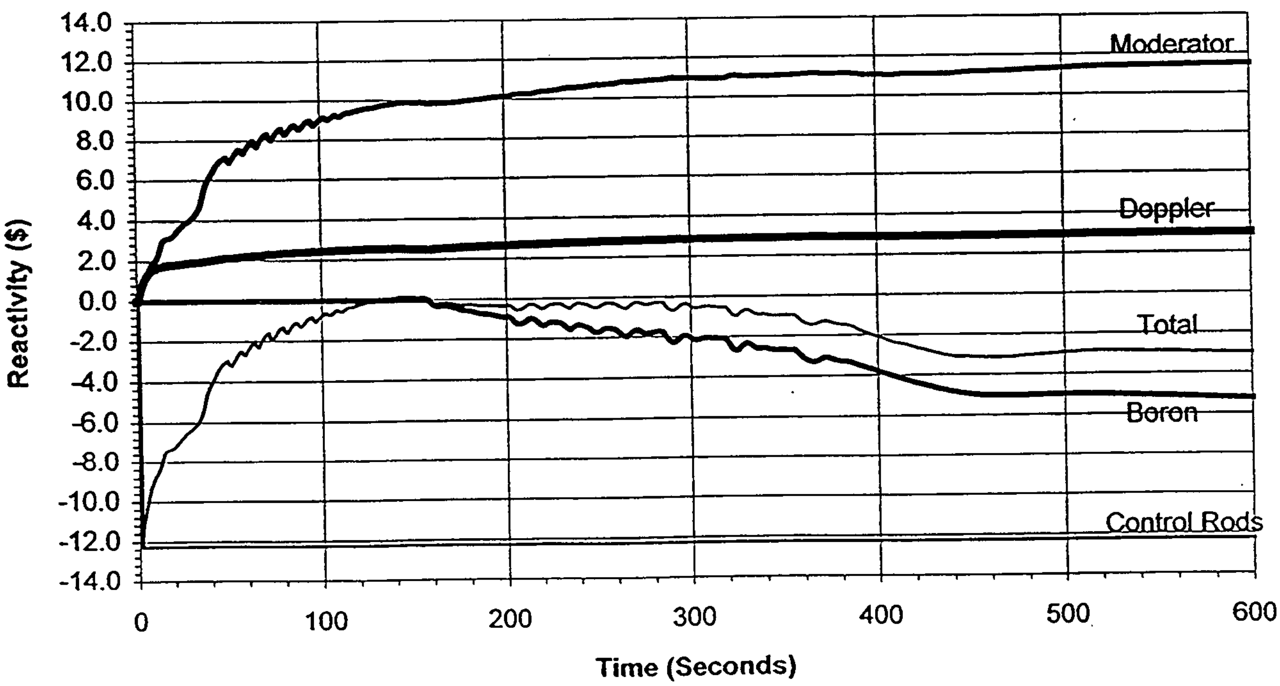


Figure 15-7

LARGE STEAM LINE BREAK

WITH OFFSITE POWER

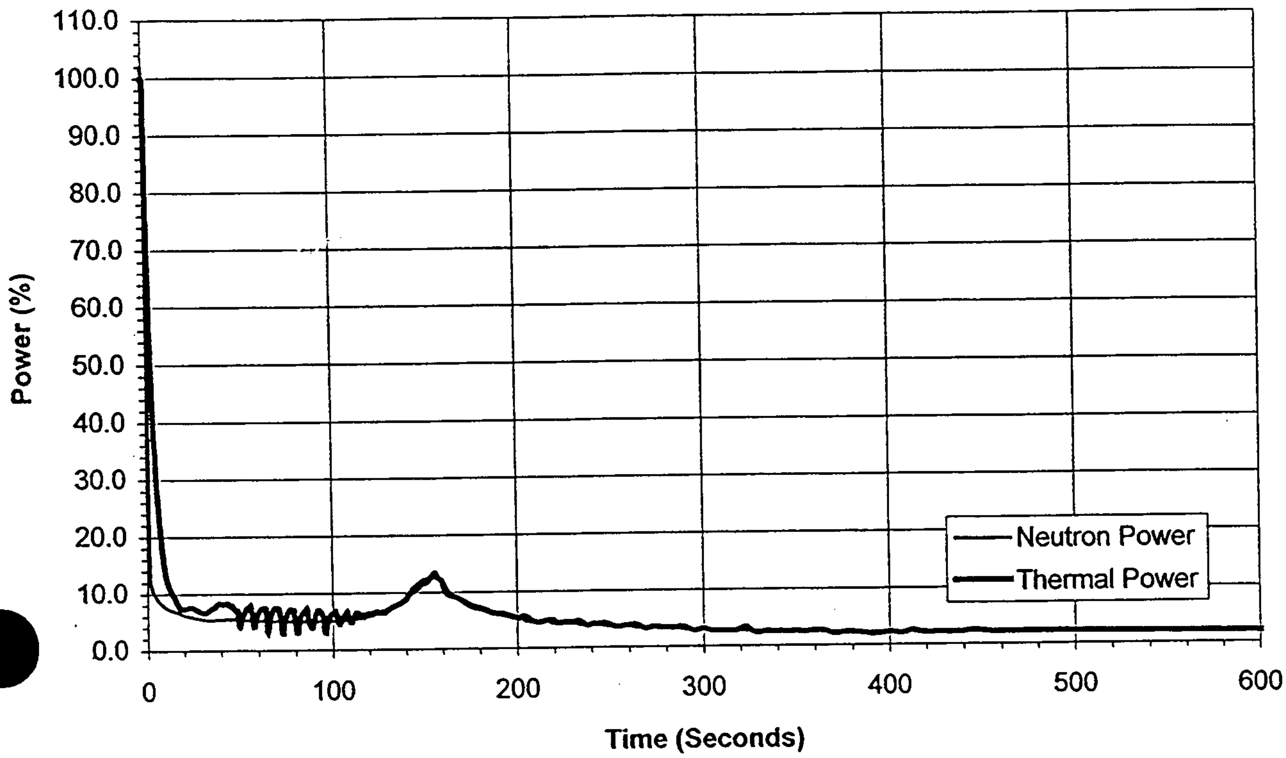


Figure 15-8

LARGE STEAM LINE BREAK

WITH OFFSITE POWER

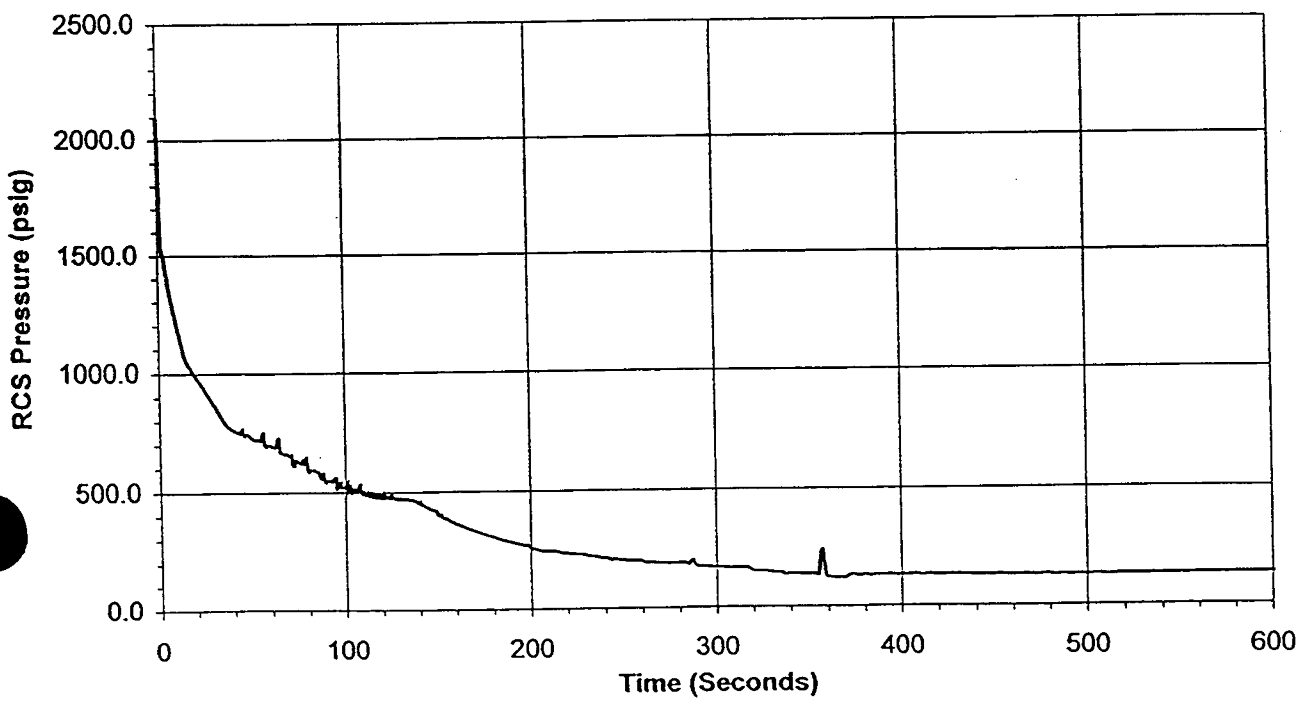
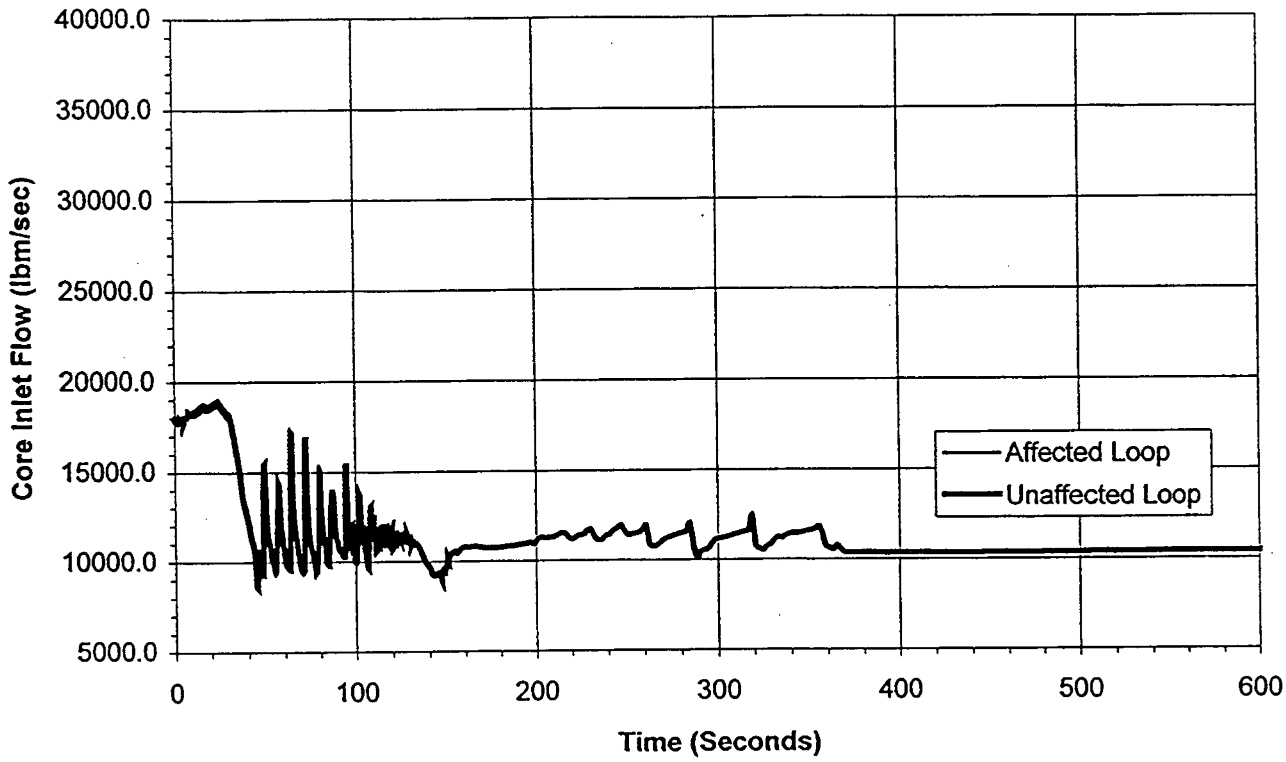


Figure 15-9
LARGE STEAM LINE BREAK
WITH OFFSITE POWER



Radial Assembly Power Distribution

**	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	**
A						0.124	0.180	0.193	0.202	0.150						A
B				0.201	0.249	0.290	0.521	0.327	0.597	0.368	0.347	0.300				B
C			0.284	0.617	0.413	0.681	0.404	0.610	0.480	0.903	0.593	0.942	0.452			C
D		0.201	0.616	0.460	0.765	0.664	0.662	0.351	0.817	0.915	1.140	0.735	1.027	0.346		D
E		0.249	0.413	0.765	0.470	0.667	0.354	0.425	0.451	0.969	0.758	1.323	0.745	0.465		E
F	0.124	0.290	0.682	0.665	0.667	0.295	0.443	0.291	0.592	0.472	1.192	1.262	1.357	0.615	0.277	F
G	0.180	0.521	0.405	0.662	0.354	0.443	0.253	0.340	0.369	0.805	0.720	1.431	0.928	1.248	0.443	G
H	0.194	0.327	0.611	0.351	0.425	0.291	0.340	0.247	0.552	0.622	1.061	0.948	1.692	0.920	0.547	H
K	0.202	0.598	0.480	0.818	0.452	0.593	0.369	0.551	0.638	1.434	1.347	2.651	1.588	1.950	0.650	K
L	0.150	0.368	0.904	0.916	0.971	0.472	0.805	0.622	1.436	1.232	3.215	3.400	3.439	1.384	0.538	L
M		0.348	0.595	1.142	0.759	1.193	0.721	1.062	1.348	3.218	2.824	5.266	2.825	1.596		M
N		0.300	0.944	0.736	1.324	1.263	1.433	0.948	2.652	3.402	5.270	5.505	5.369	1.585		N
O			0.453	1.028	0.745	1.359	0.929	1.693	1.589	3.441	2.827	5.374	2.547			O
P				0.346	0.466	0.616	1.248	0.921	1.953	1.385	1.597	1.587				P
R						0.277	0.442	0.548	0.652	0.539						R
**	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	**

Peak Pin Power Distribution

**	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	**
A						0.259	0.294	0.310	0.330	0.306						A
B				0.360	0.386	0.527	0.610	0.504	0.725	0.652	0.532	0.538				B
C			0.484	0.737	0.683	0.797	0.609	0.673	0.782	1.093	0.961	1.122	0.771			C
D		0.360	0.736	0.614	0.847	0.723	0.767	0.518	1.002	1.012	1.256	0.985	1.230	0.619		D
E		0.386	0.683	0.847	0.646	0.780	0.582	0.498	0.779	1.133	1.132	1.491	1.263	0.747		E
F	0.259	0.527	0.798	0.723	0.780	0.452	0.503	0.429	0.705	0.728	1.427	1.389	1.564	1.146	0.596	F
G	0.293	0.611	0.609	0.768	0.582	0.502	0.362	0.400	0.575	0.919	1.140	1.586	1.257	1.481	0.731	G
H	0.316	0.506	0.674	0.516	0.499	0.427	0.394	0.427	0.797	1.076	1.439	1.560	1.981	1.524	0.943	H
K	0.331	0.727	0.783	1.003	0.780	0.707	0.573	0.785	1.182	2.010	2.596	3.438	2.752	2.541	1.075	K
L	0.307	0.654	1.095	1.014	1.136	0.728	0.920	1.070	2.014	2.219	4.301	4.022	4.513	2.474	1.047	L
M		0.533	0.962	1.258	1.127	1.428	1.142	1.440	2.596	4.307	4.707	6.057	5.092	2.719		M
N		0.539	1.125	0.986	1.492	1.391	1.589	1.553	3.439	4.024	6.065	6.029	6.695	2.883		N
O			0.772	1.231	1.264	1.566	1.259	1.981	2.753	4.515	5.097	6.699	4.595			O
P				0.620	0.748	1.147	1.481	1.531	2.542	2.475	2.721	2.886				P
R						0.596	0.730	0.963	1.077	1.049						R
**	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	**

Figure 15-10
LARGE STEAM LINE BREAK
WITH OFFSITE POWER
Core Power Distribution

Figure 15-11

LARGE STEAM LINE BREAK

WITHOUT OFFSITE POWER

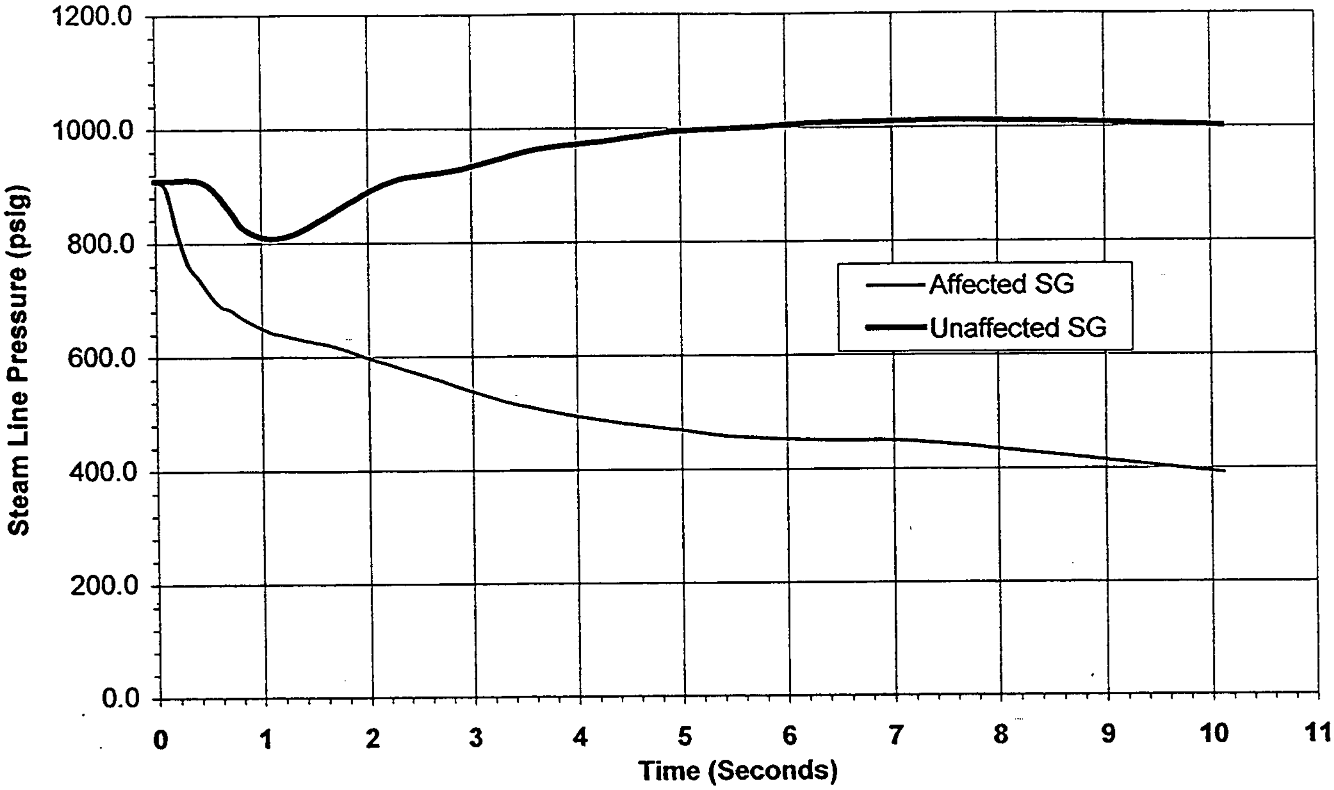


Figure 15-12

LARGE STEAM LINE BREAK

WITHOUT OFFSITE POWER

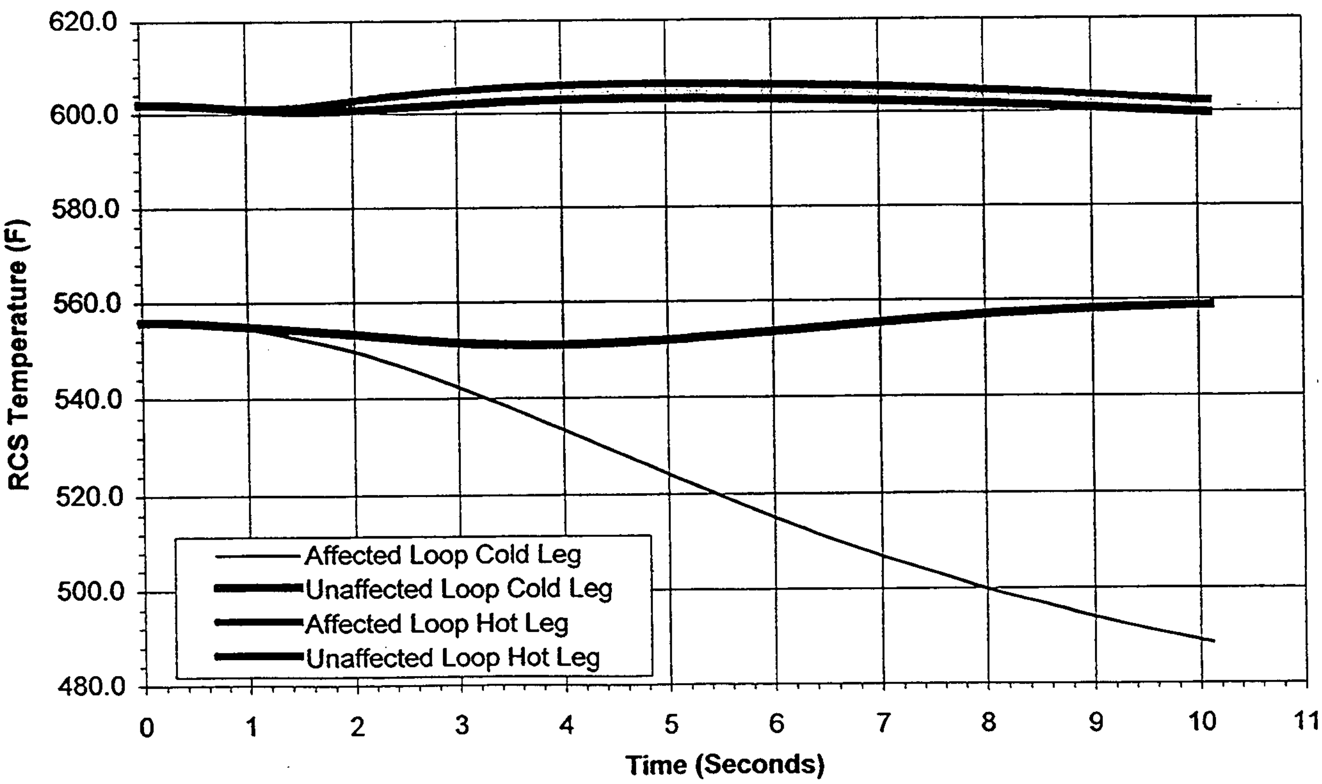


Figure 15-13
LARGE STEAM LINE BREAK
WITHOUT OFFSITE POWER

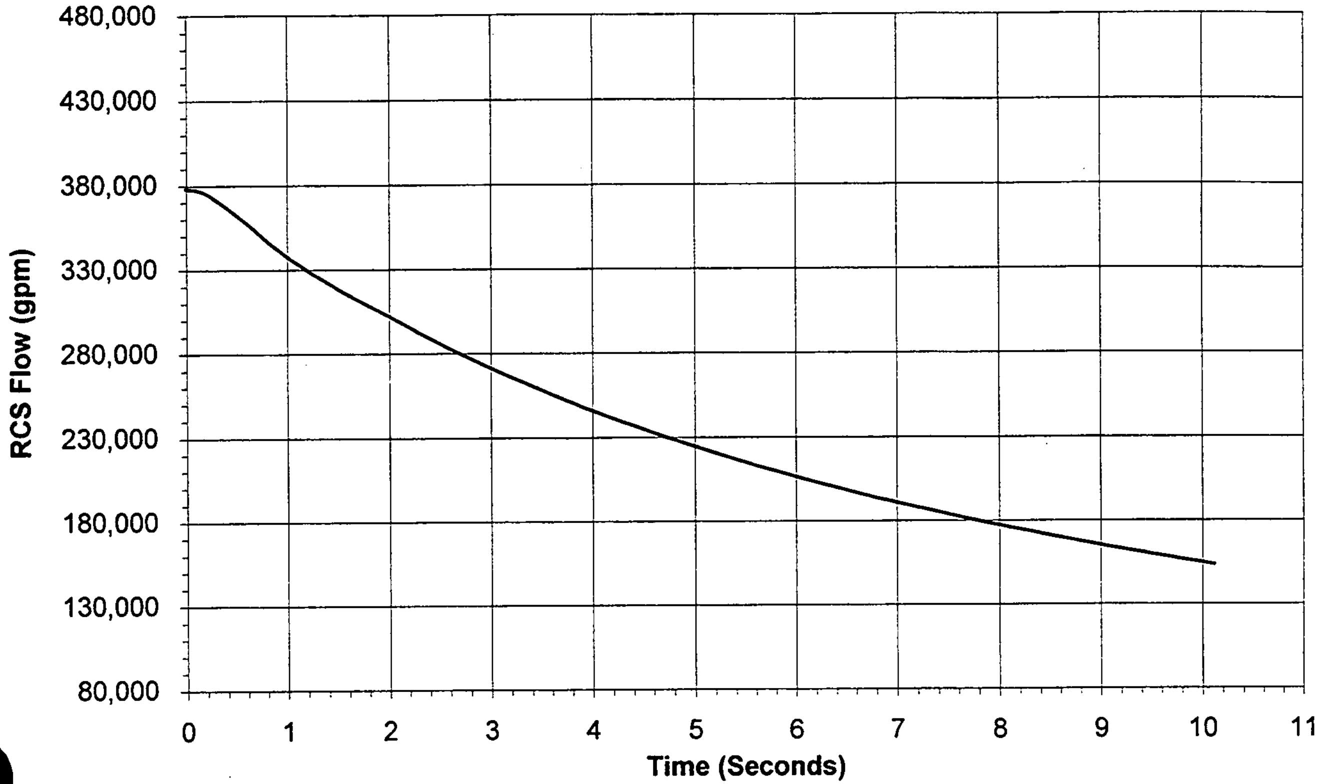


Figure 15-14
LARGE STEAM LINE BREAK
WITHOUT OFFSITE POWER

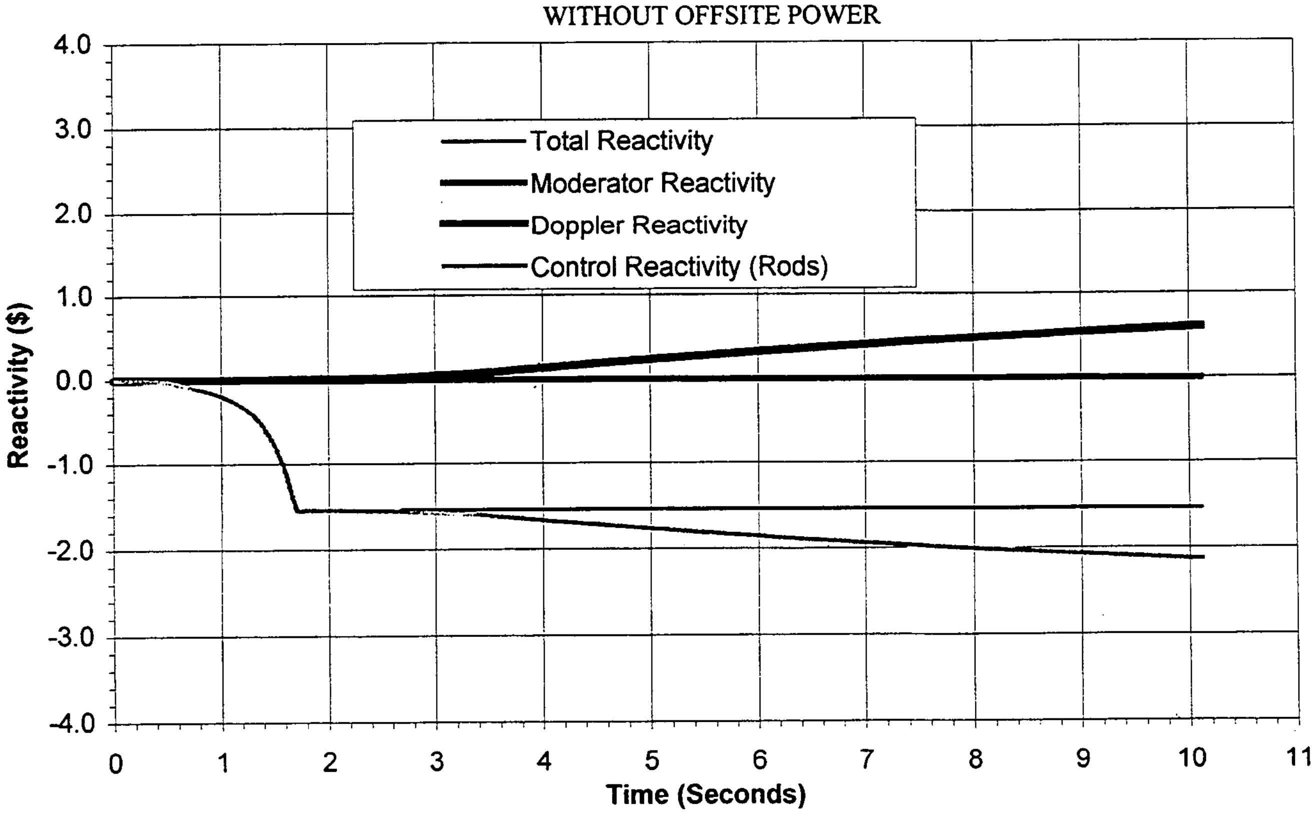


Figure 15-15

LARGE STEAM LINE BREAK

WITHOUT OFFSITE POWER

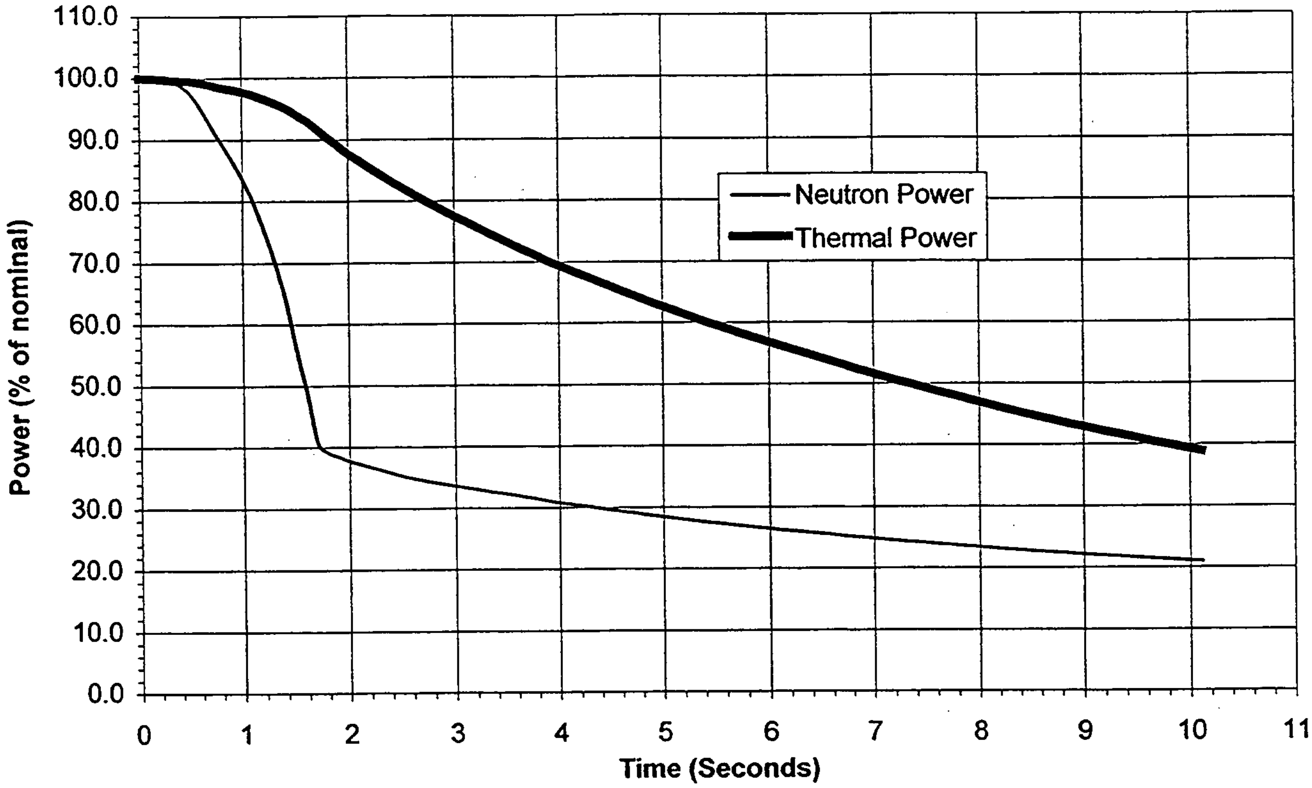


Figure 15-16

LARGE STEAM LINE BREAK

WITHOUT OFFSITE POWER

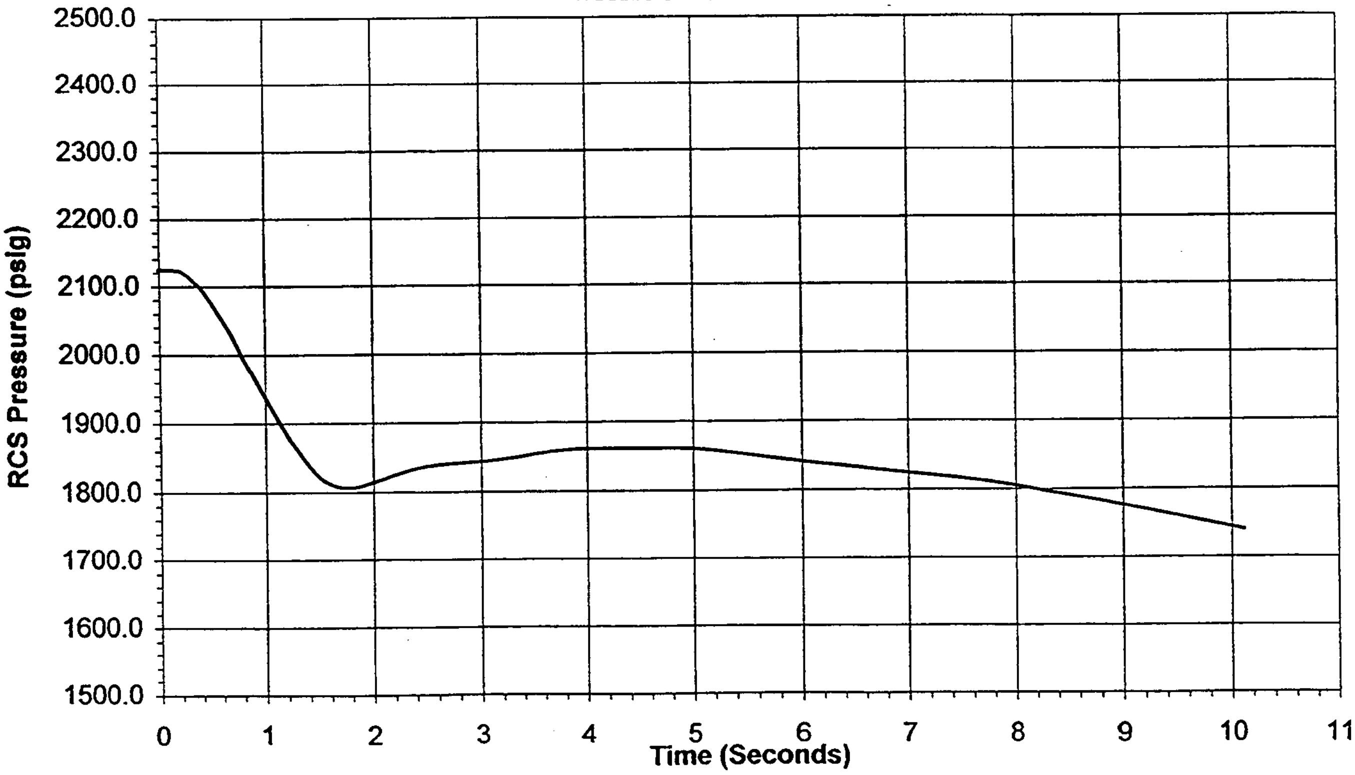


Figure 15-17
LARGE STEAM LINE BREAK
WITHOUT OFFSITE POWER

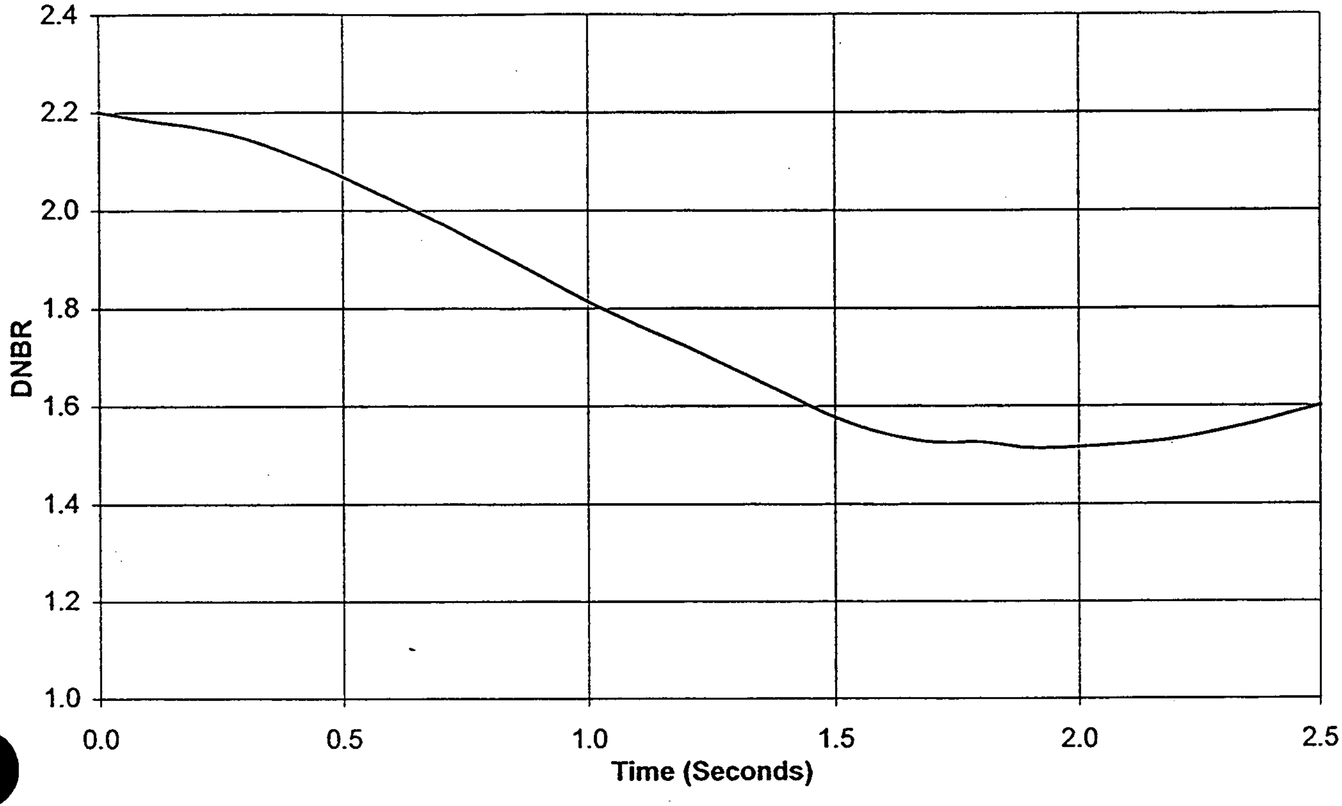


Figure 15-18
LARGE STEAM LINE BREAK
WITHOUT OFFSITE POWER

Figure 15-19

LARGE STEAM LINE BREAK

WITHOUT OFFSITE POWER

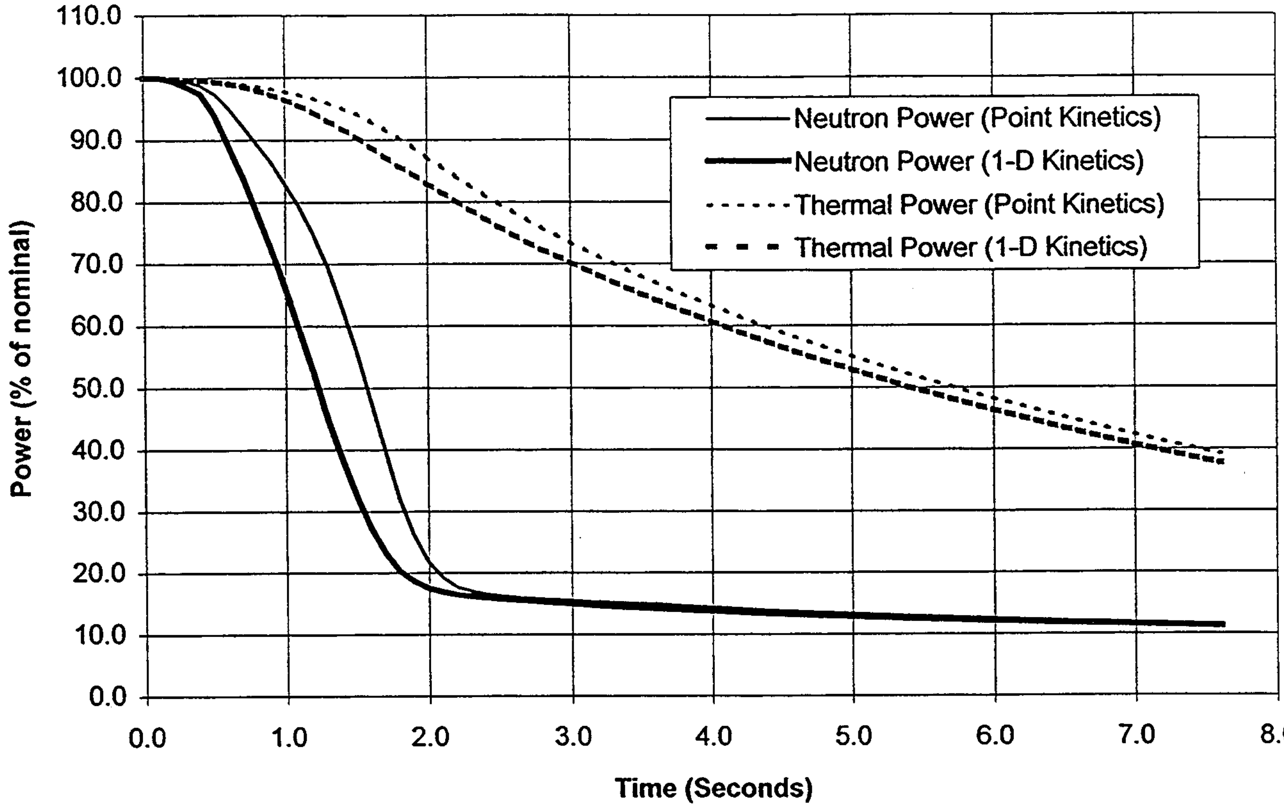
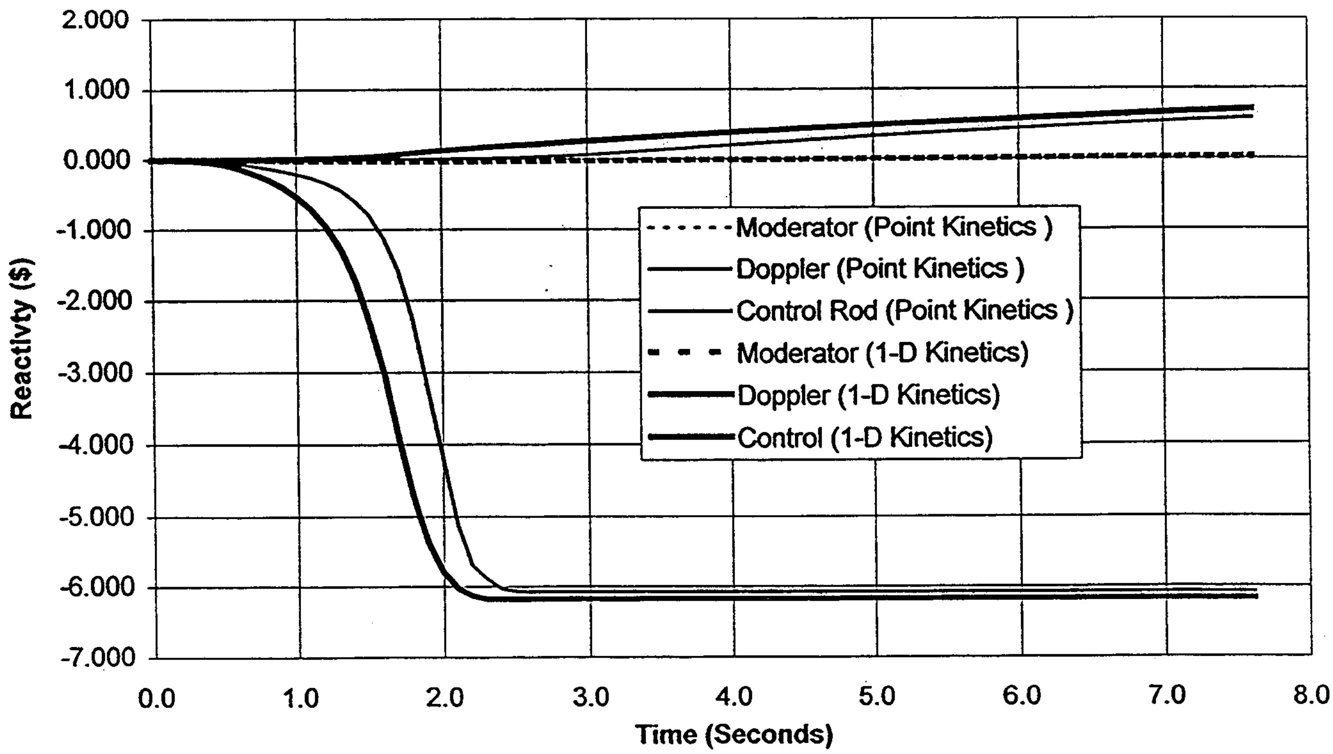


Figure 15-20

LARGE STEAM LINE BREAK

WITHOUT OFFSITE POWER



16.0 SMALL STEAM LINE BREAK

A small steam line break can be initiated by either a control system failure in which valves in the main steam line fail open or a mechanical failure of the steam line piping itself. Regardless of which type of failure occurs, the increased steam flow will cause both steam generators to depressurize until they are separated by turbine stop valve closure on turbine trip. The transient response is an overcooling event that results in an increase in power level. The system response is determined by the break size, the moderator temperature coefficient, and the Integrated Control System (ICS) assumption (manual or automatic). The most adverse combination of these conditions is analyzed to determine the worst RCS overcooling and power excursion, which should be the limiting case for DNB and centerline fuel melt (CFM). The limiting cases do not result in a reactor trip due to the reduction in reactor vessel downcomer temperature affecting the excore flux channels. Without a reactor trip a new steady-state condition at an elevated power level will result. The system response is simulated with RETRAN-02 (Reference 16-1). Both full power four-pump and part-power three-pump cases are analyzed. The RETRAN analysis provides the input for the DNB and CFM analyses.

The acceptance criteria for this analysis are that no fuel damage will occur, and that the offsite doses will be within 10% of the 10CFR100 limits. The fuel damage evaluation includes both DNB and CFM. The minimum DNBR is determined using the Statistical Core Design (SCD) methodology and the VIPRE-01 core thermal-hydraulic code (Reference 16-2).

16.1 RETRAN-02 Analysis

16.1.1 Nodalization

For the four-pump operating condition, the system response is symmetric and can be analyzed using either the single-loop or two-loop RETRAN-02 Oconee base model (Reference 16-3). The three-pump operating condition is asymmetric and requires the use of the two-loop RETRAN-02 Oconee base model. A steam chest junction is added to the two-loop base model to connect the steam lines upstream of the turbine for modeling the simultaneous depressurization of both generators prior to turbine trip. This junction closes on turbine trip when the turbine stop valves close. Two volumes are added to model the piping between the turbine control valves and the

close. [

]

16.1.2 Initial Conditions

Power Level

A high initial power level for four- and three-pump operation maximizes the primary system heat flux. The uncertainty for this parameter is incorporated in the SCD limit.

RCS Pressure

Low initial RCS pressure is conservative. The uncertainty for this parameter is incorporated in the SCD limit.

Pressurizer Level

Sensitivity cases are performed to ensure that a conservative pressurizer level is assumed for the conditions analyzed.

RCS Temperature

High initial average temperature is conservative with the uncertainty for this parameter incorporated in the SCD limit.

RCS Flow

Low initial flow is conservative. The uncertainty associated with this parameter is incorporated in the SCD limit.

Core Bypass Flow

A high core bypass flow is assumed to minimize the coolant flow along the fuel rods.

Fuel Temperature

A low initial fuel temperature will result in a higher gap conductivity which will result in a higher heat flux. Maximizing heat flux is conservative for both DNB and CFM.

Steam Generator Mass

A low initial steam generator mass is assumed to allow more cold MFW to enter the generators thereby maximizing RCS overcooling.

Steam Generator Tube Plugging

Low steam generator tube plugging is assumed to maximize primary-to-secondary heat transfer.

16.1.3 Boundary Conditions

Break Size

A range of break sizes is analyzed in combination with a range of moderator temperature coefficients to determine the most severe RCS overcooling transient which does not result in a reactor trip. This set of analyses will determine the limiting cases for DNB and CFM.

Excore Flux Detector Error Due To Overcooling

As the steam generators depressurize, the saturation temperature decreases and causes excessive primary-to-secondary heat transfer. The resulting decrease in cold leg temperature upon entering the reactor vessel downcomer attenuates the neutron flux leakage exiting the reactor. This attenuation effect reduces the flux incident on the excore neutron detectors, thereby creating an error in the indicated flux value (indicated excore detector power less than true reactor power). A conservative attenuation factor is assumed as a function of the change in reactor vessel downcomer temperature.

Pressurizer Inter-Region Heat Transfer Coefficient

Sensitivity cases are performed to ensure that a conservative inter-region heat transfer coefficient is assumed for the conditions analyzed.

Single Failure

No single failure has been identified which adversely affects this transient.

16.1.4 Physics Parameters

Moderator Temperature Coefficient

A full range of moderator temperature coefficients are considered from the BOC most positive value to the EOC most negative value.

Doppler Temperature Coefficient

The EOC least negative Doppler temperature coefficient value is assumed. This conservatively minimizes the negative reactivity feedback resulting from the fuel heatup during the power increase.

Beta-Effective and Neutron Lifetime

β_{eff} affects both the moderator and Doppler reactivity feedback during the transient. However, since the moderator temperature coefficient assumption is variable, β_{eff} is conservatively chosen to minimize the Doppler feedback. Thus, an EOC maximum value is assumed. The prompt neutron lifetime associated with the maximum β_{eff} value is assumed. The delayed neutron fractions and decay constants are insensitive parameters. Therefore, typical EOC values are used.

Scram Curve and Worth

Since the critical point of this transient occurs during steady-state operation at elevated power levels, the scram curve and worth are unimportant.

16.1.5 Control, Protection, and Safeguards Systems

Reactor Control

The reactor control subsystem of the ICS is assumed to be in manual. A power increase would result in the ICS inserting rods to maintain the initial power level, so the ICS is not credited.

Reactor Trip

All RPS trip functions are credited during this transient to identify which analyses do not result in a reactor trip. Conservative trip delay times are assumed. A penalty for a reduction in the excore flux signal due to a decrease in the reactor vessel downcomer temperature is modeled.

RCS Pressure Control

Generally, minimizing RCS pressure is conservative for DNB. However, sensitivity cases on pressurizer heaters and spray are performed to ensure that a conservative result is obtained for the conditions analyzed.

Pressurizer Level Control

Sensitivity cases are performed on net makeup/letdown to ensure that a conservative result is obtained for the conditions analyzed.

Main Feedwater System

The main feedwater subsystem of the ICS is assumed to be in manual control. As the SGs depressurize the feedwater flowrate increases. Also, as steam is lost out the break, there is less steam flow to the turbine, less steam flow to the feedwater heaters, and a reduction in feedwater temperature. These assumptions are conservative for maximizing the overcooling of the RCS and the increase in power level.

Turbine Control

The turbine control subsystem of the ICS is assumed to be in manual. As the steam generators depressurize, the turbine control valves would normally close to raise steam line pressure back to the controlling setpoint. Since it is conservative to maximize the depressurization of the steam generators, the initial turbine control valve position is unchanged for the duration of the transient.

Main Steam Line Break Detection and Main Feedwater Isolation Instrumentation

This instrumentation is not credited in the analysis.

16.2 VIPRE-01 Analysis

The forcing functions necessary to perform the DNB analysis (core average heat flux, core inlet flow and temperature, core exit pressure) are obtained from the RETRAN-02 analysis results and input to VIPRE-01. The VIPRE-01[ ]channel model (Reference 16-3) is then used to determine the time of the minimum DNBR statepoint for the transient conditions analyzed. At these statepoint conditions a set of maximum allowable radial peak (MARP) curves is developed for determining if the DNBR limit is exceeded.

16.3 Results

The peak power levels predicted by RETRAN are 126% for the full power initial condition, and 113% for the three-pump initial condition. These power levels are in excess of the RPS high flux and flux/flow/imbalance trip setpoints due to the attenuation of the flux signal by the cooldown of the reactor vessel downcomer water. The results of the analyses show that the core power peaking and core thermal-hydraulic conditions at these power levels will not exceed the DNB or CFM limits.

16.4 Reload Cycle-Specific Evaluation

Physics parameters that are checked for each reload core include the following:

- Moderator temperature coefficient
- Doppler temperature coefficient
- Maximum allowable radial peak limits
- Centerline fuel melt limits

16.5 References

16-1 RETRAN-02: A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, EPRI NP-1850-CCM, Revision 4, EPRI, November 1988

16-2 VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM-A, Revision 3, EPRI, August 1989

16-3 Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000, Duke Power Company, July 1987

DPC-NE-3005-A
Revision 1

List of Attached Docketed Correspondence

1. 7/30/97 original submittal letter, M. S. Tuckman to NRC
2. 4/13/98 NRC RAI letter, D. E. LaBarge to M. S. Tuckman
3. 7/23/98 letter responding to NRC RAI, M. S. Tuckman to NRC
4. 8/28/98 NRC letter acknowledging proprietary information
5. 2/1/99 letter submitting Revision 1, M. S. Tuckman to NRC
6. 3/17/99 NRC RAI letter on Revision 1, D. E. LaBarge to M. S. Tuckman
7. 4/19/99 letter responding to NRC RAI, M. S. Tuckman to NRC
8. 5/5/99 letter with minor text revision, M. S. Tuckman to NRC

DPC-NE-3005-A, Revision 1
August 1999

The following is the list of changes in converting
Revisions 0 and 1, versions, to the Revision 1 –A version

Item	Page	Description
1.	front	New title page with –PA and dated August 1999
2.		Inserted all Revision 1 pages without margin bars
3.	front	Inserted the Revision 0 and Revision 1 SERs
4.	5-8	The following text: "*However, preliminary analyses indicate that the reactor trip functions that rely on an indicated flux signal will not occur. This is because of the large control rod shadowing effect and the large nuclear instrumentation calibration errors potentially present at HZP. Therefore, only the high pressure trip function is credited in the analysis.*" is replaced with "*Control rod shadowing and nuclear instrumentation calibration errors present at HZP are accounted for.*" This change is based on additional experience analyzing the startup accident, and corrects the original text. The new information is that the reactor can trip on indicated flux, not just on high pressure. The methodology is not changed.
5.	8-5	Change "less" to "greater" to correct obvious incorrect sentence regarding the DNBR limit.
6.	12-1	Change "trust" to "thrust" to correct mispelling
7.	16-1	Revised acceptance criteria wording per 5/5/99 letter
8.	back	Inserted list of and copies of all documented correspondence
9.	back	Inserted this list of changes
10.	back	Inserted list of changes from Revision 0 to Revision 1 that was submitted to the NRC in the 2/1/99 letter

DPC-NE-3005, Revision 1
January 1999

The following is a list of changes which are included in Revision 1

Item	Page	Description
1.	iii	Editorial: Replaced "DNB" with "Core Cooling Capability"
2.	v	Deleted Section 10.3.4 Offsite Dose Analysis Results
3.	vi	Deleted Section 14.4.1.3 Offsite Dose Analysis Results, renumbered 14.4.1.4 to 14.4.1.3
4.	vi	Section 15.2.1.4 added
5.	vii	Section 15.3.1 revised and Section 15.3.2 subsection details deleted
6.	xi	Figures 15-1 and 15-10 added, other figures renumbered
7.	1-1	Added references for Revisions 1 and 2 to DPC-NE-3000 and related SERs
8.	1-4	Added references for SERs for Revisions 1 and 2 to DPC-NE-3000
9.	1-6	Added references for SERs for Revisions 1 and 2 to DPC-NE-3000
10.	1-6	Added a statement identifying a new steam line break VIPRE model
11.	1-7	Deleted credit for main feedwater isolation in steam line break analyses
12.	1-8	Revised to note that the digital ICS has been installed
13.	1-9	Reference 1-6 updated to Revision 3
14.	1-10	Reference 1-19 updated to Revision 3
15.	1-11	Reference 1-30 deleted (related to main feedwater isolation)
16.	1-11	References 30, 31, and 32 added for submittal letters and SERs related to DPC-NE-3000 Revisions 1 and 2
17.	2-11	Revised description of the VIPRE-01/MOD2 Duke internal version used from "MOD2F" to "a Duke version".
18.	2-11	Added BWU-N CHF correlation to note that this version of the BWU CHF correlation is used in the lower part of the Mk-B11 fuel assemblies below the mixing vane grids
19.	2-12	Added a reference to Section 15.2.2 for the VIPRE steam line break model
20.	2-16	Reference 2-7 and Reference 2-8 updated to Revision 3
21	3-10	Add small steam line break to Table 3-1
22.	4-7	Updated HPI setpoints and related footnotes to current values
23.	5-3	Editorial: Replaced "DNB" with "Core Cooling Capability"
24.	5-8	Added the maximum reactivity insertion rate of 11.5 pcm/sec
25.	5-10	Reference 5-2 updated to Revision 3
26.	6-3	Deleted "and Worth" which is incorrect
27.	6-7	Deleted "and Worth" which is incorrect
28.	6-11	Reference 6-2 updated to Revision 3
29.	10-1	Revised to state that peak RCS pressure is not a concern
30.	10-1	Revised to include 110% of RCS design pressure as an acceptance criterion for the locked rotor accident

(cont.)

(List of Changes (cont.))

31.	10-11	Deleted Section 10.3.4 Offsite Dose Analysis Results
32.	11-1	Added centerline fuel melt as a fuel damage criterion for the dropped rod accident
33.	11-2	Revised to note that the digital ICS has been installed
34.	11-2	Added centerline fuel melt as a fuel damage criterion
35.	11-6	Added results of centerline fuel melt analysis
36.	11-7	Reference 11-2 updated to Revision 3
37.	12-7	Reference 12-3 updated to Revision 3
38.	13-4	Revised "maximize ECCS injection" to "initiate flow from the High Pressure Injection (HPI) System"
39.	13-4	Deleted operator action to manually trip the reactor for SGTR
40.	13-5	Revised to state that the cooldown to 450°F occurs after shift changeover is completed (rather than after one RCP per loop is tripped off)
41.	13-5	Revised to include steam generator draining as an operator action to be consistent with the preceeding text
42.	13-6	Revised to state that the Reactor Protective System is assumed to trip the reactor at 20 minutes (rather than manual trip)
43.	13-7	Deleted statement regarding the results of the offsite dose analysis
44.	14-19	Deleted Section 14.4.1.3 Offsite Dose Analysis Results, renumbered Section 14.4.1.4
45.	14-23	Reference 14-5 updated to Revision 3
46.	14-25	Added a footnote to Table 14-1
47.	14-25	Corrected some rod ejection analysis results in Table 14-2
48.	14-26	Corrected some rod ejection anlaysis results in Table 14-3
49.	14-27	Corrected some rod ejection analysis results in Table 14-5 and added some details and footnotes for clarity
50.	15-1 to -43	Large steam line break with offsite power reanalyzed to not credit main feedwater isolation. New VIPRE model for DNBR analysis described. The entire steam line break section including unrevised pages is included for continuity. The offsite dose analysis results are deleted.
51.	16-1 to –7	Small steam line break reanalyzed to not credit main feedwater isolation. The offsite dose acceptance criterion was revised to 10% of 10 CFR Part 100. The offsite dose analysis results are deleted.

Duke Power
A Duke Energy Company

M. S. Tuckman
Executive Vice President
Nuclear Generation

Duke Power Company
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EC07H
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Charlotte, NC 28201-1006

(704) 382-2200 OFFICE
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July 30, 1997

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Oconee Nuclear Station
Docket Numbers 50-269, -270, and -287
UFSAR Chapter 15 Transient Analysis Methodology,
DPC-NE-3005-P

Gentlemen:

Please find enclosed proprietary topical report DPC-NE-3005-P, "UFSAR Chapter 15 Transient Analysis Methodology," which is submitted for NRC review and approval. A non-proprietary version is also enclosed. This report describes the Duke Energy Corporation (at the time the report was written, Duke Power Company) methodology for analyzing the non-LOCA UFSAR Chapter 15 transients and accidents for the Oconee Nuclear Station.

The objective of this report is to implement a modern non-LOCA transient and accident analysis methodology to enable a complete revision to the 1970s vintage analyses in Chapter 15 of the Oconee UFSAR. Reanalysis using this methodology will establish a new licensing basis which will significantly improve the knowledge of the design requirements, assumptions, and inputs used in the analyses. As a result, the quality and thoroughness of safety evaluations conducted in support of operations and resolution of regulatory issues will be enhanced. Most of the computer codes and simulation models used in this topical report have been previously reviewed and approved by the NRC for application to Oconee reload design and in response to Generic Letter 83-11. This topical report is a specific application of those models to the analysis of Oconee UFSAR Chapter 15 transients and accidents. Duke Power previously received NRC approval for a similar methodology that is applicable to the McGuire and Catawba Nuclear Stations.

In addition to the above objective, a new fuel assembly design designated as the Mk-B11 will be phased into core reload designs beginning with Oconee Unit 3 Cycle 19, with startup scheduled for March 2000. This new fuel design will incorporate mixing vane grids and an improved CHF correlation. These changes require reanalysis of many of the UFSAR Chapter 15 events.

To support the schedule for fabrication and design of the fuel for Oconee Unit 3 Cycle 19, review of this topical report is requested by October 1998.

In accordance with 10CFR 2.790, Duke Power Company requests that this report be considered proprietary. Information supporting this request is included in the attached affidavit.

If there are any questions, or additional information is needed, please call Scott Gewehr at (704) 382-7581.

M. S. Tuckman

M. S. Tuckman

cc: Mr. D. E. LaBarge, Project Manager
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Mr. L. A. Reyes, Regional Administrator
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M. A. Scott
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Oconee Nuclear Station

bxc (w/o enclosures):
G. A. Copp
G. B. Swindlehurst
J. E. Burchfield (ONS)

(w/ Enclosure 1)
ELL

AFFIDAVIT OF M. S. TUCKMAN

1. I am Executive Vice President, Nuclear Generation Department, Duke Energy Corporation ("Duke"), and as such have the responsibility of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear plant licensing, and am authorized to apply for its withholding on behalf of Duke.

2 I am making this affidavit in conformance with the provisions of 10 CFR 2.790 of the regulations of the Nuclear Regulatory Commission ("NRC") and in conjunction with Duke's application for withholding which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.

4. Pursuant to the provisions of paragraph (b)(4) of 10 CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

    (i) The information sought to be withheld from public disclosure is owned by Duke and has been held in confidence by Duke and its consultants.

    (ii) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs, relative to a method of analysis that provides a competitive advantage to Duke.

    (iii) The information was transmitted to the NRC in confidence and under the provisions of 10 CFR 2.790, it is to be received in confidence by the NRC.

    (iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

M. S. Tuckman

M. S. Tuckman

(continued)

(v) The proprietary information sought to be withheld in this submittal is that which is marked in the proprietary version of the report DPC-NE-3005-P, "UFSAR Chapter 15 Transient Analysis Methodology" and supporting documentation, and omitted from the non-proprietary versions. This information enables Duke to:

(a) Respond to Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions."

(b) Simulate UFSAR Chapter 15 transients and accidents for the Oconee Nuclear Station and other PWRs.

(c) Perform safety reviews per 10 CFR 50.59.

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke.

(a) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

(b) Duke intends to sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke.

M. S. Tuckman

M. S. Tuckman

(continued)

5. Public disclosure of this information is likely to cause harm to Duke because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring a commensurate expense or allowing Duke to recoup a portion of its expenditures or benefit from the sale of the information.

M. S. Tuckman, being duly sworn, on his oath deposes and says that he is the person who subscribed his name to the foregoing statement, and that the matters and facts set forth in the statement are true.

M. S. Tuckman

M. S. Tuckman

Sworn to and subscribed before me this 30th day of July, 1997. Witness my hand and official seal.

Linda Case Smith

Notary Public

My commission expires 5-6-2000.

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

April 13, 1998

Mr. M. S. Tuckman
Executive Vice President
Nuclear Generation
Duke Energy Corporation
P. O. Box 1006
Charlotte, NC 28201-1006

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - REVIEW OF OCONEE NUCLEAR STATION UNITS 1, 2, AND 3 UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) CHAPTER 15 TRANSIENT ANALYSIS METHODOLOGY, DPC-NE-3005-P (TAC NOS. M99349, M99350, AND M99351)

Dear Mr. Tuckman:

By letter dated July 30, 1997, Duke Energy Corporation (DEC) submitted Topical Report DPC-NE-3005-P, "UFSAR Chapter 15 Transient Analysis Methodology," for NRC review and approval. It describes the DEC methodology for analyzing the non-Loss-of-Coolant Accident UFSAR Chapter 15 transients and accidents for the Oconee Nuclear Station.

In order to continue its review of the document, the staff has determined that additional information is needed as described in the enclosure.

Sincerely,

David E. LaBarge, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosure: As stated

cc w/encl: See next page

Oconee Nuclear Station

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REQUEST FOR ADDITIONAL INFORMATION
TOPICAL REPORT DPC-NE-3005-P

1. Please explain why there are two options available for input of the nodal axial power profile to the computer code VIPRE-01.

    a) What differences result between the use of a linear interpolation compared to a spline fit?

    b) How does the user determine which option to use?

2. When will the power hot channel factor and the local heat flux hot channel factor be input to VIPRE-01 to calculate the departure from nucleate boiling ratio (DNBR) in a subchannel? How will this cause the results of a DNBR calculation to differ from what is currently obtained?

3. The Babcock and Wilcox (B&W) Correlation (BWC) and the BWC mixing vane critical heat flux correlations have been previously approved for use in VIPRE-01. Why are these considered additional features to be added to VIPRE-01/MOD2F?

4. Briefly describe the enhanced iteration logic used by VIPRE-01 to converge to a minimum DNBR (MDNBR) limit.

5. The moderator (boron) dilution accident for Oconee is analyzed only from conditions of power operation (Mode 1) and refueling (Mode 6). Provide analyses for Modes 2 through 5 and demonstrate that the results conform to those specified in Standard Review Plan 15.4.6.

6. The acceptance criterion for the Oconee analysis of the rod ejection accident is that the offsite dose will be less than 100 percent of the 10 CFR Part 100 limits. However, NRC Regulatory Guide 1.77 and Standard Review Plan 15.4.8 specify that calculated doses should be well within 10 CFR Part 100 limits, where "well within" is defined as 25 percent of the 10 CFR Part 100 exposure guideline values. Please modify the Oconee dose acceptance criterion accordingly.

7. Section 1.3 - Describe the Emergency Operating Procedures (EOPs) available at Oconee that will manually start the Emergency Feedwater (EFW) System to backup the nonsafety grade equipment in mitigating the loss of reactor coolant flow transient. Confirm that the operator actions could be taken in time to bound the results of the analysis.

8. Section 1.3 - It is stated that for certain failures in the safety grade EFW System, credit is taken for realigning EFW flow through the nonsafety Main Feedwater System and this design aspect has been reviewed and approved by the NRC. Provide the documentation for this issue.

Enclosure

9. Section 1.3 - One of the turbine trip circuitry channels has a slower response time than the value assumed in the analysis methodology. Confirm that the required modification will be completed prior to the approval of the proposed methodology.

10. Section 1.3 - Confirm that the use of the nonsafety grade turbine trip circuitry in the transient analysis is consistent with the Oconee current licensing basis.

11. Section 1.3 - It is stated that the capability to remotely throttle certain nonsafety grade valves (including the steam generator drain lines) is credited in the analysis methodology. Identify the areas that are affected by this assumption and confirm that they are within the Oconee current licensing basis.

12. Section 2.2 - It is indicated that the advanced solution scheme and correlations of the RETRAN-3D computer code are used in the proposed analysis methodology. Provide further discussions on how the proposed methodology could be approved without a detailed review of the RETRAN-3D code.

13. Section 9.3 - Assuming a single failure of the pump monitor trip, will Cases 3 and 5 become more limiting than Cases 2 and 4 due to a reactor trip from flux/flow?

14. Section 9.3.1.4 - Discuss the assumed delay time of the reactor trip.

15. Section 10 (Locked Rotor) - Provide a revised analysis methodology to incorporate the following:

    a) Assume a loss of offsite power with this event.

    b) Include peak system pressure as a part of the acceptance criteria for this event.

    c) Since a flux/flow is the trip function for this event, a single failure of the pump monitor trip becomes a nonlimiting failure. Identify the most limiting single failure for this event.

16. Section 12.0 (Turbine Trip) - It is indicated that no credit is taken for EFW flow in this event since the peak reactor coolant system (RCS) pressure will occur prior to the EFW actuation. The staff does not agree with this approach. We will require an analysis that shows there will not be a second peak pressure higher than the first peak during this transient. With an insufficient EFW flow rate, the RCS pressure could become the problem later in the transient. Also, the concern of the solid pressurizer should be addressed during the longer-term with respect to EFW flow.

17. Section 13.0 (Steam Generator Tube Rupture (SGTR)) - Provide the results of a revised analysis methodology to incorporate the following:

a) Assume a loss of offsite power with this event.

b) Assume a stuck-open atmospheric dump valve to maximize the radiological consequences.

18. Section 13.0 - Discuss the consequences of the SGTR event assuming the nonsafety grade pressurizer heater and spray systems become inoperable.

19. Section 13.0 - Discuss the EOPs available at Oconee that affect the following:

    a) Operator isolation of the ruptured steam generator following the SGTR event.

    b) Prevention of steam generator overfill, assuming the maximum EFW flow rate.

20. Section 15.0 (Large Steamline Break (SLB)) - Provide discussion in the following areas:

    a) Why is the SLB with loss of offsite power very similar to a loss of RCS flow event. Should an SLB with rapid RCS cooldown lead to a more severe DNBR transient?

    b) Should a low initial pressurizer level lead to a lower transient pressure and lower DNBR?

    c) If a single failure of the EFW control valve is assumed, will a second MDNBR occur later into the transient due to further cooldown of the RCS?

21. Section 16.0 (Small SLB) - Provide discussion in the following areas:

    a) Explain why the acceptance criteria allow fuel failure and the offsite dose within 100 percent of 10 CFR Part 100 limits for a small SLB (including an inadvertent opening of a main steam relief valve), which is an incident of moderate frequency.

    b) Discuss the consequences of the event assuming failure of the nonsafety grade main feedwater system and EFW is needed.

Duke Power
A Duke Energy Company

M. S. Tuckman
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July 23, 1998

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555-0001
Attention: Document Control Desk

Subject: Duke Energy Corporation

Oconee Nuclear Station, Units 1, 2 and 3
Docket Numbers 50-269, 50-270, and 50-287

Response to NRC Request for Additional Information on Topical Report DPC-NE-3005-P, "UFSAR Chapter 15 Transient Analysis Methodology."

This submittal contains information that Duke Energy Corporation considers PROPRIETARY and is being made pursuant to 10CFR 2.790.

By letter dated April 13, 1998 the NRC requested additional information on Topical Report DPC-NE-3005P, "UFSAR Chapter 15 Transient Analysis Methodology." This topical report had been previously submitted for NRC review by Duke letter dated July 30, 1997.

The questions contained in the April 13 NRC letter, and the corresponding Duke answers, are provided in Attachment 1 to this letter.

Additionally, Attachment 2 provides editorial corrections and minor revisions to Topical Report DPC-NE-3005-P. These changes to this topical report have been identified since the July 30, 1997 submittal of this document. Attachment 3 provides the non-proprietary version of these changes.

Some of the information contained in Attachment 2 is considered proprietary. In accordance with 10CFR 2.790, Duke requests that this information be withheld from public disclosure. An affidavit which attests to the proprietary nature of the affected information is included with this letter.

In the response to Question 9 (see Attachment 1), Duke provides the schedule for implementing station modifications to correct slow response time associated with the turbine trip circuitry channels. Oconee has preliminarily scheduled the turbine stop valve closure circuitry modification beginning with the Unit 3 End-of-cycle 18 outage, currently scheduled for March 2000. Implementation for Unit 1 would be in September 2000, and Unit 2 would be in April 2001. If the modification process alters the preliminary schedule, Duke will notify the NRC of this change by the end of 1997.

Please address any comments or questions regarding this matter to J. S. Warren at (704) 382-4986.

Very truly yours,

M. S. Tuckman

M. S. Tuckman

Attachments

xc:

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M. A. Scott
NRC Senior Resident Inspector
Oconee Nuclear Station

AFFIDAVIT

1. I am Executive Vice President of Duke Energy Corporation; and as such have the responsibility for reviewing information sought to be withheld from public disclosure in connection with nuclear power plant licensing; and am authorized on the part of said Corporation (Duke) to apply for this withholding.

2. I am making this affidavit in conformance with the provisions of 10CFR 2.790 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke's application for withholding, which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.

4. Pursuant to the provisions of paragraph (b)(4) of 10CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned by Duke and has been held in confidence by Duke and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs relative to a method of analysis that provides a competitive advantage to Duke.

M. S. Tuck

M. S. Tuckman

(Continued)

(iii) The information was transmitted to the NRC in confidence and under the provisions of 10CFR 2.790, it is to be received in confidence by the NRC.

(iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is marked in Attachment 2 to Duke Energy Corporation letter dated July 23, 1998; SUBJECT: Response to NRC Request for Additional Information on Topical Report DPC-NE-3005P, "UFSAR Chapter 15 Transient Analysis Methodology." This information enables Duke to:

(a) Respond to NRC requests for information regarding transient response of Babcock & Wilcox PWRs.

(b) Simulate UFSAR Chapter 15 transients and accidents for Oconee Nuclear Station.

(c) Perform safety evaluations per 10CFR50.59.

(d) Support Facility Operating Licenses/Technical Specifications amendments for Oconee Nuclear Station.

M. S. Tuck

M. S. Tuckman

(Continued)

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke.

(a) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

(b) Duke intends to sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke.

5. Public disclosure of this information is likely to cause harm to Duke because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring commensurate expense or allowing Duke to recoup a portion of its expenditures or benefit from the sale of the information.

M. S. Tuckman

M. S. Tuckman

(Continued)

U. S. Nuclear Regulatory Commission
July 23, 1998
Page 6

M. S. Tuckman, being duly sworn, states that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth within are true and correct to the best of his knowledge.

M. S. Tuckman

M. S. Tuckman, Executive Vice President

Subscribed and sworn to before me this 23RD day of

July, 1998

Mary P. Nelms
Notary Public

My Commission Expires:
JAN 22, 2001

SEAL

bxc:

L. A. Keller
J. E. Burchfield
G. B. Swindlehurst
J. E. Smith (Commitment Included in Response to Question 9)
ELL

Attachment 1

Question 1
Please explain why there are two options available for input of the nodal axial power profile to the computer code VIPRE-01.

a) What differences result between the use of a linear interpolation compared to a spline fit?
b) How does the user determine which option to use?

Response to Question 1

Originally, the spline fit was developed to replace the linear interpolation routine because it was recognized that linear interpolation did not conserve area under the curve (straight line from point to point), and would therefore under-predict the axial shape uniformly (non-conservative for DNB predictions). Furthermore, the linear interpolation routine would grossly under-normalize unless the user specifically input data at the endpoints (if not, linear interpolation would default to zero values at the endpoints). After developing the spline fit, Duke recognized the need to be able to duplicate previous analyses that used linear interpolation; so, instead of replacing linear interpolation with the spline fit, an additional option was added.

Like linear interpolation, the spline fit also has limitations at the endpoints. The spline fit conserves area under the curve at all locations except the endpoints, for which it makes an estimate. For this reason, the spline fit also does not normalize to 1.0 unless the user specifies endpoints. Sensitivity studies have shown that, unlike linear interpolation, specifying endpoints to force the spline fit to normalize to 1.0 has negligible effect on DNBR predictions, since the spline fit makes an estimate at the endpoints (as opposed to defaulting to a zero value as with the linear interpolation routine). Generally, bottom peaked spline fit shapes slightly under-normalize, and top peaked spline fit shapes slightly over-normalize. When comparing the two options directly, the spline fit (without specifying endpoints that force normalization to 1.0) will usually give an equal or slightly more conservative DNBR prediction than the linear interpolation routine (with or without specifying endpoints that force it to normalize to 1.0).

If the user is duplicating analyses that have previously been performed with linear interpolation, then the linear

interpolation option should be selected for consistency. If the user is performing an analysis for which axial shape input must be generated for a large number of axial levels, the spline fit should be selected for the reasons stated above.

Question 2

When will the power hot channel factor and the local heat flux hot channel factor be input to VIPRE-01 to calculate the departure from nucleate boiling ratio (DNBR) in a subchannel? How will this cause the results of a DNBR calculation to differ from what is currently obtained?

Response to Question 2

The local heat flux hot channel factor has not been applied in Oconee DNBR analyses beginning with the Oconee Unit 1 Cycle 14 reload (Reference the Oconee 1 Cycle 14 Reload Report, Letter, M. S. Tuckman (Duke) to NRC Document Control Desk, May 7, 1991)

The power hot channel factor is input to VIPRE-01 to calculate the DNBR when the DNBR analysis accounts for uncertainties directly, i.e. when the Statistical Core Design (SCD) methodology is not used. The rod ejection DNB analysis is the only transient DNB analysis in DPC-NE-3005-P not employing the SCD methodology. This is due to the core power level exceeding the range of the SCD methodology during the rod ejection analysis.

The power hot channel factor has been applied during the entire operating history of Oconee. The application of the power hot channel factor is not new to the Oconee licensing basis, therefore there is no difference being introduced in the proposed methodology.

Question 3

The Babcock and Wilcox (B&W) Correlation (BWC) and the BWC mixing vane critical heat flux correlations have been previously approved for use in VIPRE-01. Why are these considered additional features to be added to VIPRE-01/MOD2F?

Response to Question 3

Section 2.3.1 of DPC-NE-3005-P shows a summary of additional features and editorial changes existing in the Duke VIPRE-01/MOD2F code version relative to the standard EPRI VIPRE-01/MOD2 version. With the exception of adding the BWU-Z CHF correlation, all other features and changes listed in Section 2.3.1 were added and made in the previous Duke versions (MOD2A to MOD2E). Therefore, the BWC and BWC mixing vane critical heat flux correlations are not considered as additional features to be added to VIPRE-01/MOD2F. They are listed only to distinguish from the standard EPRI code version.

Question 4

Briefly describe the enhanced iteration logic used by VIPRE-01 to converge to a minimum DNBR (MDNBR) limit.

Response to Question 4

The VIPRE-01 code uses the secant method to perform the option to iterate on a specified parameter to an MDNBR limit. The logic in the original VIPRE-01/MOD2 code did not always cause the iteration to converge when the input parameter yielded a MDNBR value significantly different from the target MDNBR limit. Logic was therefore added such that if the MDNBR resulting from the parameter input value is different from the target MDNBR limit by a value of 0.75, the first guess at the iterated parameter would increase the input value by 10% instead of the 1% as in the original code version.

Additionally, each iteration of the specified parameter is used to calculate a MDNBR in the secant method. The calculated MDNBR is then compared to the target MDNBR limit to generate a delta-difference MDNBR value. The next guess of the parameter value is based on this delta-difference MDNBR value. In the original code version, if the next iteration yielded the same delta-difference MDNBR value, the logic would not increase the parameter value and the iteration process would stall. This problem is circumvented by increasing the specified parameter value by an amount of 0.01 when the delta-difference MDNBR does not change between iterations.

Question 5

The moderator (boron) dilution accident for Oconee is analyzed only from conditions of power operation (Mode 1) and refueling (Mode 6). Provide analyses for Modes 2 through 5 and demonstrate that the results conform to those specified in SRP 15.4.6.

Response to Question 5

The analysis methodology and acceptance criteria that were utilized in the original UFSAR Chapter 15 analyses constitute, in part, the design basis of the plant on which the issuance of the operating license was based. Oconee's operating license was issued prior to the Standard Review Plan. Therefore, the Standard Review Plan is not applicable to Oconee. Duke is not proposing to adopt the Standard Review Plan guidelines for including moderator dilution accident analyses for Modes 2 through 5. Duke is upgrading the Modes 1 and 6 analyses to be more representative of modern analyses.

Question 6

The acceptance criterion for the Oconee analysis of the rod ejection accident is that the offsite dose will be less than 100 percent of the 10 CFR Part 100 limits. However, NRC Regulatory Guide 1.77 and Standard Review Plan 15.4.8 specify that calculated doses should be well within 10 CFR Part 100 limits, where "well within" is defined as 25 percent of the 10 CFR Part 100 exposure guidelines values. Please modify the Oconee dose acceptance criterion accordingly.

Response to Question 6

The analysis methodology and acceptance criteria that were utilized in the original UFSAR Chapter 15 analyses constitute, in part, the design basis of the plant on which the issuance of the operating license was based. Oconee's operating license was issued prior to the Standard Review Plan. Therefore, the Standard Review Plan is not applicable to Oconee. Duke is not proposing to adopt the Regulatory Guide 1.77 or the Standard Review Plan guidelines for offsite dose acceptance criteria being limited to a fraction of the 10 CFR Part 100 limits. The Part 100 dose acceptance criteria were the basis for the issuance of the Oconee operating license, and were accepted by the staff at that

time. Duke's methodology will continue to comply with the original licensing basis.

Question 7

Section 1.3 - Describe the Emergency Operating Procedures (EOPs) available at Oconee that will manually start the Emergency Feedwater (EFW) System to back up the non-safety grade equipment in mitigating the loss of reactor coolant flow transient. Confirm that the operator actions could be taken in time to bound the results of the analysis.

Response to Question 7

The EFW System is a three pump/two train safety-grade system designed to automatically start and feed both steam generators on low main feedwater pump turbine hydraulic oil pressure or low steam generator level. Either of these start signals infers that a loss of the non-safety grade main feedwater system has occurred. The multiple pump and redundant train design of the EFW System ensures that at least one steam generator will automatically be fed following this event. EFW flow will be automatically controlled to the appropriate minimum level (any RCPs on) or the natural circulation level (all RCPs off).

The discussion in Section 1.3 (Credit for Control Systems and Non-Safety Components and Systems) describes a scenario where 1) a loss of all RCPs occurs, which requires that the SG level be raised to the natural circulation setpoint, 2) the MFW System continues to operate, 3) the non-safety Integrated Control System (ICS) fails to raise the SG level from the post-trip minimum level to the natural circulation level setpoint, and 4) the EFW autostart setpoints are not reached and EFW does not actuate. For this scenario operator action is required to identify the ICS failure and to respond by increasing SG levels to the natural circulation setpoint with either MFW or EFW. In the Oconee EOP, Step 5.3 (the third step) in the Subsequent Actions section directs the operator to "Throttle Main or Emergency FDW as required to control SG level(s) and RCS temperature". Based on simulator experience, this step will be reached in less than 5 minutes for this scenario. EOP Step 15.12 specifies "IF no RCPs are operating . . . Verify SG levels approaching 50% OR" (50% OR is the natural circulation setpoint). It is expected that this step will be reached in 10 minutes for this scenario. These steps clearly guide the operator to address the ICS failure of concern. Increasing SG levels beginning at this time is very sufficient for

establishing conditions for natural circulation and long-term decay heat removal.

The duration of the loss of flow analysis is very short and focuses on the approach to the DNBR limit as a result of the loss of some or all RCPs. MFW and EFW are not relied on to demonstrate that the DNBR acceptance criterion is met. Therefore, there is no explicit modeling of the MFW or EFW Systems in the analysis other than the use of MFW to establish the secondary heat sink at the initial conditions.

Question 8

Section 1.3 - It is stated that for certain failures in the safety grade EFW System, credit is taken for realigning EFW flow through the non-safety Main Feedwater System and this design aspect has been reviewed and approved by the NRC. Provide the documentation for this issue.

Response to Question 8

The documentation for this issue is the following references:

1. Letter, William O. Parker (Duke), to Harold R. Denton (NRC), December 21, 1979, Attachment 2 - Emergency Feedwater System Reliability Analysis for the Oconee Nuclear Generating Station. Refer to Section 2.1.3 on p. 6 for a docketed description of the EFW realignment.

2. Letter, William O. Parker (Duke), to Harold R. Denton (NRC), July 23, 1980, Attachment 2 - Emergency Feedwater System. Refer to Section 2.1.3 on p. 2-2 for a docketed description of the EFW realignment.

3. Letter, William O. Parker (Duke), to Harold R. Denton (NRC), April 3, 1981, Response to NRC RAI Dated 11/14/80 - Auxiliary Feedwater System Reliability Evaluation. Refer to the response to Question 14.

4. Letter, John F. Stolz (NRC) , to William O. Parker (Duke), August 25, 1981, SER for NUREG-0737 Item II.E.1.1, "Auxiliary Feedwater System Evaluation" (Refer to Item 6 on p. 18).

Question 9

Section 1.3 - One of the turbine trip circuitry channels has a slower response time than the value assumed in the analysis methodology. Confirm that the required modification will be completed prior to the approval of the proposed methodology.

Response to Question 9

Oconee has preliminarily scheduled the turbine stop valve closure circuitry modification for implementation beginning with the Oconee Unit 3 end-of-cycle 18 outage, currently scheduled for March 2000. Implementation for Unit 1 would be in September 2000, and Unit 2 would be in April 2001 based on their respective refueling outages. The modification process will finalize the scope and schedule for this modification by the end of 1998. If the modification process alters the preliminary schedule, Duke will notify the staff of this change by the end of the year.

Considering that the current design has been the licensing basis since 1973, the above implementation plan, which is an enhancement to the redundancy of the turbine stop valve actuation circuitry, is considered sufficient. This is supported by the excellent reliability data for the first actuation channel, which has been credited in the current UFSAR licensing basis analyses.

Implementation of the proposed methodology is planned beginning with the startup of Oconee Unit 1 Cycle 18 in June 1999. This is desired in order to enable upgrading the UFSAR Chapter 15 analyses to establish a new baseline for safety reviews and UFSAR verification activities in progress. With the proposed implementation plan, this represents essentially one year of operation with the proposed analysis methodology in place prior to the upgrade of the turbine stop valve circuitry. NRC approval of this implementation plan is requested.

Question 10

Section 1.3 - Confirm that the use of the non-safety grade turbine trip circuitry in the transient analysis is consistent with the Oconee licensing basis.

Response to Question 10

The UFSAR Chapter 15 analyses include an immediate turbine trip following all reactor trip. This assumption is typical of UFSAR Chapter 15 analyses, including other PWRs such as Westinghouse plants. A failure of the turbine to trip is considered as an initiating event that causes an overcooling event that is bounded by the steam line break analysis. Since this assumption has been in the Oconee UFSAR since the operating license was issued, it is the licensing basis assumption.

Question 11

Section 1-3 - It is stated that the capability to remotely throttle certain non-safety grade valves (including the steam generator drain lines) is credited in the analysis methodology. Identify the areas that are affected by this assumption and confirm that they are within the Oconee current licensing basis.

Response to Question 11

The non-safety grade valves credited in the analysis methodology are the atmospheric dump valves (ADVs), the turbine bypass valves (TBVs), valves in the startup feedwater flow path, the pressurizer spray valve, and valves in the steam generator drain lines. The TBVs are remotely controlled, while the remainder of these valves are manual-local controlled valves.

The ADVs and TBVs are credited in the proposed SGTR analysis to both cool the unit down to DHR system conditions and to control level in the ruptured SG. The TBVs are considered part of the current Oconee licensing basis since they are credited in the current Ch. 15 SGTR analysis for cooling the plant down to DHR System conditions. The ADVs are not considered part of the current licensing basis with regard to UFSAR Ch. 15.

The proposed SGTR analysis methodology credits bypassing a stuck-closed EFW control valve via the alternate Main Feedwater System flowpath. The use of this alternate flowpath is part of the current licensing basis (References: 1) Letter, William O. Parker (Duke), to Harold R. Denton (NRC), December 21, 1979, Attachment 2 - Emergency Feedwater System Reliability Analysis for the Oconee Nuclear Generating Station. Refer to Section 2.1.3 on p. 6 for a docketed description of the EFW realignment; 2) Letter,

John F. Stolz (NRC) , to William O. Parker (Duke), August 25, 1981, SER for NUREG-0737 Item II.E.1.1, "Auxiliary Feedwater System Evaluation" (Refer to Item 6 on p. 18)).

The pressurizer spray valve is assumed to be operable in the proposed SGTR analysis methodology. The use of the spray valve during a SGTR event is part of the current licensing basis (References: 1) Letter, M. S. Tuckman (Duke), to NRC Document Control Desk, March 27, 1991, G. L. 90-06; 2) Letter, L. A. Wiens (NRC), to J. W. Hampton (Duke), June 9, 1994, SER regarding G. L. 90-06)

The SG drain lines are credited in the proposed SGTR analysis methodology to control level in the ruptured SG when steaming of this SG is no longer effective in accomplishing this action (i.e., when SG pressure is very low). The use of these drain lines is not currently within the licensing basis for ONS.

Question 12

Section 2.2 - It is indicated that the advanced solution scheme and correlations of the RETRAN-3D computer code are used in the proposed analysis methodology. Provide further discussions on how the proposed methodology could be approved without a detailed review of the RETRAN-3D code.

Response to Question 12

DPC-NE-3005-P includes several comparison analyses which demonstrate similar transient results for RETRAN-02 and RETRAN-3D. In these analyses RETRAN-3D is used in the "RETRAN-02 mode", which refers to applying the code without any of the significant new RETRAN-3D models (three-dimensional core kinetics, non-equilibrium field equations, non-condensible gas flow). The new solution method (faster execution time) and some improved correlations in RETRAN-3D are the scope of the application of RETRAN-3D that is proposed in the topical report. The intent of the submittal is to demonstrate by direct comparison of analysis results that the improved RETRAN-3D code gives essentially identical results as RETRAN-02. Comparing a new code version (RETRAN-3D) to an approved code version (RETRAN-02) is a logical process for validating the new code, and has precedent in the industry.

RETRAN-3D is maintained by EPRI as a quality-assured code under EPRI's Appendix B program. RETRAN-3D has been submitted for NRC review and approval (Letter, G. B.

Swindlehurst (Duke Power on behalf of RETRAN Maintenance Group) to T. E. Collins (NRC), July 8, 1998). The details of RETRAN-3D, including the new solution method and upgraded correlations are in this reference. The proposed approach was intended as a first step in the NRC approval process for RETRAN-3D.

Implementation of the DPC-NE-3005-P methodology does not require NRC-approval of the RETRAN-3D scope of the submittal.

Question 13

Section 9.3 - Assuming a single failure of the pump monitor trip, will Cases 3 and 5 become more limiting than Cases 2 and 4 due to a reactor trip from flux/flow?

Response to Question 13

Case 3 is a loss of four RCPs from an initial condition that assumes four RCPs are operating. Case 5 is a loss of three RCPs from an initial condition that assumes three RCPs are operating. The pump monitor trip is designed to trip the reactor when two or more RCPs trip at power levels greater than 2% rated thermal power. Assuming a single failure in the pump monitor trip function results in a reactor trip when three or more RCPs trip. For both Cases 3 and 5, a single failure in the pump monitor trip will not prevent a pump monitor trip since the combination of the number of pumps tripped and/or not operating is greater than or equal to three. Therefore, the single failure of concern has no effect on Cases 3 and 5. Analyses have shown that Cases 2 and 4 result in more severe transient results than Cases 3 and 5.

Question 14

Section 9.3.1.4 - Discuss the assumed delay time of the reactor trip.

Response to Question 14

The conservative trip delay time referred to in Section 9.3.1.4 actually refers to a conservative trip delay time for each trip. Thus, the trip delay time assumed for the pump monitor trip function conservatively bounds the calculated pump monitor trip time delay, while the trip delay time assumed for the flux/flow/imbalance trip function

conservatively bounds the calculated flux/flow/imbalance trip time delay. The delay times are given in Table 4-2.

Question 15

Section 10 (Locked Rotor) - Provide a revised analysis methodology to incorporate the following:

a) Assume a loss of offsite power with this event

b) Include peak system pressure as a part of the acceptance criteria for this event

c) Since a flux/flow is the trip function for this event, a single failure of the pump monitor trip becomes a non-limiting failure. Identify the most limiting single failure for this event.

Response to Question 15

a) The assumption of a loss of offsite power concurrent with the locked rotor event is not part of Oconee's current licensing basis. The operating license was issued based on the current assumption. Duke is not proposing to change this assumption.

b) Duke will include a peak Reactor Coolant System pressure acceptance criterion of 110% of design pressure for the locked rotor accident. The results will be included in the UFSAR.

c) For locked rotor, the case initiated from three RCPs in operation is the limiting case in terms of approach to DNB. Since the pump monitor trip normally actuates when two or more pumps are in a tripped condition, it is normally expected that a pump monitor trip would be actuated for this case. By assuming a single failure in the pump monitor trip function, the three RCP case becomes limiting due to the longer trip time delay associated with the flux/flow/imbalance trip. For the four RCPs in operation initial condition, no single failure could be identified which would affect the analysis results due to the short duration of the event.

Question 16

Section 12.0 (Turbine Trip) - It is indicated that no credit is taken for EFW flow in this event since the peak reactor coolant system (RCS) pressure will occur prior to the EFW actuation. The staff does not agree with this approach. We will require an analysis that shows there will not be a second peak pressure higher than the first peak during this transient. With an insufficient EFW flow rate, the RCS pressure could become the problem later in the transient. Also, the concern of the solid pressurizer should be addressed during the longer-term with respect to EFW flow.

Response to Question 16

The analysis methodology for the turbine trip event was intended to quantify the short-term Reactor Coolant System peak pressure, which occurs at 7.5 seconds, shortly after the reactor trips. The results show that the pressurizer code safety valves are not challenged during this short-term pressure peak following a turbine trip event, and therefore the turbine trip is non-limiting relative to other events that do challenge the code safety valves. The turbine trip event does not require an assumption of a loss of the Main Feedwater System. That would constitute two initiating events. On p. 12-5 of the topical report, it states, "Therefore, main feedwater is isolated on turbine trip to maximize the primary system pressure ." This statement refers to the post-trip runback of Main Feedwater to control to the minimum steam generator level setpoint. To reach this setpoint, main feedwater flow is "isolated" until the level setpoint is reached, and then flow is restored to the steam generator. Since the analysis is of short duration and the low level setpoint is not reached, the main feedwater flow remains isolated, With continued availability of the Main Feedwater System, the high flow capacity ensures that the long-term peak pressure will be bounded by those events for which the MFW System is not available. The focus of the analysis was on the short-term pressure peak, with the result being that this event did not even lift the pressurizer code safety relief valves. With the Main Feedwater System still available to provide an abundant heat sink, the turbine trip event is clearly non-limiting in the long-term relative to other events in which only the small capacity Emergency Feedwater System is available. For this reason, an extended analysis of the turbine trip is not necessary. The report will be revised (see Attachment 2) to state that the pressure peak is bounded by other events which challenge the code safety valves, and that the long-term response with continued main

feedwater availability is also bounded by events with only the Emergency Feedwater System in operation to provide a secondary heat sink. The Oconee UFSAR does not include an acceptance criterion related to overfilling the pressurizer. The current acceptance criteria are the basis on which the operating license was issued. Duke is not proposing adding this new acceptance criterion.

Question 17

Section 13.0 (Steam Generator Tube Rupture (SGTR)) - Provide the results of a revised analysis methodology to incorporate the following:

a) Assume a loss of offsite power with this event
b) Assume a stuck-open atmospheric dump valve to maximize the radiological consequences.

Response to Question 17

a) The proposed SGTR analysis methodology is based on the current licensing basis which does not assume a loss of offsite power during this event. Duke is not proposing to change the current licensing basis in regard to this assumption.
b) The atmospheric dump valves at Oconee are manual-local valves. Oconee's single failure philosophy was developed prior to the issuance of most of the present day industry standards, and is thus only partially based on the present day standard. Only specific systems are considered for the single failure criterion, and of these systems, a single failure in the EFW system will result in the highest offsite radiological releases during a SGTR event. A failure of a manually-opened valve, such as the atmospheric dump valve, is not in the current licensing basis and is therefore not considered in the proposed SGTR analysis methodology. Duke is not proposing to change the current licensing basis in regard to this assumption.

Question 18

Section 13.0 - Discuss the consequences of the SGTR event assuming the non-safety grade pressurizer heater and spray systems become inoperable.

Response to Question 18

Assuming that the pressurizer heaters are inoperable during an SGTR event will result in a slower repressurization of the RCS following a reactor trip. This will reduce the primary-to-secondary leakage during this portion of the transient, resulting in lower offsite radiological releases. It is therefore conservative to assume that the pressurizer heaters are operable.

During the course of this transient, operators will minimize the RCS subcooled margin to minimize primary-to-secondary leakage. There are three methods that can be used to depressurize the RCS: 1) normal pressurizer spray, 2) auxiliary pressurizer spray, and 3) manual actuation of the pressurizer PORV. The availability of normal pressurizer spray during an SGTR event has been reviewed and approved by the NRC (References: 1) Letter, M. S. Tuckman (Duke), to NRC Document Control Desk, March 27, 1991, G. L. 90-06; 2) Letter, L. A. Wiens (NRC), to J. W. Hampton (Duke), June 9, 1994, SER regarding G. L. 90-06). The least effective method in depressurizing the RCS is normal pressurizer spray. The use of either auxiliary pressurizer spray or the pressurizer PORV will result in a more rapid minimization of the subcooled margin. This will reduce primary-to-secondary leakage during this portion of the transient and result in lower offsite radiological releases. Therefore, the least effective method of depressurizing the RCS, the pressurizer spray valve, is credited.

Question 19

Section 13.0 - Discuss the EOPs available at Oconee that affect the following:

a) Operator isolation of the ruptured steam generator following the SGTR event.

b) Prevention of steam generator overfill, assuming the maximum EFW flow rate.

Response to Question 19

a) The EOPs at Oconee will have operators diagnose that a tube leak has occurred in either the Immediate Manual Actions section or the Subsequent Actions section, depending on whether or not the tube leak has resulted in a reactor trip. This diagnosis will require entry into the SG tube leak section of the EOP, which provides the

guidance to mitigate this event. The EOP requires that the RCS be cooled down to a temperature $\leq$ 532°F. After this has been completed the EOP provides guidance to isolate the SG with the tube leak. Isolation refers to stopping both steaming and feeding of the SG with the tube leak. Due to the design of the OTSGs, the tube leak flow can only be minimized (not stopped) as the plant is cooled down and depressurized. Steaming will subsequently be used to prevent SG overfill and to prevent exceeding the tube-to-shell ΔT limits, as necessary. After completion of these initial isolation steps, the EOP refers the operators to the procedure "Control of Secondary Contamination" which will complete the isolation of the SG with the tube leak. This procedure addresses the isolation of drain lines and other possible flowpaths from the secondary side of the SG with the tube leak.

b) EFW is automatically controlled by the safety-grade EFW Control System to a low minimum level setpoint (any RCPs on) once actuated. Manual operator control of EFW to prevent SG overfill is not required, but is included as a backup action in procedures should the control system fail. Due to the SG tube leakage slowly filling the SG, the SG level rapidly exceeds the minimum level setpoint, and EFW is automatically isolated. SG overfill is then a result solely of the continuing SG tube leakage. SG overfill is mitigated by steaming and draining once the overfilled level setpoint is reached.

Question 20

Section 15.0 (Large Steamline Break (SLB)) - Provide discussion in the following areas:

a) Why is the SLB with loss of offsite power very similar to a loss of RCS flow event? Should a SLB with rapid RCS cooldown lead to a more severe DNBR transient?

Response to Question 20 (a)

The SLB with loss of offsite power (LOOP) is very similar to a loss of RCS flow event because in both events the reactor coolant pumps (RCPs) lose power early in the transient. Upon loss of power, the RCPs coast down, resulting in a decrease in core flow. Decreased core flow when reactor power is still high is a DNB concern.

For the SLB with coincident LOOP, the LOOP is assumed to result in the loss of power to the control rod drives, resulting in control rod insertion. In a loss of flow analysis, the initiating event that results in all RCPs coasting down is not assumed to cause control rod insertion. Control rod insertion relies on a reactor trip signal initiated by the Reactor Protection System sometime after the RCPs have actually begun to coast down.

In a loss of flow accident, the RCS pressure remains fairly constant or increases due to the heatup. In the SLB, the cooldown results in the RCS pressure decreasing rapidly, which is a DNB penalty. Due to a finite loop transport time, the SLB induced cooldown of the RCS inventory in the tube bundle does not reach the core before the minimum DNB ratio occurs. Thus, the SLB induced cooldown does not influence the MDNBR during the time period of interest (core inlet temperature remains fairly constant). However, the pressure reduction is immediately sensed in the core region. The combined effect of a flow coastdown and depressurization is a DNB concern.

In summary, the loss of flow event has a delayed insertion of the control rods, and the SLB/LOOP event has a much lower pressure. Both of these effects are DNB penalties, and the limiting event can only be determined by analysis.

b) Should low initial pressurizer level lead to a lower transient pressure and lower DNBR?

<u>Response to Question 20 (b)</u>

For the SLB with offsite power maintained, minimizing the volume of relatively hot pressurizer inventory that drains into the hot leg is conservative with respect to maximizing the cooldown of the core inlet temperature. With a negative moderator temperature coefficient, a greater RCS cooldown maximizes the chances of a post-trip return to power. Sensitivity cases have investigated a high initial pressurizer level coupled with a high initial RCS pressure assumption. The results of these sensitivities indicate that a low initial level with a low initial pressure result in a more severe challenge to DNB.

For the SLB with coincident LOOP, sensitivity cases have been run assuming both high and low initial pressurizer levels. As discussed in the response above, the core inlet temperature does not change before the MDNBR occurs. Thus, system temperature effects associated with varying

pressurizer levels do not affect the DNB results. The RCS pressure response has also been shown to be approximately the same regardless of the level assumption during the time period of interest. Thus, the initial pressurizer level assumption is not significant.

c) If a single failure of the EFW control valve is assumed, will a second MDNBR occur later in the transient due to further cooldown of the RCS?

Response to Question 20 (c)

A second MDNBR will not occur later in the transient since there is no return-to-power. Once the High Pressure Injection System (HPI) is actuated and injecting borated water, subcriticality is assured. Boron injection from HPI occurs approximately 85 seconds into the event. The core flood tanks may also inject borated water to prevent a return-to-criticality. The negative reactivity resulting from control rod insertion on reactor trip is sufficient to prevent a return to power in this time period. SLB sensitivity cases have been extended to 10 minutes and confirmed no return to criticality will occur. At 10 minutes operator action to isolate the EFW flow to the affected steam generator is credited to terminate the cooldown.

Question 21

Section 16.0 (Small SLB) - Provide discussion in the following areas:

a) Explain why the acceptance criteria allow fuel failure and the offsite dose within 100 percent of the 10 CFR Part 100 limits for a small steam line break (including an inadvertent opening of a main steam relief valve), which is an incident of moderate frequency.

b) Discuss the consequences of the event assuming failure of the non-safety grade main feedwater system and EFW is needed.

Response to Question 21

a) The offsite dose acceptance criteria in the Oconee UFSAR are the 10 CFR Part 100 dose limits. These were accepted by the NRC when Oconee's operating license was issued. The Standard Review Plan approach of restricting offsite doses

for some events to a fraction of the Part 100 limits is not applicable to Oconee. The analyses performed using the DPC-NE-3005-P methodology have demonstrated that no fuel failure will occur for all small steam line breaks, and therefore offsite doses will not be limiting regardless of the acceptance criteria.

b) The small SLB transient is an overpower event that relies on a continued high main feedwater flowrate, well in excess of the full power flowrate, to provide a heat sink that can maintain the core overpower condition without resulting in a reactor trip. If the Main Feedwater System is assumed to fail, then a large mismatch between the core power and the small EFW System heat sink capacity will develop, and the reactor will rapidly trip. In addition, assuming a loss of the Main Feedwater System along with the small steam line break constitutes two initiating events, which is not required. For the small SLB transient, it is the large capacity of the Main Feedwater System which aggravates the plant transient response and causes an approach to the DNB limits. Assuming a loss of the non-safety grade Main Feedwater System would result in a less-limiting transient.

Attachment 3

Revisions and Editorial Corrections to
DPC-NE-3005-P - July 1997

The following revisions and editorial corrections are provided for clarification purposes, and to correct minor errors that have been identified since the submittal of DPC-NE-3005-P on July 30, 1997. One of these revisions is in response to the NRC RAI letter dated April 13, 1998.

1) Page iii: Editorial correction for consistency

2) Page 4-5: Corrected values in the table for the range of initial pressurizer level.

3) Page 5-3: Editorial correction for consistency

4) Page 11-2: Added details of base model nodalization changes to the main steam line and the feedwater boundary condition.

5) Page 12-5: Revised to clarify the modeling of the MFW System.

6) Page 12-6: Revised to state that the long-term response of the turbine trip is bounded by other events.

7) Pages 16-1 and 16-2: Rewritten for clarity.

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

August 28, 1998

Mr. M. S. Tuckman
Senior Vice President
Nuclear Generation
Duke Energy Corporation
P.O. Box 1006
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SUBJECT: REQUEST FOR WITHHOLDING INFORMATION FROM PUBLIC DISCLOSURE-- OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 (TAC NOS. M99349, M99350, AND M99351)

Dear Mr. Tuckman:

By letter dated July 23, 1998, Duke Energy Corporation (Duke) responded to an NRC request for additional information on Topical Report DPC-NE-3005, "UFSAR [Updated Final Safety Analysis Report] Chapter 15 Transient Analysis Methodology," and considered the information proprietary. An affidavit dated July 23, 1998, was also included. As stated in the affidavit, the information included in the submittal enables Duke to respond to NRC requests for information regarding transient response of Babcock & Wilcox pressurized water reactors, simulate the UFSAR Chapter 15 transients and accidents for the Oconee Nuclear Station (ONS), perform safety evaluations in accordance with 10 CFR 50.59, and support Facility Operating License/Technical Specification amendments for the ONS. You requested that the information be withheld from public disclosure pursuant to 10 CFR 2.790. A nonproprietary version of the document was also included in the submittal for placement in the NRC's public document room.

You stated that the submitted information should be considered exempt from mandatory public disclosure for the following reasons:

(i) The information sought to be withheld from public disclosure is owned by Duke and has been held in confidence by Duke and its consultants.

(ii) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs relative to a method of analysis that provides a competitive advantage to Duke.

(iii) The information was transmitted to the NRC in confidence and under the provisions of 10 CFR 2.790, it is to be received in confidence by the NRC.

(iv) The information sought to be protected is not available in public to the best of [Duke's] knowledge and belief.

We have reviewed your application and the material in accordance with the requirements of 10 CFR 2.790 and, on the basis of your statements, has determined that the submitted information sought to be withheld contains proprietary commercial information.

Therefore, the version of the submitted information marked as proprietary will be withheld from the public disclosure pursuant to 10 CFR 2.790(b)(5) and Section 103(b) of the Atomic Energy Act of 1954, as amended.

Withholding from public inspection shall not affect the right, if any, of persons properly and directly concerned to inspect the documents. If the need arises, we may send copies of this information to our consultants working in this area. We will, of course, ensure that the consultants have signed the appropriate agreements for handling proprietary information.

If the basis for withholding this information from public inspection should change in the future such that the information could then be made available for public inspection, you should promptly notify the NRC. You also should understand that the NRC may have cause to review this determination in the future, for example, if the scope of a Freedom of Information Act request includes your information. In all review situations, if the NRC makes a determination adverse to the above, you will be notified in advance of any public disclosure.

Sincerely,

David E. LaBarge, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

cc: See next page

Oconee Nuclear Station

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(704) 382-2200 OFFICE
(704) 382-4360 FAX

February 1, 1999

U. S. Nuclear Regulatory Commission
Washington D. C. 20555

ATTENTION: Document Control Desk

Subject: Oconee Nuclear Station
Docket Numbers 50-269, -270, -287
UFSAR Chapter 15 Transient Analysis Methodology,
Topical Report DPC-NE-3005-P, Revision 1

References: 1) Letter, M. S. Tuckman (Duke), to NRC, July 30, 1997

2) D. E. LaBarge (NRC) to W. R. McCollum (Duke), October 1, 1998

Gentlemen:

By means of Reference 1, Duke Energy Corporation submitted Topical Report DPC-NE-3005-P which describes the new Duke methodology for analyzing the Oconee UFSAR Chapter 15 non-LOCA transients and accidents. NRC review and acceptance of this topical report were documented in a SER forwarded by Reference 2. Several issues requiring revisions to the topical report methodology were identified in Reference 2 and in a meeting with the NRC staff in Rockville on September 15, 1998. The requested revisions to the topical report have been addressed in the attached Revision 1 to DPC-NE-3005-P, dated January 1999. Several minor changes in addition to those requested by the NRC are also included in Revision 1. The following are the more significant issues addressed in Revision 1.

- The locked rotor analysis has been revised to include an acceptance criterion of 110% of design pressure per the SER for Revision 0.

- The SGTR methodology has been revised to not credit operator action for tripping the reactor per the SER for Revision 0. The Reactor Protective System is assumed to trip the reactor at 20 minutes.

- The large and small steam line break methodology has been revised to not credit the main feedwater isolation system per the SER for Revision 0. This methodology revision includes a new VIPRE-01 core thermal-hydraulic model for predicting the DNBR.

- An acceptance criterion has been added to the small steam line break analysis to require no fuel failures per the SER for Revision 0. The offsite dose acceptance criterion was changed to 10% of Part 100 based on NRC staff comments at the September 15, 1998 meeting.

- The text sections which state that the offsite dose acceptance criteria have been met have been deleted based on NRC staff comments at the September 15, 1998 meeting.

- Added the BWU-N critical heat flux correlation for application to the Mk-B11 fuel assembly design for DNBR predictions below the mixing vanes and for the steam line break analysis.

- Added centerline fuel melt as a fuel damage criterion for the dropped rod analysis.

The above revisions and all minor changes are listed and identified by margin bars in the attached Revision 1 to DPC-NE-3005 (Attachment 1 - proprietary, and Attachment 2 - non-proprietary versions). The implementation of the DPC-NE-3005-P Revision 1 methodology will be with the Oconee 2 Cycle 18 reload. Submittal of the license amendment request associated with the Oconee 2 Cycle 18 reload is scheduled for February 1999. Approval of Revision 1 to DPC-NE-3005-P is requested by April 1, 1999 to support reload design activities for Oconee 2 Cycle 18, which is scheduled for startup in the fall of 1999.

In accordance with 10CFR 2.790, Duke Power requests that this report be considered proprietary. Information supporting this request is included in the attached affidavit.

If there are any questions or if additional information is needed, please call J. S. Warren at (704) 382-4986.

M. S. Tuckman

M. S. Tuckman

Attachments (2)

xc w/o Attacment 1:

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M. A. Scott
NRC Senior Resident Inspector (ONS)

AFFIDAVIT

1. I am Executive Vice President of Duke Energy Corporation; and as such have the responsibility for reviewing information sought to be withheld from public disclosure in connection with nuclear power plant licensing; and am authorized on the part of said Corporation (Duke) to apply for this withholding.

2. I am making this affidavit in conformance with the provisions of 10CFR 2.790 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke's application for withholding, which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.

4. Pursuant to the provisions of paragraph (b)(4) of 10CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned by Duke and has been held in confidence by Duke and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs relative to a method of analysis that provides a competitive advantage to Duke.

M. S. Tuckman

M. S. Tuckman

(Continued)

(iii) The information was transmitted to the NRC in confidence and under the provisions of 10CFR 2.790, it is to be received in confidence by the NRC.

(iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is marked in Attachment 1 to Duke Energy Corporation letter dated February 1, 1999; SUBJECT: UFSAR Chapter 15 Transient Analysis Methodology, Topical Report DPC-NE-3005-P, Revision 1. This information enables Duke to:

(a) Respond to NRC requests for additional information regarding transient response of Babcock & Wilcox PWRs.

(bc) Simulate UFSAR Chapter 15 transients and accidents for Oconee Nuclear Station.

(cd) Perform safety evaluations per 10CFR50.59.

(de) Support Facility Operating Licenses/Technical Specifications amendments for Oconee Nuclear Station.

M. S. Tuck

M. S. Tuckman

(Continued)

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke.

(a) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

(b) Duke intends to sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke.

5. Public disclosure of this information is likely to cause harm to Duke because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring commensurate expense or allowing Duke to recoup a portion of its expenditures or benefit from the sale of the information.

M. S. Tuck

M. S. Tuckman

(Continued)

M. S. Tuckman, being duly sworn, states that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth within are true and correct to the best of his knowledge.

M. S. Tuck

M. S. Tuckman, Executive Vice President

Subscribed and sworn to before me this 2ND day of

February, 1999

Mary P. Nelms
Notary Public

My Commission Expires:

JAN 22, 2001

SEAL

bxc:

w/Attachments:
G. B. Swindlehurst
J. E. Burchfield
J. A. Perry
R. R. St. Clair
J. S. Warren

w/o Attachments:
L. A. Keller

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

March 17, 1999

Mr. M. S. Tuckman
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P. O. Box 1006 (EC07H)
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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - TRANSIENT ANALYSIS METHODOLOGY, TOPICAL REPORT DPC-NE-3005P - OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 (TAC NOS. MA4713, MA4714, AND MA4715)

Dear Mr. Tuckman:

By letter dated February 1, 1999, Duke Energy Corporation (Duke) submitted Revision 1 of Topical Report DPC-NE-3005P, that describes the new Duke methodology for analyzing the Oconee Updated Final Safety Analysis Report Chapter 15 Non-Loss-of-Coolant Accident transients and accidents.

The principal issue remaining is that credit for main feedwater system isolation would no longer be assumed in the analysis of main steam line breaks. Isolation of main feedwater is not accomplished with safety related equipment. The staff, therefore, required that analyses be performed assuming continued feedwater runout. The licensee performed the requested analyses using the RETRAN code that was previously approved by the staff for Oconee.

In order to complete its review, the staff needs additional information, as shown in the enclosure. We request that you respond by April 16, 1999, as discussed with Mr. Gregg Swindlehurst of your staff.

Sincerely,

David E. LaBarge, Senior Project Manager
Project Directorate II-2
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Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270, and 50-287

Enclosure: Request for Additional Information

cc w/encls: See next page

Oconee Nuclear Station

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Request for Additional Information

Topical Report DPC-NE3005-P, Revision 1

UFSAR Chapter 15 Transient and Accident Methodology

Oconee Nuclear Station, Units 1, 2, and 3

The following questions deal with the revised RETRAN analyses of a postulated main steam line break (MSLB) described in Section 15 of DPC-NE-3005-P. The analyses include the consequences of failures to isolate main feedwater.

1. Pages 15-3 and 15-4 state that an inlet core mixing fraction is of percent was used with the MSLB analyses and that the mixing fraction was obtained from tests performed on Oconee Unit 1. Discuss the mixing tests and how the mixing fraction was derived. Justify that this fraction is appropriate for MSLB analysis.

2. Page 15-4 states that the RETRAN transport delay model was disabled for the MSLB analyses since flow in the intact loop becomes stagnant so that use of the model is inappropriate. The staff understands that enthalpy transport was also disabled in the affected coolant loop that would not become stagnant. Consideration of enthalpy transport in the affected coolant loop might lead to an increased rate of core overcooling. Provide the results of an evaluation showing the effect of considering enthalpy transport in the affected loop on core power and the departure from nucleate boiling ratio (DNBR).

3. A postulated MSLB with off site power available would permit continued main feedwater flow and reactor coolant pump operation. Page 15-10 states that the reactor coolant pumps in the unaffected loop are assumed to trip at 100 seconds. This was stated to be conservative since reverse heat transfer was occurring in the steam generator, which was minimizing overcooling. Operators are trained to trip all reactor coolant pumps on loss of subcooling margin. Evaluate the MSLB for the case of continued main feedwater flow but with operator action to trip all reactor coolant pumps. Determine the effect on DNBR of any return to power.

4. During a large main steam line break at Oconee the core flood tanks (CFTs) would be expected to discharge and add boric acid, which would act to reduce core power production. As the CFTs discharge, the cover gas will cool to a temperature lower than the liquid. RETRAN does not have a non-equilibrium CFT model. To assume equilibrium might lead to a higher CFT pressure and a greater discharge than would actually occur. Justify that the CFT model in RETRAN is appropriate for analysis of main steam line breaks.

5. Page 15-12 states that the method of calculating boron reactivity feedback involves use of the average core boron concentration and the average core boron worth. Boric acid from the CFTs and safety injection would reach the bottom of the core first, which would have a lower reactivity importance than the average core. Justify that using the average core boron concentration and average core boron worth is appropriate.

6. Page 15-14 states that the limiting assumption with respect to maximizing overcooling and reactivity addition has been determined by analysis to be the case when the Integrated

Control System (ICS) is controlling main feedwater. This case was found to cause more overcooling than the case that assumes failure of the ICS, which would permit unlimited main feedwater addition to the affected steam generator. Provide comparisons of the conditions within the affected steam generator secondary that would cause the ICS operating case to be the worst case. Compare steam generator level, pressure, temperature, and heat transfer coefficients that are calculated by the RETRAN code.

7. To enable the NRC staff to perform audit calculations and sensitivity analyses if needed, please provide electronic copies (on computer disks) of the RETRAN and VIPRE input decks used for MSLB analysis.

Duke Energy

M. S. Tuckman
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April 19, 1999

U. S. Nuclear Regulatory Commission
Washington D. C. 20555

ATTENTION: Document Control Desk

Subject: Oconee Nuclear Station
Docket Numbers 50-269, -270, -287
UFSAR Chapter 15 Transient Analysis Methodology,
Topical Report DPC-NE-3005-P, Revision 1

References: 1) Letter, M. S. Tuckman (Duke), to NRC, July 30, 1997

2) D. E. LaBarge (NRC) to W. R. McCollum (Duke), October 1, 1998

3) Letter, M. S. Tuckman (Duke), to NRC, February 1, 1999

By means of Reference 1, Duke Energy Corporation submitted Topical Report DPC-NE-3005-P which describes the new Duke methodology for analyzing the Oconee UFSAR Chapter 15 non-LOCA transients and accidents. NRC review and acceptance of this topical report were documented in a SER forwarded by Reference 2. Several issues requiring revisions to the topical report methodology were identified in Reference 2 and in a meeting with the NRC staff in Rockville on September 15, 1998. The requested revisions to the topical report were addressed by Duke in Revision 1 to DPC-NE-3005-P, which was submitted to the NRC by Reference 3. NRC letter dated March 17, 1999 contained seven questions on DPC-NE-3005-P, Revision 1. The Duke responses to these questions are contained in the attachments to this letter.

Some of the information contained in the attachments is considered proprietary. The proprietary information is that which is indicated by the bold brackets shown in Attachment 2. In accordance with 10CFR 2.790, Duke Energy Corporation requests that this information be considered proprietary. An affidavit supporting this request is included with this letter. A non-proprietary version is also included within as Attachment 1.

If there are any questions or if additional information is needed on this matter , please call J. S. Warren at (704) 382-4986.

M. S. Tuckman

M. S. Tuckman

Attachments (2)

xc w/Attacment 1 only:
L. A. Reyes, Regional Administrator
U. S. Nuclear Regulatory Commission
Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, GA 30303

M. A. Scott
NRC Senior Resident Inspector (ONS)

xc w/Attachment 1, Attachment 2, and Diskette:
D. E. Labarge, NRC Senior Project Manager (ONS)
U. S. Nuclear Regulatory Commission
Mail Stop O-8 H12
Washington, DC 20555-0001

AFFIDAVIT

1. I am Executive Vice President of Duke Energy Corporation; and as such have the responsibility for reviewing information sought to be withheld from public disclosure in connection with nuclear power plant licensing; and am authorized on the part of said Corporation (Duke) to apply for this withholding.

2. I am making this affidavit in conformance with the provisions of 10CFR 2.790 of the regulations of the Nuclear Regulatory Commission (NRC) and in conjunction with Duke's application for withholding, which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.

4. Pursuant to the provisions of paragraph (b)(4) of 10CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned by Duke and has been held in confidence by Duke and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs relative to a method of analysis that provides a competitive advantage to Duke.

M. S. Tuck

M. S. Tuckman

(Continued)

(iii) The information was transmitted to the NRC in confidence and under the provisions of 10CFR 2.790, it is to be received in confidence by the NRC.

(iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is marked in Attachment 2 to Duke Energy Corporation letter dated April 19, 1999; SUBJECT: UFSAR Chapter 15 Transient Analysis Methodology, Topical Report DPC-NE-3005-P, Revision 1. This information enables Duke to:

(a) Respond to NRC requests for additional information regarding transient response of Babcock & Wilcox PWRs.

(bc) Simulate UFSAR Chapter 15 transients and accidents for Oconee Nuclear Station.

(cd) Perform safety evaluations per 10CFR50.59.

(de) Support Facility Operating Licenses/Technical Specifications amendments for Oconee Nuclear Station.

M. S. Tuckman

M. S. Tuckman

(Continued)

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke.

(a) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

(b) Duke intends to sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke.

5. Public disclosure of this information is likely to cause harm to Duke because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring commensurate expense or allowing Duke to recoup a portion of its expenditures or benefit from the sale of the information.

M. S. Tuck

M. S. Tuckman

(Continued)

M. S. Tuckman, being duly sworn, states that he is the person who subscribed his name to the foregoing statement, and that all the matters and facts set forth within are true and correct to the best of his knowledge.

M. S. Tuckman

M. S. Tuckman, Executive Vice President

Subscribed and sworn to before me this 19TH day of April, 1999

Mary P. Nelms

Notary Public

My Commission Expires:

JAN 22, 2001

SEAL

bxc:

<u>w/Attachments 1 and 2:</u>
G. B. Swindlehurst
J. E. Burchfield
J. W. Sawyer
R. R. St. Clair
J. E. Smith
J. S. Warren
ELL

<u>w/o Attachments:</u>
C. J. Thomas

Attachment 1

Responses to Request For Additional Information
DPC-NE-3005P Revision 1

Q1. Pages 15-3 and 15-4 state that an inlet core mixing fraction of [ ] was used with the MSLB analyses and that the mixing fraction was obtained from tests performed on Oconee Unit 1. Discuss the mixing tests and how the mixing fraction was derived. Justify that this fraction is appropriate for MSLB analysis.

A1. [ ]

Q2. Page 15-4 states that the RETRAN transport delay model was disabled for the MSLB analyses since flow in the intact loop becomes stagnant so that use of the model is inappropriate. The staff understands that enthalpy transport was also disabled in the affected coolant loop that would not become stagnant. Consideration of enthalpy transport in the affected coolant loop might lead to an increased rate of core cooling. Provide the results of an evaluation showing the effect of considering enthalpy transport in the affected loop on core power and the departure from nucleate boiling ratio (DNBR).

A2. The RETRAN temperature transport delay model is used to more mechanistically treat temperature changes in piping volumes. The temperature transport delay option accounts for the fact that temperature changes move through piping as a front, while the finite difference homogeneous equilibrium model approach instantaneously mixes the incoming

fluid with the volume contents. The temperature transport delay option establishes a mesh substructure with the volume to track temperature front movement. For the MSLB analysis, the temperature transport delay option is disabled in the primary loop due to flow reversals predicted in the intact loop. Experience with RETRAN has shown that the temperature transport delay model can give anomalous predictions during flow reversals. Turning off the temperature transport delay option in the affected loop will result in a more rapid propagation of the fluid cooled in the steam generator tube bundle to the core due to instantaneous mixing. This is conservative with respect to more rapidly decreasing the core moderator temperature, and thereby more likely to result in an earlier core return-to-power. Since boron injected by the ECCS systems terminates any return-to-power, and since ECCS inventory addition is subject to physical delays, having the return-to-power occur earlier in the event will lead to a more severe peak power statepoint. Sensitivity studies indicate that turning off the temperature transport delay option in the primary loop is a small penalty to the results of the MSLB analysis.

Q3. A postulated MSLB with off site power available would permit continued main feedwater flow and reactor coolant pump operation. Page 15-10 states that the reactor coolant pumps in the unaffected loop are assumed to trip at 100 seconds. This was stated to be conservative since reverse heat transfer was occurring in the steam generator, which was minimizing overcooling. Operators are trained to trip all reactor coolant pumps on loss of subcooling margin. Evaluate the MSLB for the case of continued main feedwater flow but with operator action to trip all reactor coolant pumps. Determine the effect on DNBR during any return-to-power.

A3. As described in DPC-NE-3005, Revision 1, in the MSLB return-to-power case the reactor coolant pumps (RCPs) in the unaffected loop are tripped at 100 seconds. To determine the effect of manually tripping all RCPs on a loss of the subcooled margin, the case which results in a return-to-power of approximately 13 %FP heat flux at 160 seconds, is reanalyzed. The additional case trips the RCPs in the broken SG loop at 160 seconds. The RETRAN boundary conditions for this additional case (core exit pressure, core inlet temperature, core inlet flow rate, and core average heat flux) are input to the VIPRE code to calculate the minimum DNBR. The minimum DNBR for this additional case is predicted to be 3.15 using the W-3S CHF correlation. It is noted that the Oconee emergency operating procedure does not allow the operators to trip the RCPs on loss of subcooled margin if the reactor power level is greater than 5%.

Q4. During a large main steam line break at Oconee the core flood tanks (CFTs) would be expected to discharge and add boric acid, which would act to reduce core power production. As the CFTs discharge, the cover gas will cool to a temperature lower than the liquid. RETRAN does not have a non-equilibrium CFT model. To assume equilibrium might lead to a higher CFT pressure and a greater discharge than would actually occur. Justify the CFT model in RETRAN is appropriate for analysis of main steam line breaks.

A4. The MSLB analysis uses the RETRAN separated volume model with air above subcooled liquid to model the CFTs. The initial conditions are 65 °F, 565 psia, 70.4% level and a conservatively low initial boron concentration of 1817 ppm. At the time of the peak power and minimum DNBR for the return-to-power case (approximately 160 seconds), the CFT conditions are 65 °F, 474 psia, and approximately 64% level. Note that the CFT level change is small due to the limited RCS depressurization below 565 psia. In reality the

nitrogen cover gas will cool as the CFT depressurizes, and the RETRAN equations do not model this effect. However, since the initial temperature is assumed conservatively low at 65 °F, not much further temperature reduction is possible if this effect was accurately modeled. It should also be noted that the CFT injection lines are initially modeled with zero boron concentration.

Duke has benchmarked the RETRAN CFT model against Oconee CFT full blowdown tests with good agreement with data. The blowdown test is more challenging than the limited discharge transient that occurs during the MSLB. Based on the low initial CFT temperature, the limited CFT discharge that occurs during the MSLB analysis, and the results of the benchmarking analysis, the RETRAN CFT model is appropriate for MSLB analysis.

Q5. Page 15-12 states that the method of calculating boron reactivity feedback involves the use of the average core boron concentration and the average core boron worth. Boric acid from the CFTs and safety injection would reach the bottom of the core first, which would have a lower reactivity importance than the average core. Justify that using the average core boron concentration and average boron worth is appropriate.

A5. As described in DPC-NE-3005, Revision 1, in the MSLB return-to-power case the average core boron concentration and a conservative boron worth are assumed in modeling the reactivity due to ECCS boron injection. A differential boron worth of 120 ppm/%Δk/k is selected as an upper bound over the range of temperatures expected during the MSLB accident. The fact that the boron worth decreases with decreasing temperature is conservatively ignored. To determine the effect of assuming the core average boron concentration vs. the slightly lower boron concentration in the upper part of the core, the case which results in a return-to-power of approximately 13 %FP heat flux at 160 seconds, is reanalyzed. The additional case uses the boron concentration from the top two core volumes, rather than the core average, to determine the reactivity contribution from boron. The results show an increase in the core heat flux from 13.09% to 13.12%, which is essentially the same result.

Q6. Page 15-14 states that the limiting assumption with respect to maximizing overcooling and reactivity addition has been determined by analysis to be the case when the Integrated Control System (ICS) is controlling main feedwater. This case was found to cause more overcooling than the case that assumes failure of the ICS, which would permit unlimited main feedwater addition to the affected steam generator. Provide comparisons of the conditions within the affected steam generator secondary that would cause the ICS operating case to be the worst case. Compare steam generator level, pressure, temperature, and heat transfer coefficients that are calculated by the RETRAN code.

A6. MSLB analysis results show that the highest return-to-power occurs for those cases in which the ICS controls main feedwater (MFW) flow based on steam generator level, and that the 25 inch minimum post-trip level setpoint conservatively increased to 100 inches is the limiting level setpoint. The peak return-to-power for this case bounds the return-to-power predicted for the uncontrolled MFW case, in which it is assumed that the ICS fails to either control MFW flow or to trip MFW pumps on high SG level. Without ICS control the continued MFW flow results in steam generator overfill. The results of these two cases are presented in the following figures and discussion.

Figure 1 shows the feedwater flowrates to the affected steam generator. With the ICS controlling level to 100 inches, the MFW flow is throttled, whereas for the ICS failed both MFW and EFW flowrates are uncontrolled and much higher. Figure 2 shows the resulting steam generator level. With level control the level is approximately controlled to 100 inches, although the dynamic behavior of the transient causes instability in feedwater control and level response. Without level control the level continues to fill until going offscale at approximately 250 seconds. Figure 3 shows the steam generator liquid mass. With ICS level control the liquid mass is limited to 20,000-50,000 lbm, whereas the overfilled condition without ICS control reaches 180,000 lbm. The significant difference in the feedwater flow boundary condition for the two cases is evident.

Figure 4 shows the comparison of the steam generator pressures at the top of the tube bundle. There is very little difference in these pressures, which indicates that the secondary saturation temperatures are similar. Figure 5 shows the comparison of the total heat transfer across the steam generator tubes. During the initial phase in which the steam generator is blowing down, the heat transfer rates are the same or similar. Then the case with ICS controlling level is somewhat higher in the period of 50-200 seconds. From 200-400 seconds the case without ICS control is somewhat higher. From Figures 1-5 it can be concluded that the higher feedwater flow without ICS control is offset by the lower heat transfer per lbm of feedwater, since much of the feedwater exits the steam generator in the liquid phase. Most of the feedwater in the case with the ICS controlling to 100 inches is boiled before exiting the steam generator, and therefore a higher heat transfer per lbm occurs. Figure 6 shows the core neutron power response. The case with ICS feedwater control peaks at 13.09% power at 160 seconds, and the case without ICS control peaks at 6.9% power. This occurs before the time period during which the case without ICS control experiences higher heat transfer.

Figures 7-16 show the heat transfer coefficient and the heat transfer rate for [ ] steam generator tube conductors. The [ ] were selected to show the axial variation in the predictions. The elevation and height of each conductor above the lower tubesheet are as follows:

Conductor	Elevation Span (ft)
[	]

Under full power steady-state conditions, the SG tube bundle heat conductors exhibit well defined heat transfer modes. The lower bundle region (Conductors [ ]) contains a relatively large amount of liquid, and is in the nucleate boiling heat transfer mode. The middle bundle region (Conductors [ ]) contains a larger ratio of steam to liquid, and is in the forced convection vaporization heat transfer mode. The upper bundle region (Conductors [ ]) contains steam and is in the forced convection to superheated vapor heat transfer mode. During the blowdown phase of the MSLB, conditions within the SG undergo a violent change. The decompression of the steam generator causes liquid in the lower region of the tube bundle and in the downcomer to be transported to higher elevations within the tube bundle by entrainment in the high velocity steam flow. Due to the severe pressure reduction, flashing occurs in the downcomer and results in reverse flow through the aspirator port. The reverse aspirator port flow contributes to a second boiling region in the upper tube bundle. Upper tube bundle conductors, which are at the hottest temperatures, are

periodically quenched with slugs of entrained liquid. For the ICS failure case, the SG gradually fills with liquid after the initial blowdown and experiences less boiling. The accumulation of subcooled liquid as the overfill progresses results in RETRAN predicting the forced convection to subcooled liquid heat transfer mode. For the ICS controlling to 100 inches case, MFW flow is unstable as the ICS attempts to maintain the level setpoint. During periods of high MFW flowrates the liquid levels within the tube bundle increase, causing heat transfer behavior similar to the initial blowdown. During periods of low MFW flowrates, the hotter tube bundle regions dry out. As the temperature differences between the primary and secondary decrease as the analysis progresses, heat transfer coefficients and rates decrease for all conductors for both cases. The integral effect of the heat transfer predicted for each tube bundle conductor is described in the preceeding paragraphs. The overall heat transfer rate, and therefore the consequences of the two MSLB cases are quite similar, with the case with ICS controlling to the level setpoint slightly more limiting.

Q7. To enable the NRC staff to perform audit calculations and sensitivity analyses if needed, please provide electronic copies (on computer disks) of the RETRAN and VIPRE input decks used for MSLB analysis.

A7. The RETRAN and VIPRE inputs decks on diskette have been mailed to the Oconee Project Manager.

MAIN STEAM LINE BREAK

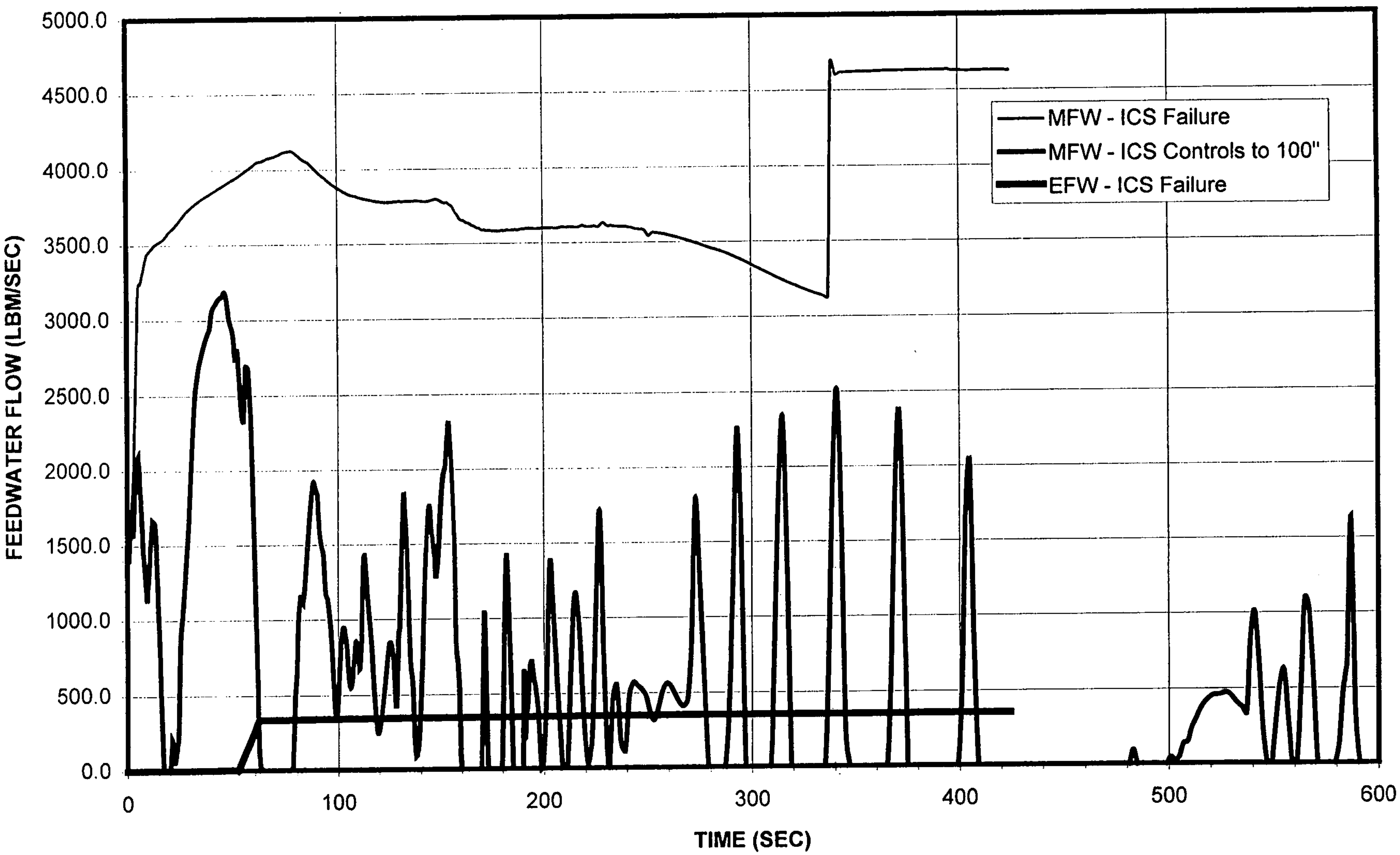

Figure 1

MAIN STEAM LINE BREAK

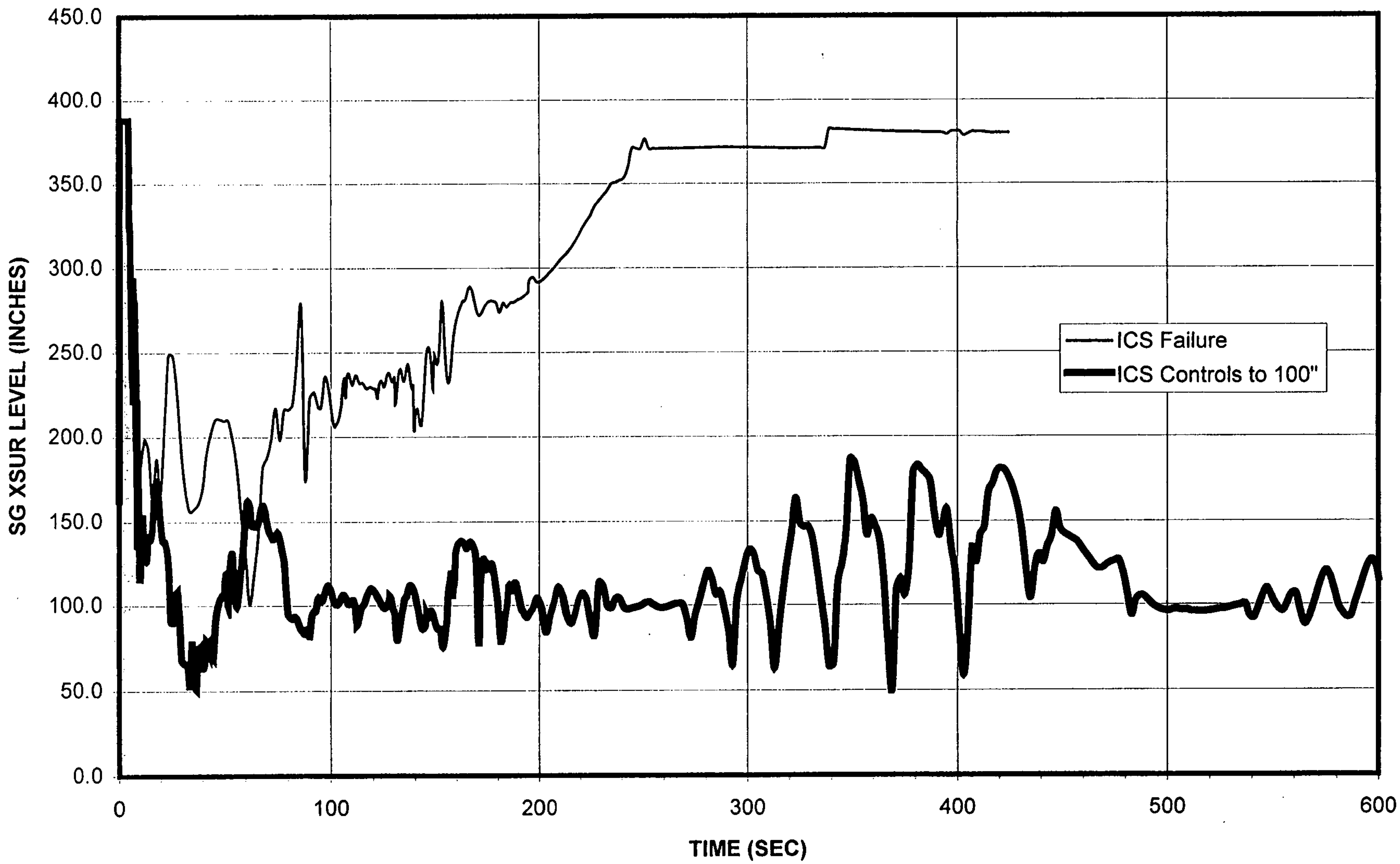

Figure 2

MAIN STEAM LINE BREAK

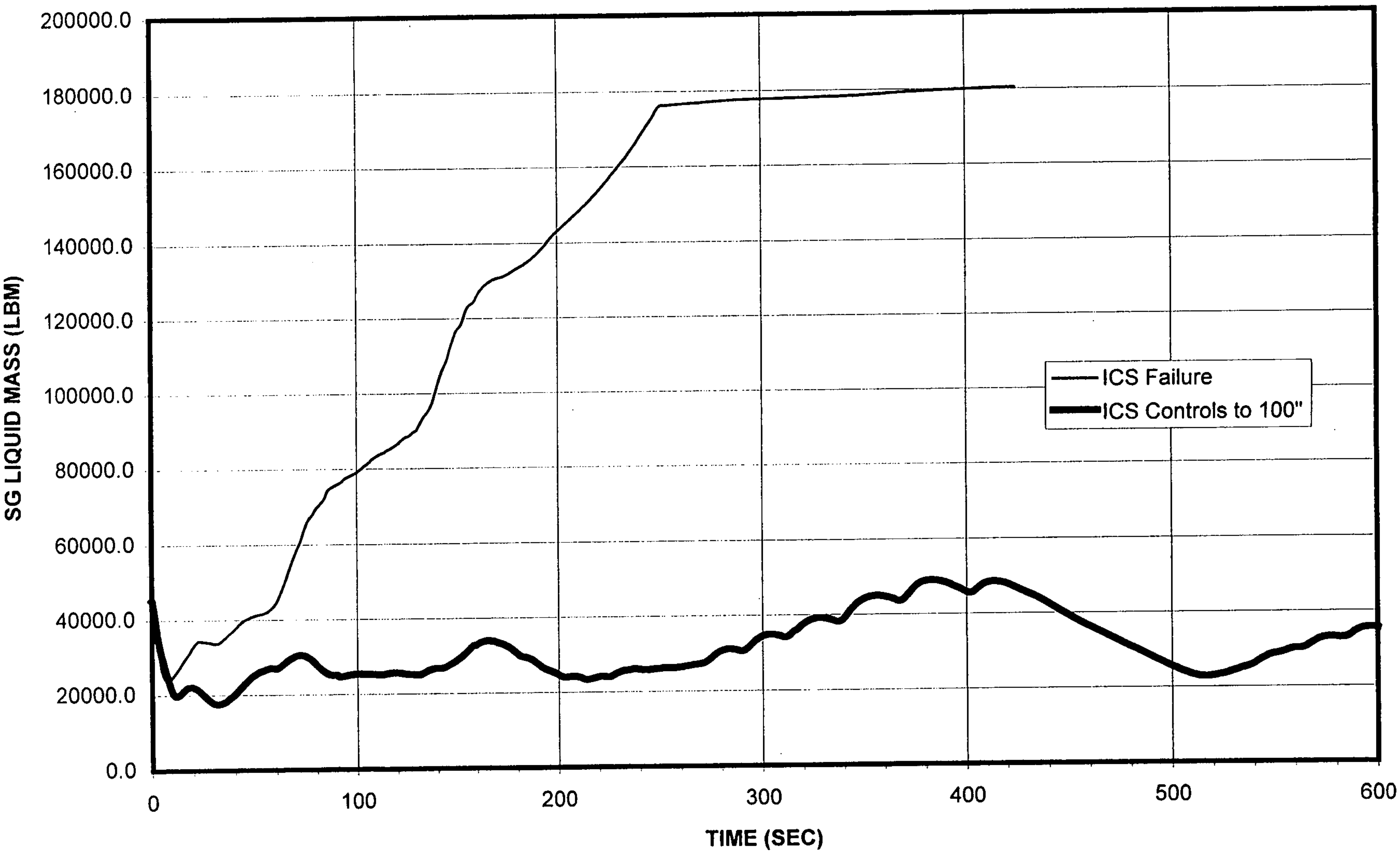

Figure 3

MAIN STEAM LINE BREAK

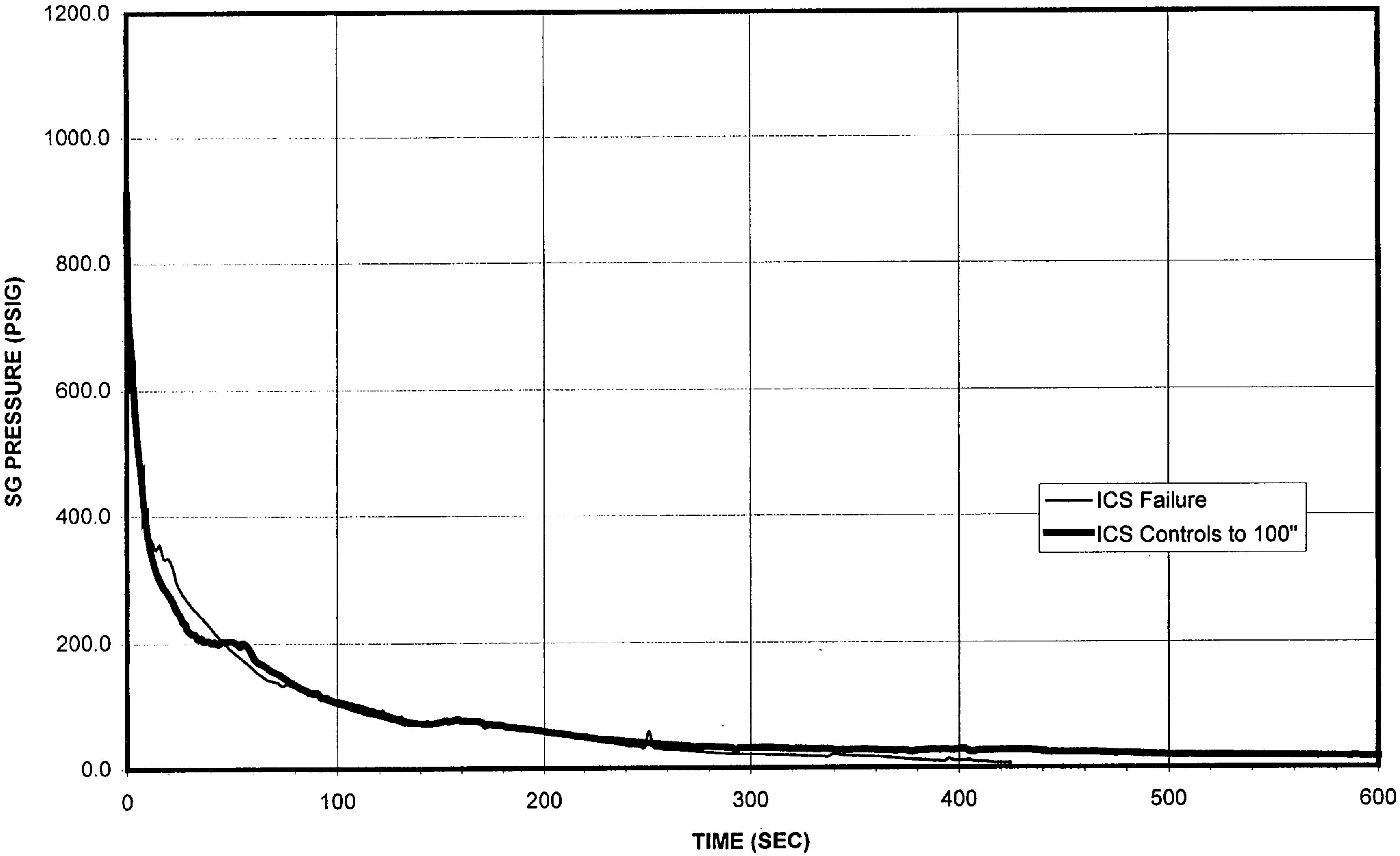

Figure 4

MAIN STEAM LINE BREAK

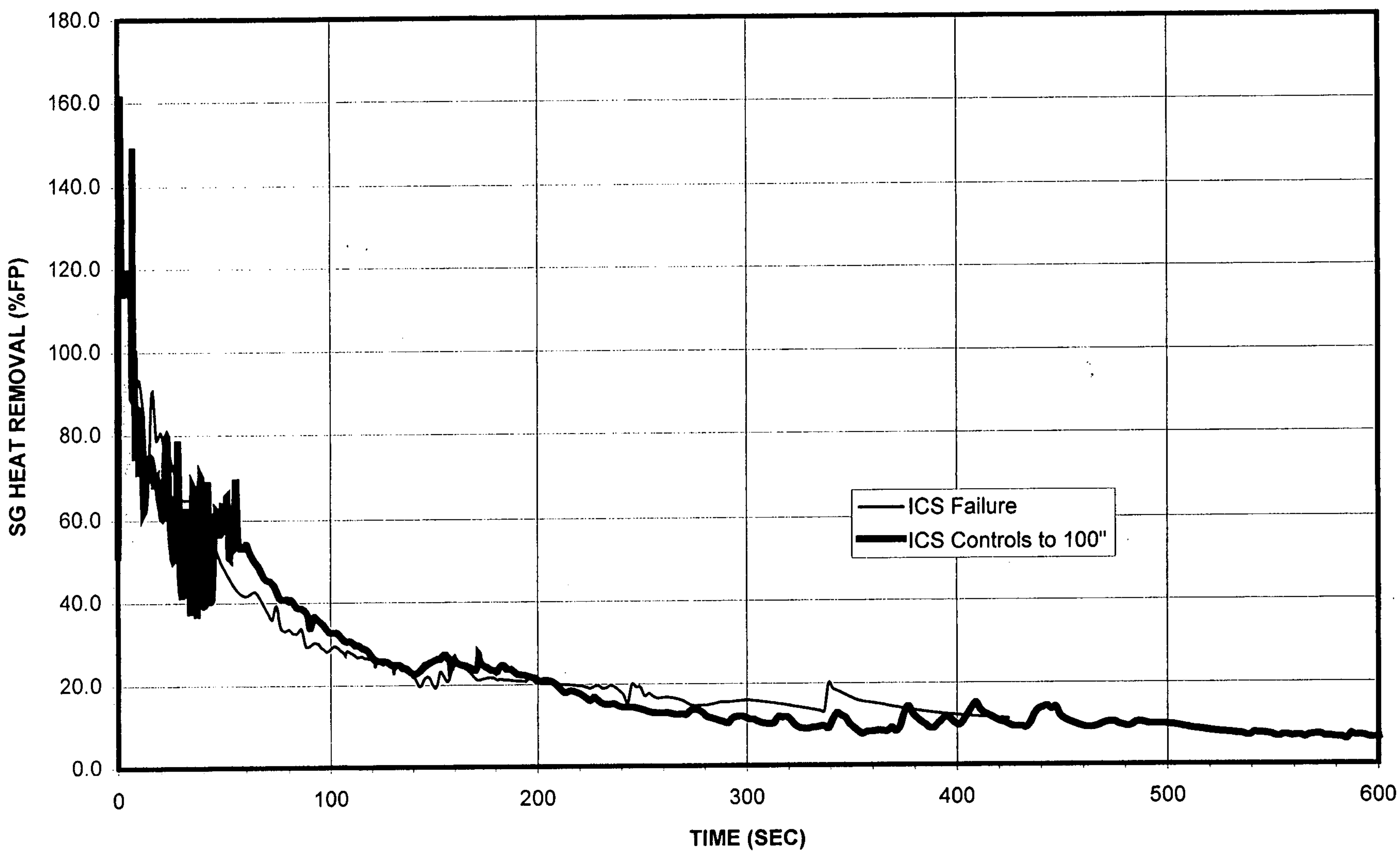

Figure 5

MAIN STEAM LINE BREAK

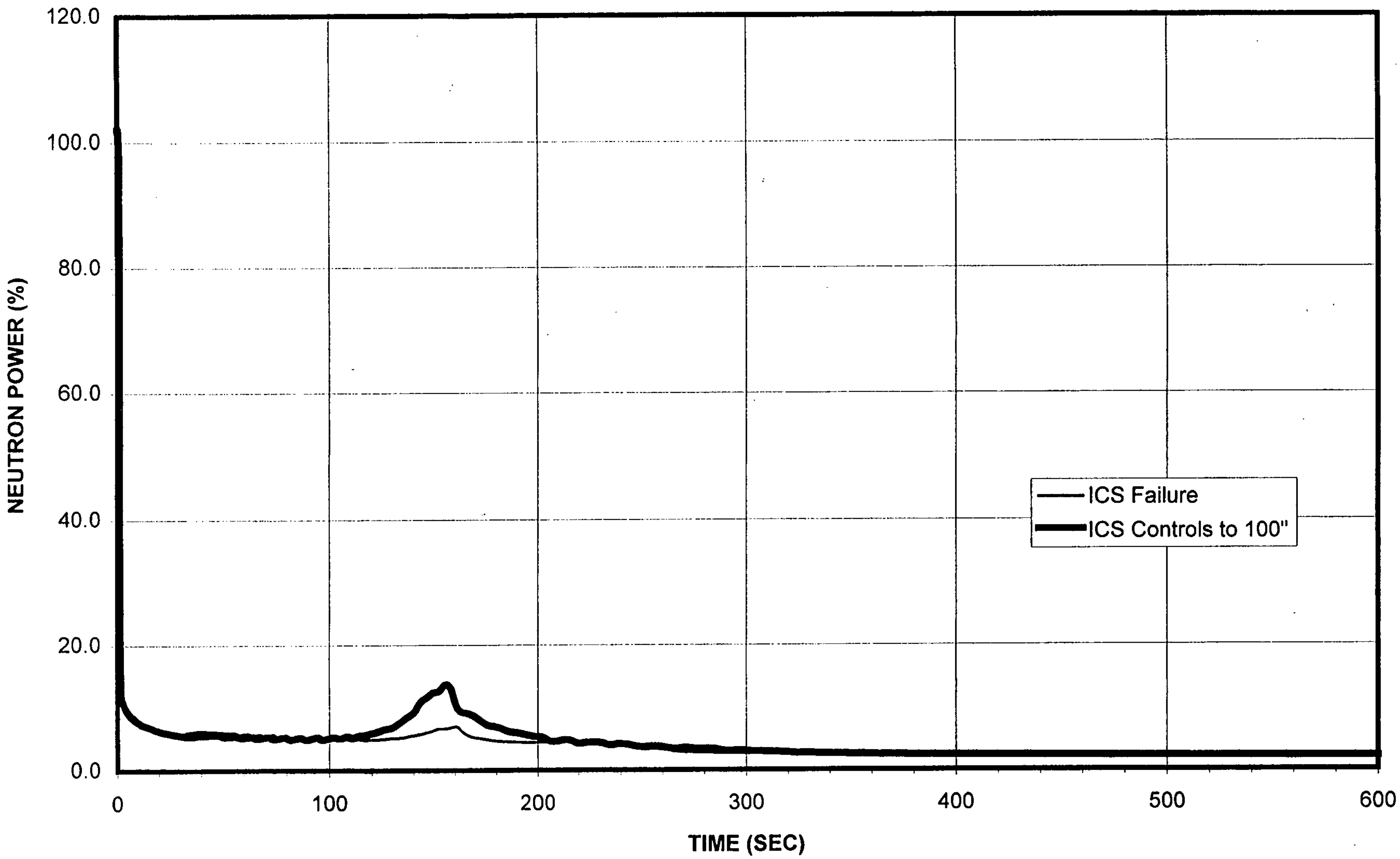

Figure 6

MAIN STEAM LINE BREAK

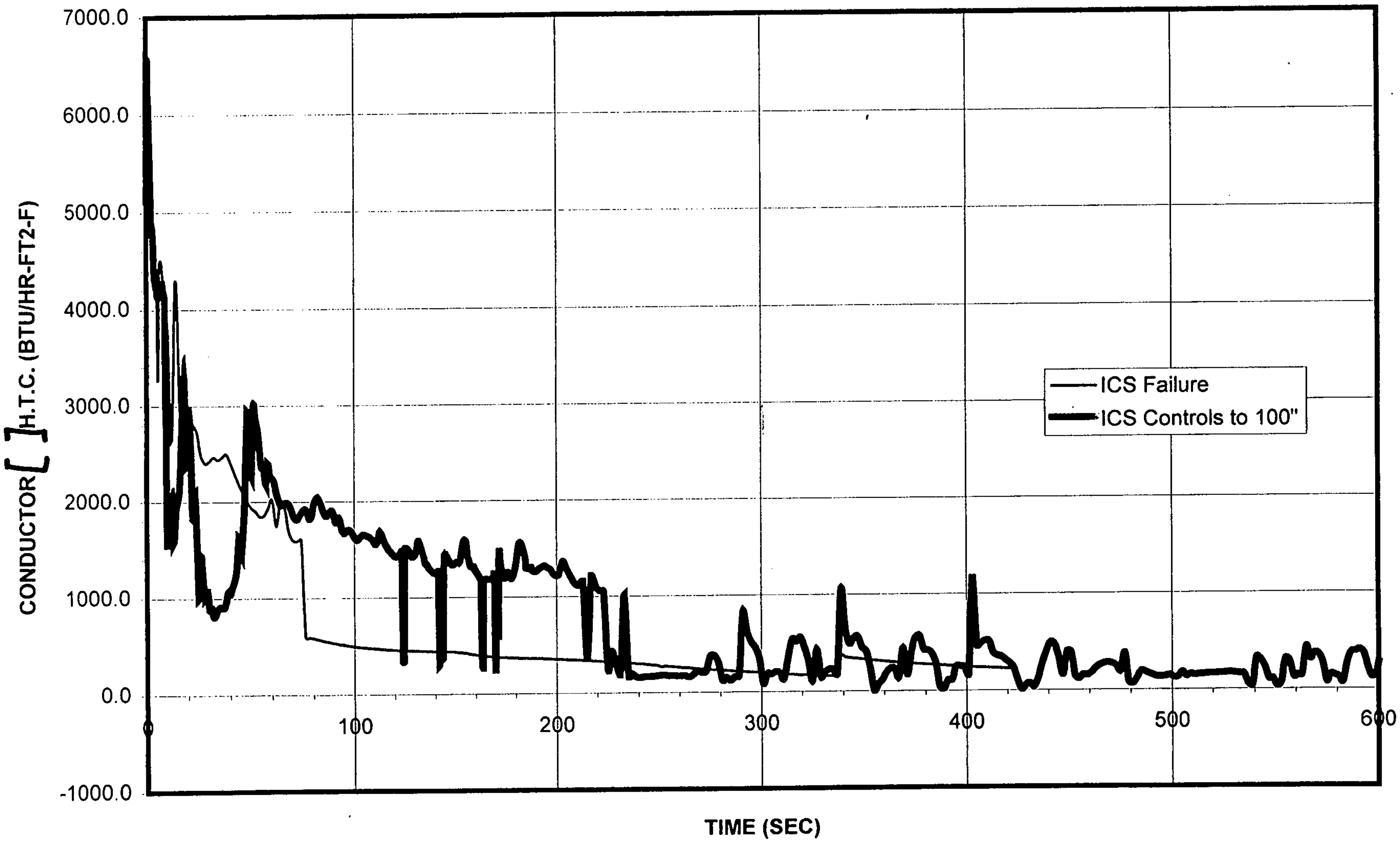

Figure 7

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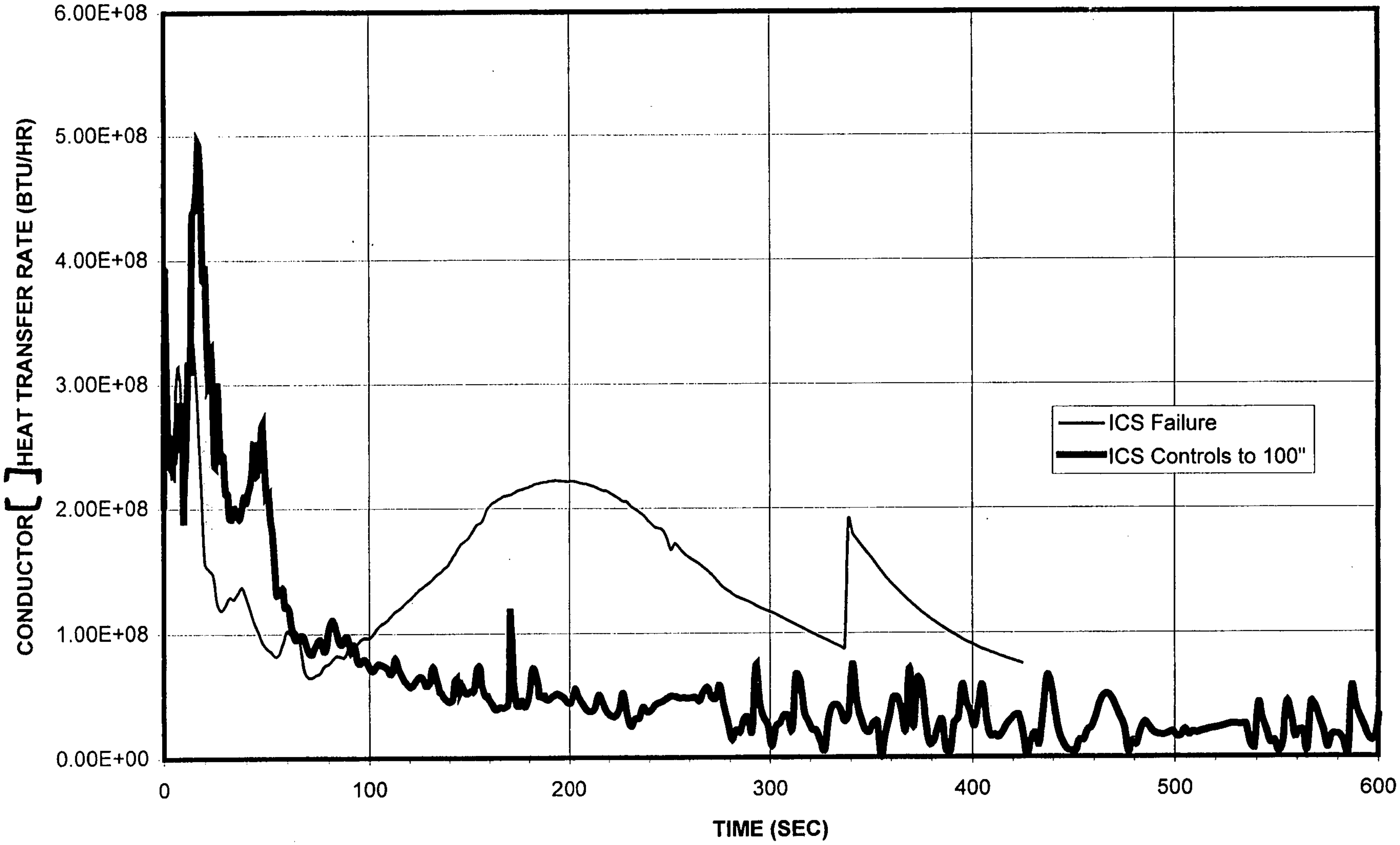

Figure 8

MAIN STEAM LINE BREAK

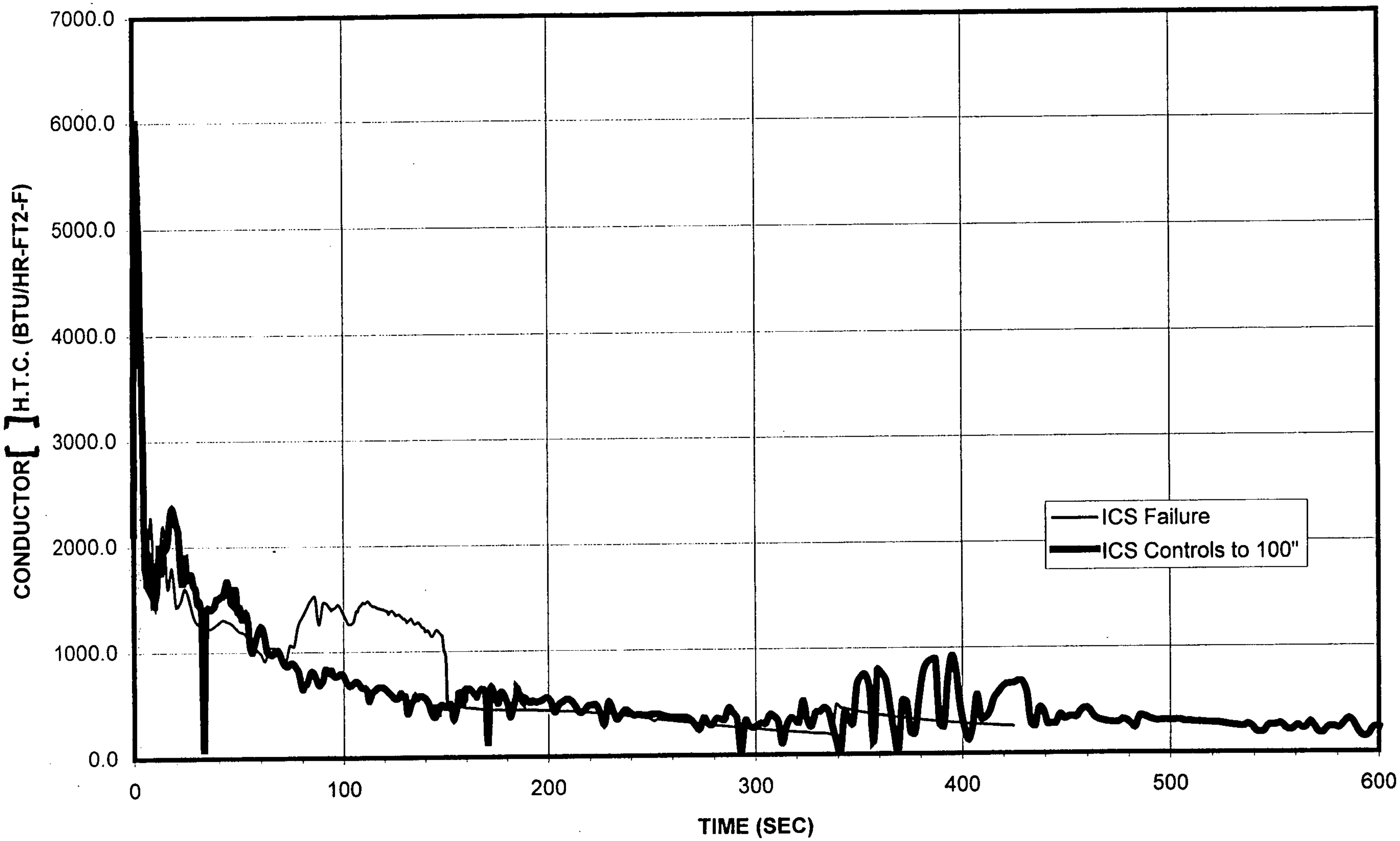

Figure 9

MAIN STEAM LINE BREAK

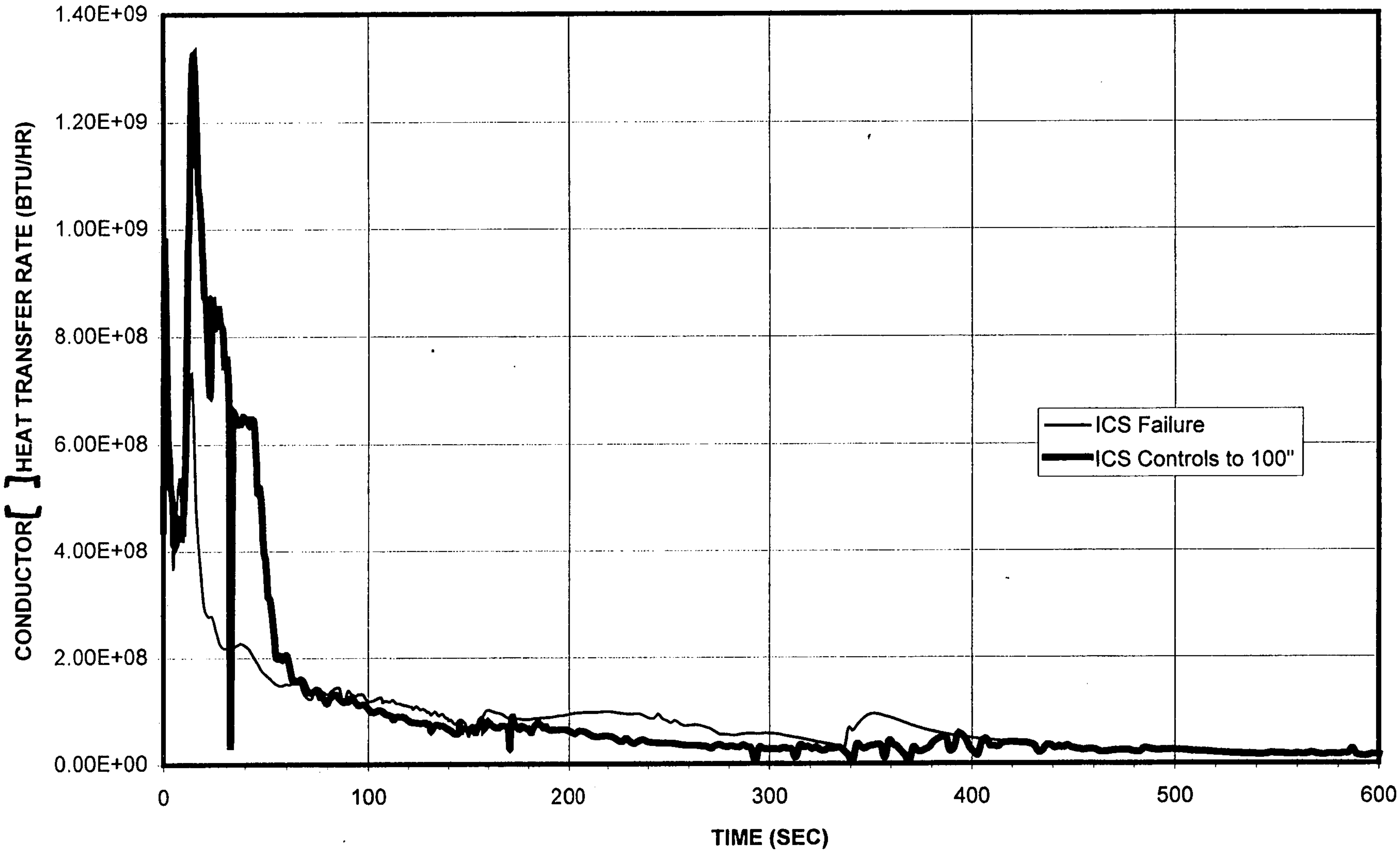

Figure 10

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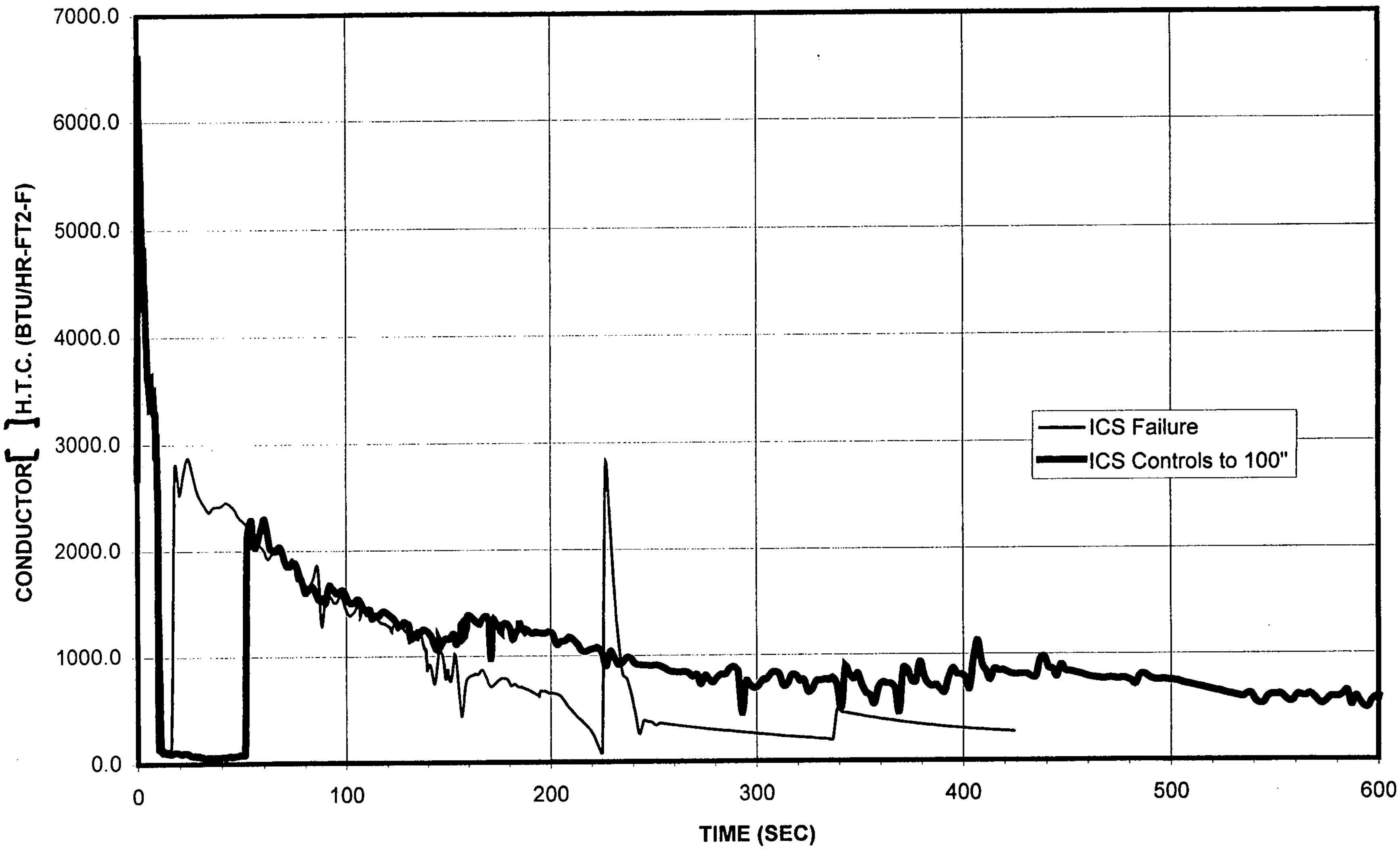

Figure 11

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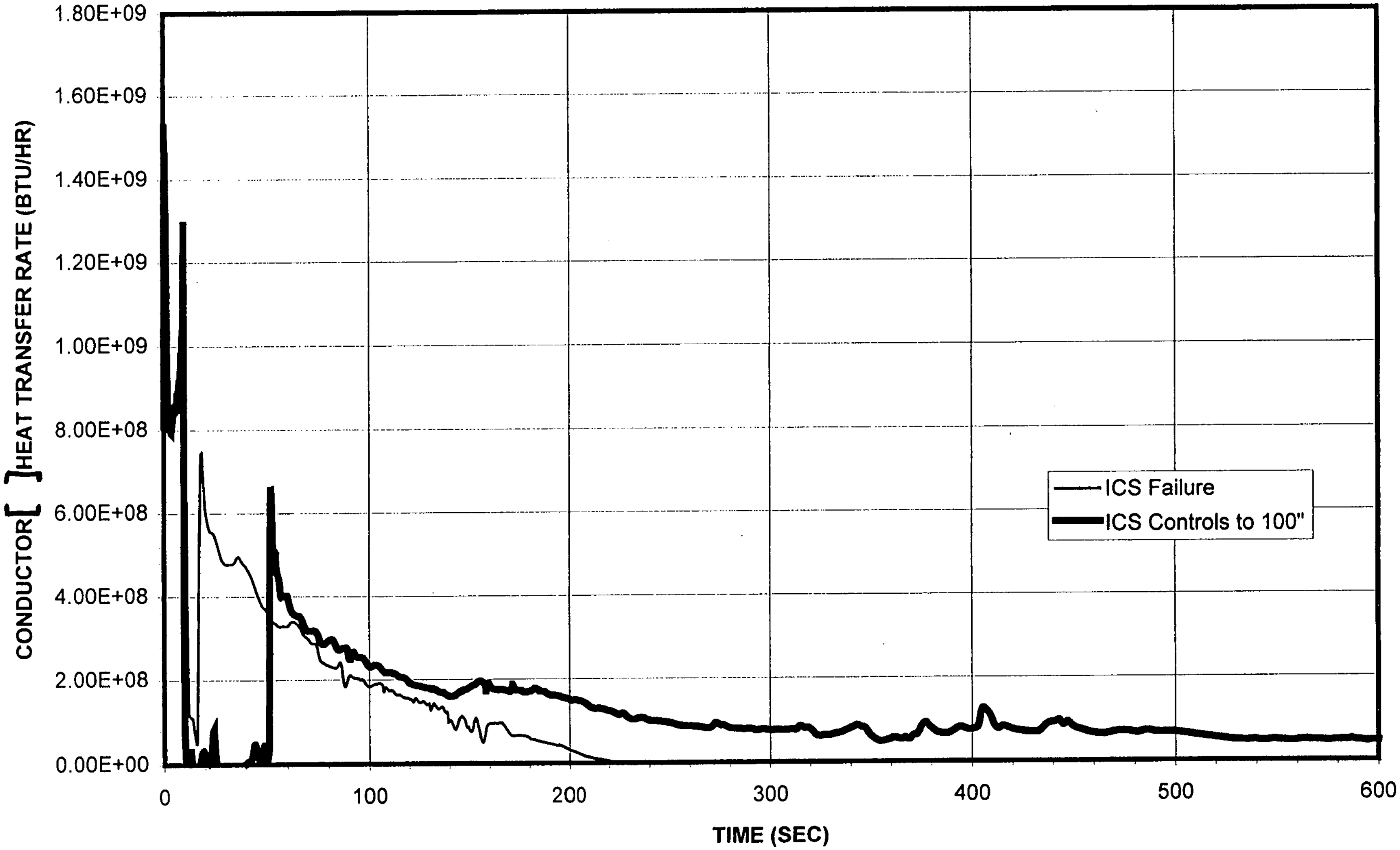

Figure 12

MAIN STEAM LINE BREAK

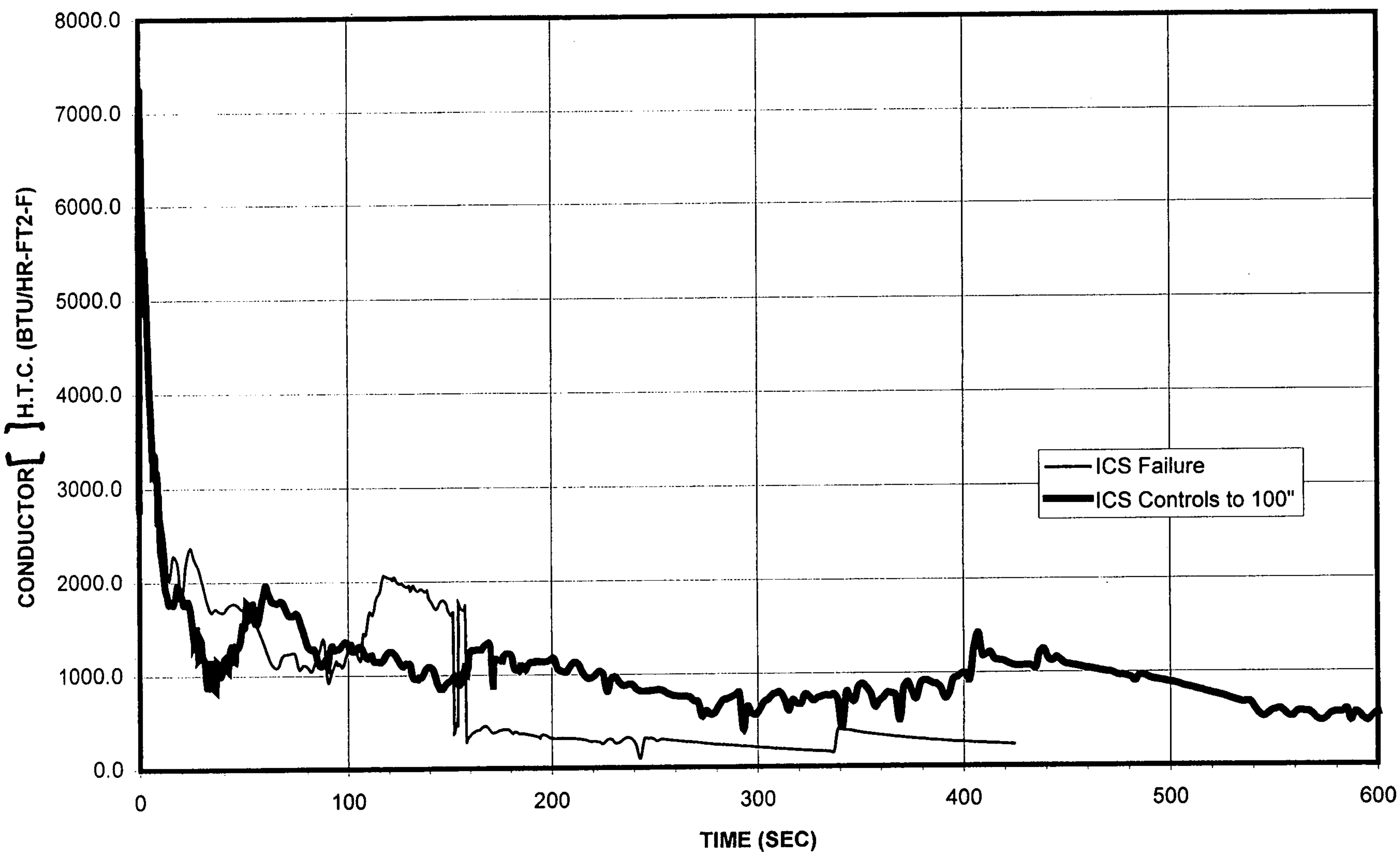

Figure 13

MAIN STEAM LINE BREAK

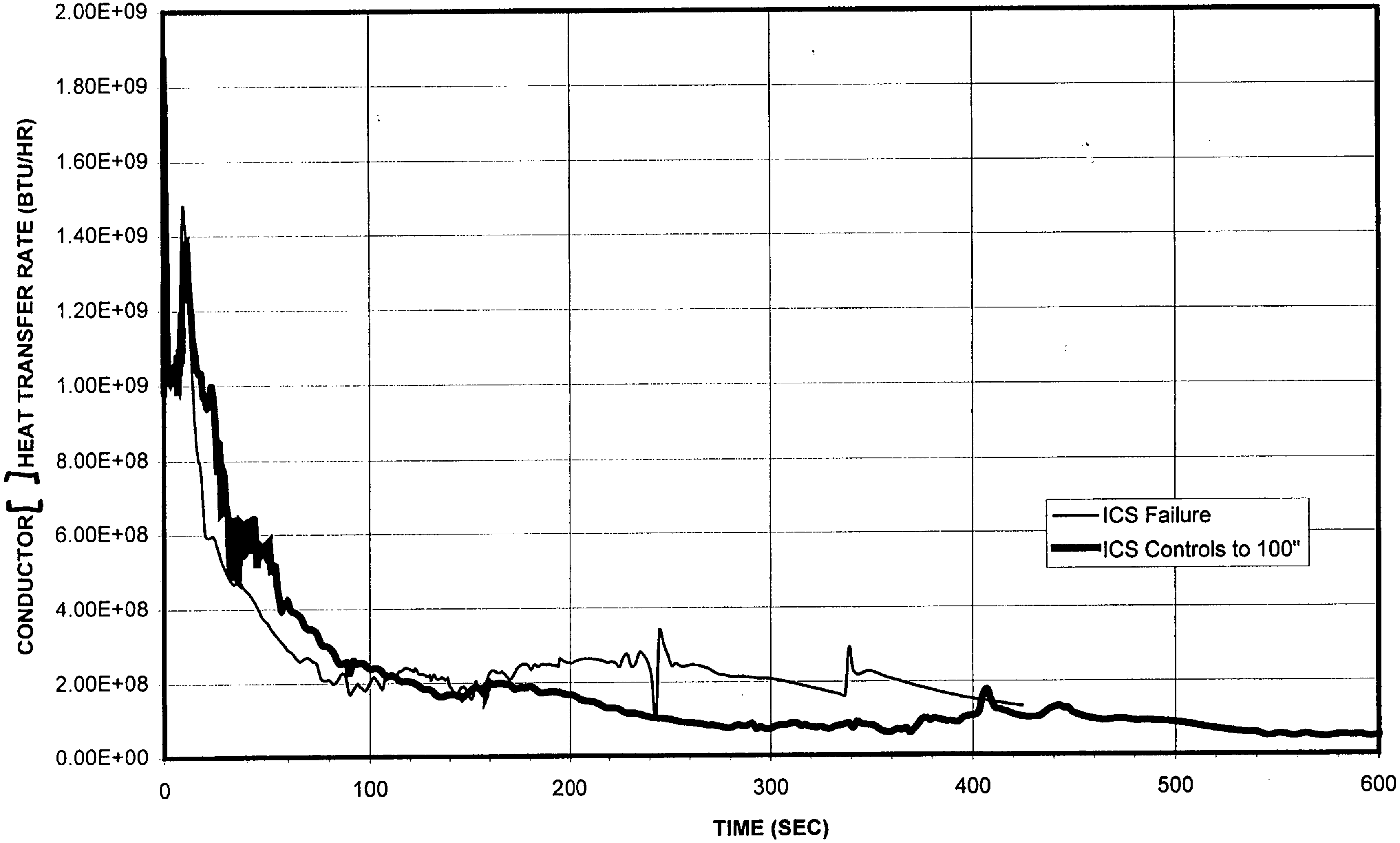

Figure 14

MAIN STEAM LINE BREAK

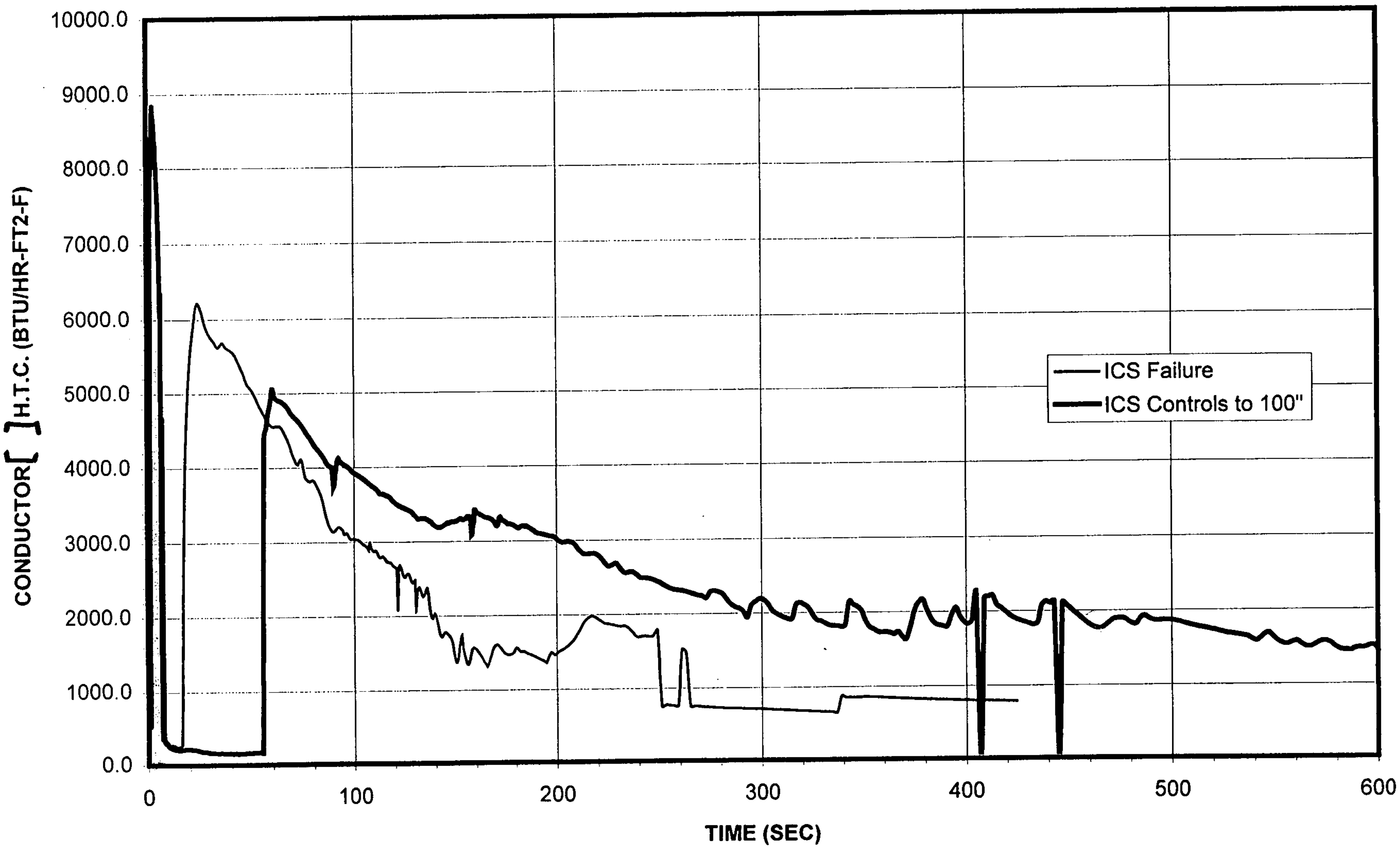

Figure 15

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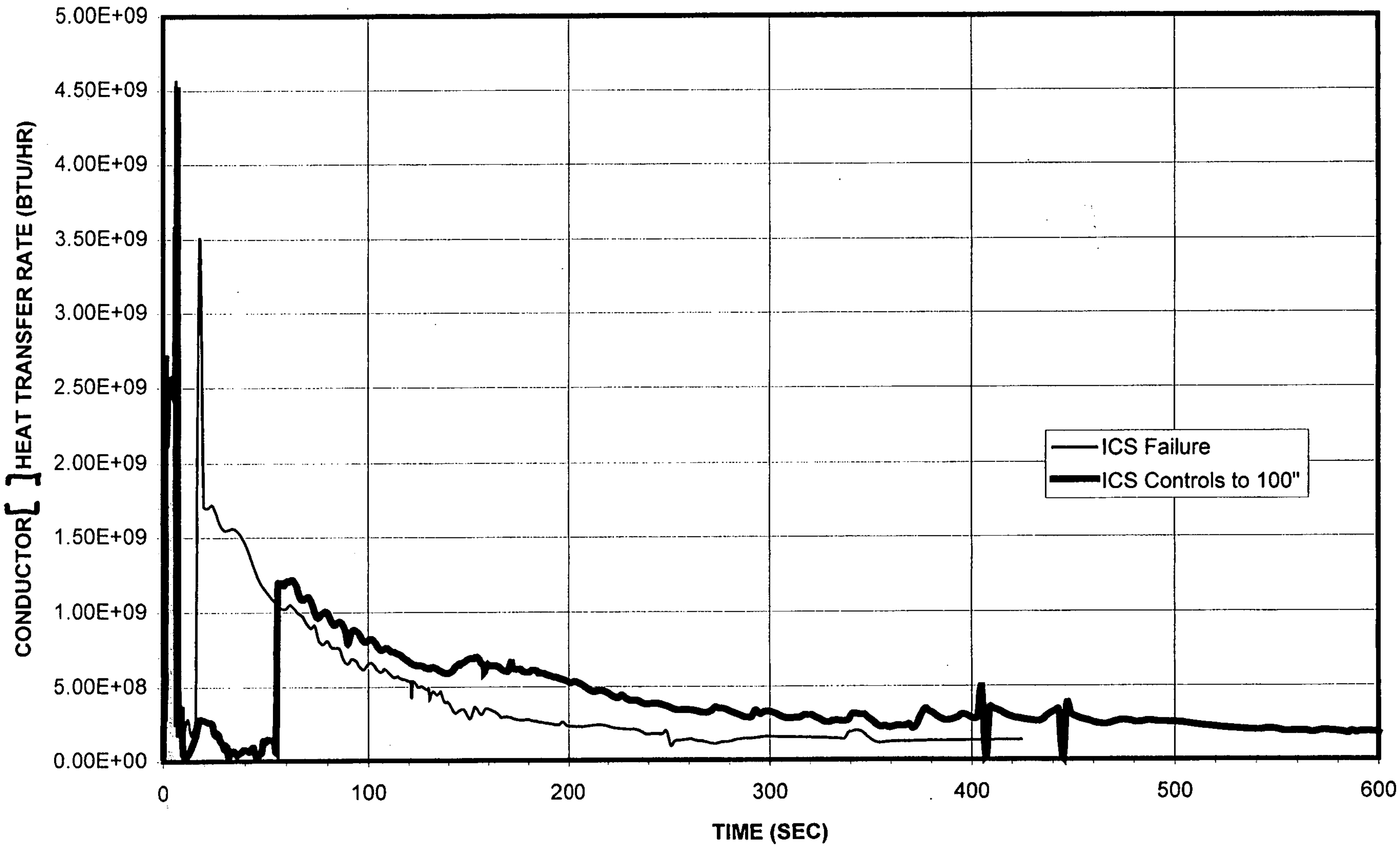

Figure 16

Duke Energy

M. S. Tuckman
Executive Vice President
Nuclear Generation

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May 5, 1999

U. S. Nuclear Regulatory Commission
Washington D. C. 20555

ATTENTION: Document Control Desk

Subject: Oconee Nuclear Station
Docket Numbers 50-269, -270, -287
UFSAR Chapter 15 Transient Analysis Methodology,
Topical Report DPC-NE-3005-P, Revision 1

References:
1) Letter, M. S. Tuckman (Duke), to NRC, July 30, 1997
2) Letter, D. E. LaBarge (NRC) to W. R. McCollum (Duke), October 1, 1998
3) Letter, M. S. Tuckman (Duke), to NRC, February 1, 1999
4) Letter, D. E. LaBarge (NRC) to M. S. Tuckman (Duke), March 17, 1999
5) Letter, M. S. Tuckman (Duke) to NRC, April 19, 1999

By means of the applicable reference documents listed above, Duke Energy Corporation has previously submitted and revised Topical Report DPC-NE-3005-P. This topical report describes the new Duke methodology for analyzing the Oconee UFSAR Chapter 15 non-LOCA transients and accidents. The NRC review of DPC-NE-3005-P, Revision 1 is currently in progress.

Based upon discussions between NRC officials and Duke representatives held on May 3, 1999 the need for an

additional change to DPC-NE-3005-P, Revision 1 was agreed upon. Therefore, in the to-be-published approved proprietary and the approved non-proprietary versions of DPC-NE-3005, Revision 1, Duke will revise Chapter 16, Page 16-1, second paragraph, as follows:

Current text: "The acceptance criteria for this analysis are to ensure that acceptable fuel damage limits are not exceeded, and that the offsite doses will be within 10% of the 10CFR100 limits."

To-be-published text: "The acceptance criteria for this analysis are that no fuel damage will occur, and that the offsite doses will be within 10% of the 10CFR100 limits."

If there are any questions or if additional information is needed on this matter , please call J. S. Warren at (704) 382-4986.

M. S. Tuckman

M. S. Tuckman

xc:

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ELL