CP&L
A Progress Energy Company

James Scarola
Vice President
Harris Nuclear Plant

SERIAL: HNP-01-078
10CFR50.4

JUN 1 1 2001

United States Nuclear Regulatory Commission
ATTENTION: Document Control Desk
Washington, DC 20555

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REGARDING THE STEAM GENERATOR REPLACEMENT AND
POWER UPRATE LICENSE AMENDMENT APPLICATIONS

Dear Sir or Madam:

By letters dated October 4, 2000 and December 14, 2000, Carolina Power & Light Company (CP&L) submitted license amendment requests to revise the Harris Nuclear Plant (HNP) Facility Operating License and Technical Specifications to support steam generator replacement and to allow operation at an uprated reactor core power level of 2900 megawatts thermal (Mwt). NRC letter dated March 27, 2001 requested additional information to support staff review of the proposed license amendment requests. The requested information is provided by the Enclosures to this letter. Please note, however, that our responses to RAI questions 4, 15, 18, 24 and 26 will be provided in a subsequent submittal.

The enclosed information is provided as a supplement to our October 4, 2000 and December 14, 2000 submittals and does not change the purpose or scope of the submittals, nor does it change our initial determinations that the proposed license amendment applications represent a no significant hazards consideration.

Please refer any questions regarding the enclosed information to Mr. Mark Ellington at (919) 362-2057.

A001

P.O. Box 165
New Hill, NC 27562

T > 919.362.2502
F > 919.362.2095

Sincerely,

James Scarola

James Scarola
Vice President
Harris Nuclear Plant

James Scarola, having been first duly sworn, did depose and say that the information contained herein is true and correct to the best of his information, knowledge, and belief, and the sources of his information are employees, contractors, and agents of CP&L.

Darlene S. Yarborough
Notary (Seal)

DARLENE S. YARBOROUGH
NOTARY
***
PUBLIC
HARNETT COUNTY, NC

My commission Expires: 2-21-2005

KWS/kws

Enclosures (5)

c: Mr. J. B. Brady, NRC Senior Resident Inspector
Mr. Mel Fry, NCDENR
Mr. R. J. Laufer, NRC Project Manager
Mr. L. A. Reyes, NRC Regional Administrator

SHEARON HARRIS NUCLEAR POWER PLANT
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
REGARDING THE POWER UPRATE LICENSE AMENDMENT REQUEST
REACTOR SYSTEMS BRANCH QUESTIONS

NRC Question 1

Provide a list of the methodologies and computer codes used in the loss-of-coolant accident (LOCA) and non-LOCA transient analysis for the steam generator replacement (SGR) and power uprate (PU) application, and reference the associated NRC acceptance letters to confirm the acceptance of the methodologies and codes used in the safety analysis for Harris Nuclear Plant (HNP). Also, provide a discussion to address the compliance with each of applicable limitations and restrictions specified in the NRC safety evaluation reports (SERs) for use of the methodologies and codes applied to the HNP SGR/PU accident analysis.

CP&L Response

Provide a list of the methodologies and computer codes used in the loss-of-coolant accident (LOCA) and non-LOCA transient analysis for the steam generator replacement (SGR) and power uprate (PU) application, . . .

A table has been prepared (see Enclosure 2) which provides a listing of the methodologies and computer codes used in the loss-of-coolant accident (LOCA) and non-LOCA transient analysis for the SGR and PU application.

Also, provide a discussion to address the compliance with each of applicable limitations and restrictions specified in the NRC safety evaluation reports (SERs) for use of the methodologies and codes applied to the HNP SGR/PU accident analysis.

Framatome-ANP (FRA-ANP, formerly Siemens Power Corporation or SPC) was the subject of an NRC Demand for Information, dated October 27, 1997 based on NRC Inspection Report 99900081/97-01. A key issue was FRA-ANP identification, control, and compliance with SER restrictions (see non-conformance 99900081/97-01-01). Response to that Demand for Information was submitted by FRA-ANP to the NRC on February 24, 1998. In January 1999, CP&L participated in an on-site assessment of FRA-ANP's follow-through of the plans and commitments outlined in their February 24, 1998 response to the NRC. The assessment was led by an independent consultant, and included representatives from several members of the FRA-ANP Fuel Users Group. One area of review by this group was the implementation of an SER commitment and restriction review process. This review confirmed that FRA-ANP had identified, clarified with the NRC, and implemented the identified methodology commitments and restrictions into FRA-ANP engineering analysis and design procedures.

CP&L specified to FRA-ANP that the methods and models used for the SGR/PUR analysis must be reviewed and approved methodologies. In general, the methods and procedures applied to the SGR/PUR analyses are the same procedures and models applied to normal reload activities. During the CP&L oversight of the SGR/PUR Chapter 15 analysis, FRA-ANP's continued implementation of long-range, programmatic upgrades from the NRC Inspection Report was monitored. Independent of the SGR/PUR licensing campaign, the normal CP&L reload fuel campaign engineering oversight continues to follow FRA-ANP analysis issues, including procedure maintenance activities which implement the FRA-ANP commitment to identify and meet SER restrictions. The calculations prepared by FRA-ANP to support the SGR/PUR project typically have a specific calculation section entitled "SER restrictions" which clearly addresses each identified methodology restriction applicable to that calculation.

NRC Question 2

Provide a list of the assumptions and ranges of initial conditions of the key plant parameters considered for each LOCA and non-LOCA transient analyzed. For example, the key parameters for the steamline break (SLB) analysis should include the initial core power level, initial core inlet temperature, initial reactor coolant system (RCS) flow rate, initial pressurizer pressure and water volume, radial peaking factor, control rod worth, initial steam generator (SG) liquid inventory, core burnup, blowdown fluid and blowdown area for each steamline. Also, confirm for each transient analyzed that the set of initial conditions used in analysis is the limiting conditions that result in the worst case.

CP&L Response

Provide a list of the assumptions and ranges of initial conditions of the key plant parameters considered for each LOCA and non-LOCA transient analyzed.

Summary tables providing assumptions and ranges of initial conditions of the key plant parameters considered for each Chapter 15 event have been extracted from a vendor report and are provided as Enclosure 3. For the numerical values corresponding to the descriptions of these inputs, refer to the Uprate Fuel Analysis Plant Parameters Document (UFAPPD), provided as Enclosure 4 (UFAPPD provided in response to staff RAI Question #3 below).

For example, the key parameters for the steamline break (SLB) analysis should include . . .

In the example cited by the staff (i.e., steam line break), the text of the vendor report, in lieu of a table, describes the inputs in general terms, and states that the inputs and biasing of these inputs were chosen to be consistent with the approved Framatome-ANP methodologies used in the analysis.

Examination of the base calculation (FRA-ANP Calculation E-6924-595-23, "Harris Power Uprate: Post-Scram Main Steam Line Break Analysis") provides a table of Hot Full Power (HFP) steady state parameters which were assumed to precede the MSLB event. From Table 7.1 of this proprietary calculation, the table provided below has been prepared:

Parameter (Post-Scram Main Steam Line Break Analysis)	ANF-RELAP Steady State Initialization Value	Target Value
Core Power, (MWt)	2900	2900
RCS Average Temperature (°F)		
T-hot	621	
T-avg	588.8	588.8
Tcold	556	557
Pressurizer Pressure (psia)	2250	2250
Pressurizer Liquid Level (%)	60	60
RCS Loop Flow Rate (lbm/sec/SG)	10130	10130
Core Sector Flow Rate (lbm/sec)		
Stuck Rod Sector	2715	2697
Affected Sector	6698	6653
Unaffected Sector	18824	18881
Core bypass Flow Rates (lbm/sec)		
Downcomer to upper head	609	608
Downcomer to outlet nozzles (total)	298	300
Core baffle region	485	486
Guide and instrument tubes	760	761
Steam Pressure (psia)		
SG-1	987	989
SG-2	985	
SG-3	987	
Steam Generator Secondary Total Mass (lbm/SG)	108474	108474
Main Steam Flow (lbm/sec/SG)		
SG-1	1192	1189.8
SG-2	1192	
SG-3	1184	
Average Fuel Temperature (°F)	1042	1037

Other parameters provided in the staff's requested list of key input parameters were incorporated into the Main Steam Line Break (MSLB) analysis, but not included as specific key inputs in a table. The blowdown fluid was conservatively modeled to be steam in accordance with the approved steam line break methodology. The blowdown area was based on the largest flow area between the affected steam generator and associated MSIV using design data for the replacement steam generators. This activity was performed during the base ANF-RELAP model development, and is not called out as a separate key input for MSLB (which uses this ANF-RELAP model). Peaking factors and control rod worths (and other reactivity parameters, like doppler and MTC) are based on the ANF-RELAP point kinetics models. These parameters are then validated (and adjustments made as necessary) by ANF-RELAP and XTGPWR comparisons.

Also, confirm for each transient analyzed that the set of initial conditions used in analysis is the limiting conditions that result in the worst case.

In general, the discussion of each event provided in the SGR/PUR Licensing Report shows that the full range of operating Modes, power levels (HZP to HFP), and (in many cases) cycle exposures (BOC and EOC) were evaluated to determine which cases specifically required evaluation. Inherent in the approved FRA-ANP methodology topical reports and incorporated into FRA-ANP's implementing procedures is the requirement that either nominal or limiting initial conditions/inputs (as allowed by the methodology topical/SER) be used in the FSAR Chapter 15 accident and transient analyses.

NRC Question 3

Provide a list of values for the input parameters that are used in the accident analysis to specifically reflect the changes of operating conditions and plant configurations for operation of HNP with the SGR and PU. Compare these values with that assumed in the existing accident analysis to identify the changes from the analysis of the record.

CP&L Response

Provide a list of values for the input parameters that are used in the accident analysis . . .

CP&L controlled the inputs to the FSAR Chapter 15 safety and transient analyses through the use of an Uprate Fuel Analysis Plant Parameters Document (UFAPPD). This document is controlled as an attachment to a plant calculation (HNP-F/NFSA-0034), the purpose of which is to generate and validate the inputs used in these analyses. A copy of the current UFAPPD is provided as Enclosure 4.

Compare these values with that assumed in the existing accident analysis to identify the changes from the analysis of the record.

The UFAPPD provides a comparison of the current (i.e., pre-SGR/PUR) parameter values to the values specified to FRA-ANP for input to the SGR/PUR analysis. With the exception of the comparison-to-previous-value feature, a very similar document is used routinely by CP&L to monitor and control inputs used for typical reload campaigns. The UFAPPD was augmented and a significant number of new values were added for SGR/PUR, but the format and use is essentially the same.

NRC Question 4

Provide a list of the systems or components that are non-safety related and credited in the accident analysis. For each of these non-safety related equipment, provide justification to show the acceptability of its use for consequence mitigation during a transient. Also, item (c)2(ii)(C) of Title 10 of the Code of Federal Regulations (10 CFR) Section 50.36 requires a technical specification (TS) for the systems or components that are used for event mitigation. Accordingly, the licensee is requested to provide the required TSs.

CP&L Response

Requested information to be provided in a subsequent submittal.

NRC Question 5

Provide a list of all the systems or components considered in determination of the single-failure events for the safety analyses. List the worst single-failure events assumed in the safety analysis for each event analyzed, and discuss the rationale of selecting the worst single-failure event for each event.

CP&L Response

The following table provides a list of the systems or components considered in determination of the single-failure events for the SGR/PUR safety analyses. As requested, the limiting single failures considered for each event analyzed and rationale for selection is also provided.

FSAR Event Designation	Event Name	Disposition	Single Failures Considered	Limiting Single Failure & Rationale
15.1	**INCREASE IN HEAT REMOVAL BY SECONDARY SYSTEM**			
15.1.1	Decrease in Feedwater Temperature	Bounded		
15.1.2	Increase in Feedwater Flow	Analyze	No single failure will prevent operation of the RPS (which terminates this event).	N/A
15.1.3	Increase in Steam Flow	Analyze	No single failure will prevent operation of the RPS (which terminates this event).	N/A
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	Bounded		

FSAR Event Designation	Event Name	Disposition	Single Failures Considered	Limiting Single Failure & Rationale
15.1.5	Steam Line Break	Analyze	Charging/HHSI pump (CSIP) failure, with one CSIP already assumed out of service for maintenance. With and without offsite power - both conditions analyzed. Failure of MSIV to close has no effect, due to location of break (EMF-84-093(P)(A), Section 2.1).	Charging/HHSI pump (CSIP) failure is limiting since it delays injection of borated water to terminate the reactivity excursion following initial shutdown. (EMF-84-093(P)(A), Section 2.3.7)
15.2	**DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM**			
15.2.1	Steam Pressure Regulator Malfunction	N/A BWR Event		
15.2.2	Loss of External Load	Bounded		
15.2.3	Turbine Trip	Analyze	No single active failure will prevent operation of any system required to function.	N/A
15.2.4	Inadvertent Closure of MSIVs	Bounded		
15.2.5	Loss of Condenser Vacuum	Bounded		

FSAR Event Designation	Event Name	Disposition	Single Failures Considered	Limiting Single Failure & Rationale
15.2.6	Loss of Non-emergency AC Power	Analyze	Failure of the available turbine driven AFW pump. Failure of an AFW isolation valve in the closed position. Analyzed with coincident Loss of Offsite Power.	Failure of the available turbine-driven AFW pump was determined, by analysis of the specific cases, to be the limiting single failure. Analyzed with coincident Loss of Offsite Power.
15.2.7	Loss of Normal Feedwater	Analyze	Same analysis as 15.2.6, but without LOOP.	With LOOP, this event and 15.2.6 are modeled the same and have the same acceptable outcome.
15.2.8	Feedline Break	Analyze	Cases with and without Offsite Power available were analyzed. Failure of the available turbine-driven AFW pump. Failure of an AFW isolation valve in the closed position.	With either offsite power available or not available, failure of TDAFW pump is the limiting scenario, as determined by specific analysis of the candidate set of conditions.
15.3	**DECREASE IN REACTOR COOLANT SYSTEM FLOW**			
15.3.1	Partial Loss of Forced Reactor Coolant Flow	Bounded		

FSAR Event Designation	Event Name	Disposition	Single Failures Considered	Limiting Single Failure & Rationale
15.3.2	Complete Loss of Forced Reactor Coolant Flow	Analyze	No single failures will adversely affect the consequences of the event.	N/A
15.3.3	RCP Shaft Seizure (Locked Rotor)	Analyze	No single active failure will adversely affect the consequences of the event.	N/A
15.3.4	RCP Shaft Break	Bounded		
15.4	**REACTIVITY AND POWER DISTRIBUTION ANOMALIES**			
15.4.1	Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition	Analyze	No single active failure will adversely affect the consequences of the event.	N/A
15.4.2	Uncontrolled RCCA Bank Withdrawal at Power	Analyze	No single active failure will adversely affect the consequences of the event.	N/A
15.4.3	RCCA Misoperation 1)Dropped Rod/Bank 2)Single Rod Withdrawal 3)Statically Misaligned RCCA	Analyze Analyze Analyze	Auto Rod Control and failure of an NI were considered 2 & 3) No single failure will prevent operation of any system required to function.	1)NI failure was determined by analysis to be the limiting single failure. 2 & 3) N/A
15.4.4	Startup of an Inactive RCP at an Incorrect Temperature	Bounded		
15.4.5	A Malfunction or Failure of the Flow Controller in a BWR Loop that Results in an Increased Reactor Coolant Flow Rate	N/A BWR Event		

FSAR Event Designation	Event Name	Disposition	Single Failures Considered	Limiting Single Failure & Rationale
15.4.6	CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant	Analyze	A boron dilution event can result from any of the following: a) Resin sluice connection to the CVCS and BTRS demineralizers b) Reactor makeup water connection to the BTRS c) Pumping water of unknown boron concentration from the recycle holdup tanks to the charging pump suction d) reactor makeup water connection to the boric acid batching tanks e) operation of the BTRS in dilution mode f) malfunction of the CVCS reactor makeup control system	The worst single active failure is malfunction of the reactor makeup water system flow indication. The flow rates assumed in the analysis satisfy this single-failure criterion.
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	Analyze	There is no single failure for this event, since no transient analysis is involved, and no protective action is required.	N/A
15.4.8	Spectrum of Rod Cluster Control Assembly Ejection Accidents	Analyze	No single failure of the reactor protection system (RPS) will negate the protection functions required for this event.	N/A

FSAR Event Designation	Event Name	Disposition	Single Failures Considered	Limiting Single Failure & Rationale
15.5	**INCREASE IN REACTOR COOLANT INVENTORY**			
15.5.1	Inadvertent Operation of the ECCS During Power Operation	SGR: Bounded by current analysis PUR: Analyzed	For SGR: Current Westinghouse analysis: No single active failure in any of these systems (FSAR Section 15.0.8 and listed in FSAR Table 15.0.8-1) will adversely affect the consequences of the accident. For PUR: New FRA-ANP analysis: For DNBR analysis, BOC and EOC cases were evaluated which assumed that the SIAS signal did NOT automatically trip the reactor. For pressurizer overfill, early reactor trip was assumed (to examine the conditions of pressurizer overfill).	Various combinations of conditions were assumed to cause this event to challenge specific criteria. Auto Rod Control was modeled as needed to make the event worse. Some RPS protective trips were disabled to ensure that the event would proceed to a challenge, instead of being a "null event." Finally, the reliable, but non-safety PORVs were assumed to initially relieve, then close as motive air is assumed depleted, in order to challenge the pressurizer safeties during the Pressurizer Overfill portion of the analysis.

FSAR Event Designation	Event Name	Disposition	Single Failures Considered	Limiting Single Failure & Rationale
15.5.2	CVCS Malfunction that Increases RCS Inventory	Bounded		
15.6	**DECREASE IN REACTOR COOLANT INVENTORY**			
15.6.1	Inadvertent Opening of a Pressurizer Safety or PORV	Analyze	No single active failure will adversely affect the consequences of the event.	N/A
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	Radiological event only		
15.6.3	Radiological Consequences of Steam Generator Tube Rupture	Analyze	Discussed in detail in 6.3.1.2 of the Licensing Report	Discussed in detail in 6.3.1.2 of the Licensing Report
15.6.4	Radiological Consequences of a Main Steam Line Failure Outside Containment	N/A (BWR event)		
15.6.5	Loss of Coolant Accidents 1) Large Break LOCA	Analyze	Loss of a diesel generator Loss of Low Head Safety Injection pump Analyzed with a coincident Loss of offsite power (LOOP)	By analysis of both candidate single failures, the Loss of LHSI was determined to yield the limiting PCT.

FSAR Event Designation	Event Name	Disposition	Single Failures Considered	Limiting Single Failure & Rationale
	2) Small Break LOCA	Analyze	Loss of a diesel generator Analyzed with a coincident Loss of offsite power (LOOP)	This was determined to be the limiting failure in consideration of this leading to a) a HHSI capability based on only one pump, and b) an auxiliary feedwater capability based on only one motor driven pump.
15.7	**RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT**			
15.7.1	Radioactive Waste Gas System Leak or Failure	Radiological event only	N/A	N/A
15.7.2	Liquid Waste System Leak or Failure	Radiological event only	N/A	N/A
15.7.3	Postulated Radioactive Releases Due to Liquid Tank Failure	Radiological event only	N/A	N/A
15.7.4	Fuel Handling Accidents	Radiological event only	N/A	N/A
15.7.5	Spent Fuel Cask Drop Accidents	Not affected by SGR/PUR	N/A	N/A

NRC Question 6

General Design Criterion (GDC) 17 in 10 CFR Part 50, Appendix A requires, in part, that "An onsite electric power system and an offsite power system shall be provided to permit functioning of structures, systems, and components import to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of AOOs [anticipated operational occurrences] and (2) the core is cooled and containment and other vital function are maintained in the event of postulated accidents." In accordance with the GDC 17 requirements, a loss-of-offsite-power (LOOP) must not be considered as a single-failure event and must be assumed as part of the event initiation in the analysis for each AOO and accident without changing the event category. The staff finds that the safety analyses in Section 6.2* do not address the LOOP effects for a majority of the transients. The licensee is requested to identify the events that do not assume an LOOP in the analyses as part of event initiation and analyze these events with an LOOP to comply with the GDC 17 requirements with respect to the LOOP assumption for the safety analysis. Submit the results of the requested analysis for the staff to review and approve.

CP&L Response

The Standard Review Plan (SRP), NUREG-0800, is very prescriptive as to which events require consideration of a LOOP. The SRP provides the applicable GCD requirements as the basis for the SRP acceptance criteria for the Chapter 15 events. GDC 17, however, is not listed as the basis for the acceptance criteria in any of the Chapter 15 events. In addition, included within the SRP acceptance criteria is the requirement that an incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently. For Condition II events, a LOOP would indeed constitute an independent fault. For this reason, it is appropriate to evaluate incidents of moderate frequency independent of a LOOP, since the SRP acceptance criteria specifically requires evaluation of the event to determine that the event by itself and without any other fault occurring independently does not generate a more serious plant condition.

NRC Question 7

The licensee submitted a new large break (LB) LOCA analysis on December 14, 2000 to replace the original LBLOCA analysis submitted on October 4, 2000. The staff finds that in both new and original analyses the same analytical methods were used and same ranges of power level, steam generator tube plugging and reactor vessel average coolant temperatures were assumed. The results of two sets of LBLOCA are different in that the limiting break and the peak clad temperature (PCT) are changed. The licensee is requested to compare the new and original LBLOCA analyses and identify any changes of the methods and assumptions that result in a different limiting case and PCT.

CP&L Response

The licensee is requested to compare the new and original LBLOCA analyses . . .

FRA-ANP identified an error in the calculation of normalized axial power shape in the original LBLOCA analysis submitted on October 4, 2000. Thus, the new LBLOCA analysis was submitted on December 14, 2000. There were no changes to methods or assumptions, only an input data error correction. Specifically, an incorrect "actinide decay" flag was set in the input deck for the analysis which generated the results of the analysis submitted to the NRC on October 4, 2000. Subsequent to completion of the work for the October submittal, this error was identified, investigated, and corrected. The corrected results were submitted to the NRC as a replacement Section 6.1.1 of the Licensing Report in the December 14, 2000 Power Uprate submittal.

For HNP Cycle 10 (pre-uprate, current operating cycle), the error correction is being tracked and managed under CP&L's 10CFR50.46 PCT tracking procedure. The error correction resulted in a change to PCT of 9°F (decrease), so it did not require immediate reporting.

. . . and identify any changes of the methods and assumptions that result in a different limiting case and PCT

The effect of the incorrect "actinide decay" flag was an improperly modeled axial power shape in the SEM/PWR-98 normalized power calculation. When the correct axial shape was used in the revised calculation, the PCT for all break sizes and most reactivity conditions (BOC, MOC, or EOC) were affected (all changes were reductions) to varying degrees. The worst-case condition for the original LBLOCA analysis (i.e., MOC, 1.0 DECLG) submitted on October 4, 2000 experienced a 16 degree F decrease, while the (originally) non-limiting MOC, 0.8 DECLG was only reduced by 2 degrees F. The net result was that, since the PCT results of the two cases were so close (within 8 degrees F) to begin with, use of the correct axial power shape shifted the limiting event from the MOC 1.0 DECLG break to the MOC 0.8 DECLG break.

NRC Question 8

Section 6.1.2* presents the results of analysis for the small break (SB) LOCA events. The events are assumed to be initiated from full-power (Mode 1) conditions. No discussion is provided to address the SBLOCA (a decreased RCS inventory event) effects for the plant operating at low power modes or shutdown conditions. For the operations other than Mode 1, a decreased RCS inventory event may occur due to inadvertent opening of valves or inadvertent RCS drainage. Under the low power and shutdown conditions, the available safety systems for accident mitigation are limited as compared to the Mode 1 conditions. The licensee is requested to address the effects of RCS drainage events initiated from the conditions of Modes 2 through 6.

CP&L Response

Appendix K of 10 CFR 50 specifies required and acceptable features of the LOCA evaluation models, including the following: " . . . *the reactor has been operating continuously at a power level at least 1.02 times the licensed power level* . . ."
This initial power level assumption is a requirement of 10 CFR 50 Appendix K as denoted by the use of the word "shall" in the regulation. Furthermore, neither 10 CFR 50 Appendix K nor 10 CFR 50.46 includes a requirement to analyze SBLOCA events by assuming initial conditions other than full-power (Mode 1) conditions.

The requirements and conservatisms applied to the evaluation of SBLOCA for PUR as prescribed by Appendix K are consistent with the current HNP analysis of record. Also, NUREG –0800 Section 15.6.5 specifies the same parameters and assumptions for analysis of SBLOCA events for staff review.

NRC Question 9

Sections 6.1.3 and 4* discuss analyses of post-LOCA long-term core cooling and hot-leg switchover (HLSO). The licensee is requested to confirm that the hydraulic and boron mixing models used in the analyses are acceptable for the licensing applications. Also, compare the calculated range of the hot-leg switchover time with that specified in the post-LOCA HLSO procedure to show that the calculated HLSO time is appropriately reflected in the HLSO procedure.

CP&L Response

The licensee is requested to confirm that the hydraulic and boron mixing models used in the analyses are acceptable for the licensing applications.

Westinghouse performed the post-LOCA Long Term Core Cooling (LTCC) and Hot Leg Switchover (HLSO) analyses for HNP SGR/PUR conditions. During the Inputs, Assumptions and Methods (IAM) review phase of the SGR/PUR analysis, the existing HNP licensing and analysis basis was examined to determine both the appropriate methods and the acceptance criteria to apply to this work. The following references were considered in developing the models and their acceptance criteria for these calculations:

1. Westinghouse Technical Bulletin NSID-TB-86-08, "Post-LOCA Long-Term Cooling: Boron Requirements", October 31, 1986.

2. WCAP-8339, "Westinghouse Emergency Core Cooling System Evaluation Model - Summary", June 1974.

3. Westinghouse Nuclear Safety Advisory Letter NSAL-94-016, Rev. 1, "Core Recriticality During Hot Leg Switchover", August 30, 1999.

4. Westinghouse Letter CLC-NS-309, C. L. Caso (W) to T. M. Novak (NRC), April 1, 1975.

5. Westinghouse Letter JOC-NS-364, J. O. Cermak (W) to T. M. Novak (NRC), July 23, 1975.

6. Westinghouse Nuclear Safety Advisory Letter NSAL-92-010, "Hot Leg Switchover Methodology", January 9, 1993.

7. NRC Information Notice 93-66: SWITCHOVER TO HOT-LEG INJECTION FOLLOWING A LOSS-OF-COOLANT ACCIDENT IN PRESSURIZED WATER REACTORS, August 16, 1993.

In the above-listed references, a variety of conservatisms in the modeling and choice of inputs were identified for different aspects of the overall issue. For SGR/PUR, the three following aspects of the issue were evaluated separately:

1. Maximum allowable switchover time (to prevent boron precipitation),
2. Minimum allowable switchover time (to ensure that sufficient injection flow is available), and
3. Mixed Mean Sump Boron Concentration (for criticality control)

By choosing conservative inputs, and by building upon the cited reference history of these issues, HNP is assured that the methods and models are both appropriate to this licensing application, and are consistent with the existing HNP licensing and analysis basis.

Also, compare the calculated range of the hot-leg switchover time with that specified in the post-LOCA HLSO procedure to show that the calculated HLSO time is appropriately reflected in the HLSO procedure.

The results of the analyses show that a conservative acceptable range of switchover times from 3.0 hours to 8.5 hours will meet each of the acceptance criteria identified in the cited reference documents. The current HLSO time of 6.5 hours documented in FSAR Section 6.3.2.5.2.3 and Plant Operating Manual Procedure EOP-EPP-011, "Transfer Between Cold Leg and Hot Leg Recirculation," Revision 10 remains within the switchover time range provided by these SGR/PUR specific calculations.

NRC Question 10

Section 6.1.5* indicates that for the control rod ejection event, no fuel is expected to fail due to either departure from nucleate boiling (DNB) or fuel melting, and the over-pressurization consequences are bounded by those of the turbine trip (TT) event. The licensee is requested to provide (1) a figure showing the calculated DNB ratios (DNBRs) during the transient, (2) a figure showing the calculated temperatures for the hot spot fuel centerline and outer fuel clad during the transient, and (3) an analysis showing that the overpressurization consequences are limited by those of the TT event.

CP&L Response

The licensee is requested to provide (1) a figure showing the calculated DNB ratios (DNBRs) during the transient, . . .

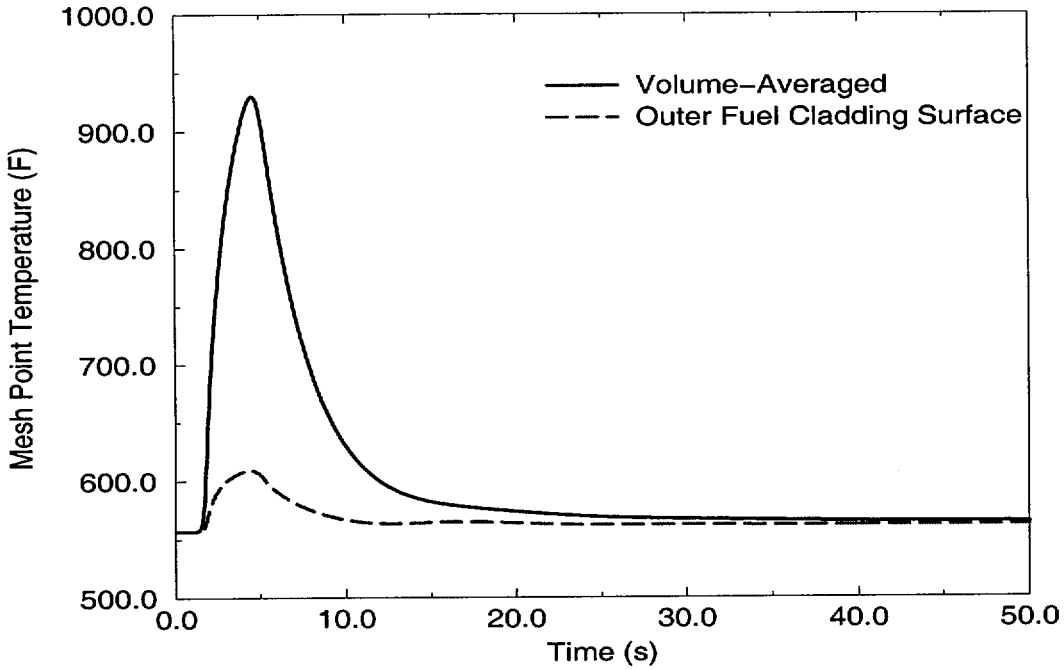
CP&L response to staff RAI question #13 provides tables (extracted from proprietary calculation) of XCOBRA-IIIC calculated DNBRs around the times of the limiting MDNBR for various conditions. These tables are included as Enclosure 6.

The licensee is requested to provide . . . (2) a figure showing the calculated temperatures for the hot spot fuel centerline and outer fuel clad during the transient, . . .

FRA-ANP calculated the hot spot fuel centerline temperature using the peak volume averaged fuel temperature for the average fuel rod. The figure provided on the following page shows the volume-averaged fuel temperature along with the outer cladding surface temperature at the highest-powered average core node from the ANF-RELAP transient model for the limiting (with respect to centerline melt) HZP EOC case. To account for the increased F_Q resulting from the ejected control rod, the hot spot centerline temperature is calculated by multiplying the volume averaged fuel enthalpy rise (converted from the temperature rise) for the average rod by a post-rod ejection F_Q calculated using FRA-ANP's approved neutronics methodology. The F_Q adjusted enthalpy rise is added to the initial fuel enthalpy and converted to a temperature. The resulting temperature is the hot spot fuel centerline temperature.

The licensee is requested to provide . . . (3) an analysis showing that the overpressurization consequences are limited by those of the TT event.

An overpressure analysis for a control rod ejection was performed by FRA-ANP. This analysis was biased to maximize the challenge to the RCS overpressure limit (e.g., pressurizer PORVs and sprays were not modeled). The results of this calculation confirmed that the turbine trip RCS overpressure case indeed bounds that for a control rod ejection. The peak RCS pressure for the turbine trip event was over 100 psi higher than for a control rod ejection.



Volume Averaged Fuel and Outer Cladding Surface Temperatures for a Control Rod Ejection Accident (HZP, EOC)

NRC Question 11

The licensee classifies the design-basis events according to their anticipated frequency of occurrence identified as Condition 1 normal operation and operational transients; Condition II faults of moderate frequency; Condition III infrequent faults; and Condition IV limiting faults. In Table 6.2.0-1*, the licensee lists the design-basis events evaluated or analyzed under Conditions II, III and IV. The staff finds that the classification of these events is generally consistent with the guidance of Standard Review Plan (SRP) and current licensing practices. However, the complete loss of forced reactor coolant flow and single rod withdrawal event are listed in the Table as a Condition III infrequent fault. This event categorization is inconsistent with SRP 15.3.1 and 15.4.3 that classify the complete loss of forced reactor coolant flow event and the single rod withdrawal event as Condition II events (faults of moderate frequency) with the acceptance criteria that require the DNBR to not exceed the specified limit. The licensee is requested to justify the deviation from the SRP related to the event classification of the complete loss forced reactor coolant flow and the single rod withdrawal events and address the adequacy of the results of analysis for these events.

CP&L Response

The section of the Standard Review Plan (NUREG-0800) described above is entitled, "15.3.1-15.3.2 Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions." In the HNP FSAR, there are two separate sections: (1) Section 15.3.1 Partial Loss of Forced Reactor Coolant Flow, and (2) Section 15.3.2 Complete Loss of Forced Reactor Coolant Flow.

Complete loss of flow is described in FSAR Table 15.0.1-1, Accident Category Used for Each Chapter 15 Event as a Condition III Postulated Accident. The description of the accident analysis and its results is covered in FSAR Section 15.3.2.

For Partial Loss of Forced Reactor Coolant Flow (FSAR Section 15.3.1), both the FSAR Table 15.0.1-1 and the Licensing Report Table 6.2.0-1 show that for HNP the Section 15.3.1 event is a Condition II Anticipated Operating Occurrence event. The table further states in the "Bounding Event or Reference" column that this event is to be bounded by the Complete Loss of Forced Reactor Coolant Flow, only if the bounding Condition III event results meet the more restrictive (no challenge to the SAFDLs) Condition II acceptance criteria. In Licensing Report Section 6.2.14.4, the text echoes the Table 6.2.0-1 requirement that the Partial Loss of Forced Reactor Coolant Flow event consequences are bounded by the Complete Loss of Forced Reactor Coolant Flow event only if the Complete loss event is held to Condition II acceptance criteria (see submittal Section 6.2.15.3 for confirmation that the Complete loss event is helped to this more restrictive requirement). Therefore, the analysis of these events (Complete Loss Of Forced Reactor Coolant Flow, 15.3.1 and Partial Loss of Forced Reactor Coolant Flow, 15.3.2) does not deviate from the event classification in the HNP FSAR Section 15.0.1.

For the Single Rod Withdrawal Event, the current HNP FSAR Section 15.0.1 defines this event to be a Condition III event. ANSI N18.2 also lists "inadvertent removal of a single neutron absorbing control rod during refueling such that Safety Limits, as defined in the Technical Specifications, would be exceeded" as a Condition III event, which is consistent with the FSAR and Licensing Report Section 6.2 classification of a Single Rod Withdrawal Event. The SGR/PUR event classification is unchanged from this current licensing basis determination in the HNP FSAR. Since the current licensing basis event classification was unchanged, no discussion of the "exception" to SRP 15.4.3 was provided.

The classification of the Complete Loss of Forced Reactor Coolant Flow (Section 15.3.2) and the Single Rod Withdrawal Event (Section 15.4.3) as Condition III Infrequent Faults remains unchanged in the HNP FSAR. Furthermore, as noted above, SGR/PUR does not change the HNP position on either event classification or analysis which determines that the partial loss of forced reactor coolant flow (Section 15.3.1) is bounded by the complete loss of forced reactor coolant flow event (15.3.2). This is consistent with current HNP licensing basis in this respect. Second, Sections 15.3.1 and 15.3.2 are presented together in one single section of the SRP, entitled "Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions." In the "combined" SRP Section 15.3.1 – 15.3.2, the specific acceptance criteria that must be met to meet the requirements of GDC 10, 15, and 26 is provided, but these criteria are not presented exclusively for incidents of moderate frequency. GDCs 10, 15, and 26 specify particular features required to protect design conditions during normal operation, including anticipated operational occurrences (AOO). Furthermore, a footnote is provided in SRP 15.3.1 – 15.3.2 to clarify that the term "moderate frequency" is used in the same sense as in the descriptions of design and plant process conditions as documented in ANSI N18.2. ANSI N18.2, however, lists examples of Condition II occurrences that include a partial loss or interruption of core coolant flow excluding reactor coolant pump locked rotor (i.e., 15.3.1), while the examples of Condition III events listed in ANSI N18.2 include a complete loss or interruption of core cooling pumped flow, excluding reactor coolant pump locked rotor (i.e., 15.3.2), which is consistent with the classification of these events in both the current HNP FSAR and Licensing Report Section 6.2 of the October 4, 2000 submittal.

In summation, the HNP SGR/PUR introduces no change to current HNP licensing basis with regard to Chapter 15 event classification for FSAR Sections 15.3.1, 15.3.2, and 15.4.3. Event classification for Sections 15.3.1 and 15.3.2 is consistent with ANSI N18.2 in which it is stated: "an inadvertent removal of a single neutron absorbing control rod during refueling such that Safety Limits, as defined in the Technical Specifications, would be exceeded" as a Condition III event, which is also consistent with the HNP FSAR and Licensing Report Section 6.2 classification of a single rod withdrawal as a Condition III event.

NRC Question 12

Provide a list of the setpoints with the associated uncertainties for normal operation and the setpoints assumed in the transient analysis for engineered safety feature actuation systems, pressurizer safety valves, power-operated relief valves (PORVs) and SG safety valves. Compare these analytical values with the applicable TS values and address the acceptability of the TS values.

CP&L Response

ESFAS Trip Functions:

A listing of SGR/PUR values (assumed in transient analyses) of Tech Spec Table 3.3-4 engineered safety features actuation system [ESFAS] trip setpoints and Total Allowance [TA] values, corresponding safety analyses limits [SALs], and applicable channel uncertainties (in terms of channel statistical allowances [CSAs]) are shown below. These values have been denoted with reference to corresponding Tech Spec Table 3.3-4 Functional Unit Numbers and ESFAS functional descriptions.

TS Trip Setpoint	SAL	TS Total Allowance [TA]	CSA Uncertainty
1.c -- Safety Injection on High Containment Pressure			
≤ 3.0 psig	5.0 psig (Note 1)	3.64% Span [(2psig / 55psig) x 100%]	2.89% Span
1.d -- Safety Injection on Low Pressurizer Pressure			
≥ 1850 psig	1700 psig	18.75% Span [(150psig / 800psig) x 100%]	12.11% Span
1.e -- Safety Injection on Low Steam Line Pressure			
≥ 601 psig	542.2 psig (Note 2)	4.52% Span [(58.8psig / 1300psig) x 100%]	3.21% Span
2.c -- Containment Spray on High-High-High Containment Pressure			
≤ 10.0 psig	12.0 psig	3.64% Span [(2psig / 55psig) x 100%]	2.89% Span
4.d -- Main Steam Line Isolation on Low Steam Line Pressure			
≥ 601 psig	542.2 psig (Note 2)	4.52% Span [(58.8psig / 1300psig) x 100%]	3.21% Span
5.b -- Turbine Trip & Main Feedwater Isolation on High-High SG Level			
≤ 78 % NR Level	100 % NR Level	22.0% Span [(22.0%Lvl / 100%Lvl) x 100%]	9.87% Span

TS Trip Setpoint	SAL	TS Total Allowance [TA]	CSA Uncertainty
6.c -- Auxiliary Feedwater on Low-Low SG Level (Main Feedwater Line Break)			
≥ 25 % NR Level	0 % NR Level	25.0% Span (Note 3) [(25.0%Lvl / 100%Lvl) x 100%]	18.67% Span (Note 3)
6.c -- Auxiliary Feedwater on Low-Low SG Level (Loss of Normal Feedwater)			
≥ 25 % NR Level	16.1 % NR Level	8.9% Span (Note 3) [(8.9%Lvl / 100%Lvl) x 100%]	7.17% Span (Note 3)
6.g -- Auxiliary Feedwater Isolation on High Steam Line Differential Pressure, coincident with Steam Line Isolation			
≤ 100 psig	165 psig	5.0% Span [(65psig / 1300psig) x 100%]	4.52% Span

Note 1: Final safety analyses utilized 5.0 psig, in lieu of original UFAPPD specification (4.5 psig).

Note 2: SAL denoted is the value used within Westinghouse high energy line break mass & energy release analyses. FRA-ANP reload fuel analyses use a conservative SAL value of 370.9 psig (as originally specifed in UFAPPD, Table 2-18).

Note 3: TAs and CSAs are shown for both normal operation and for MFW line break conditions, as noted with Tech Spec functional description. Tech Spec Table 3.3-4 TA reflects worst-case uncertainty condition [with MFW line break environmental allowance] and a conservatively chosen SAL [for level below the top of SG tubes].

The TA is defined by Tech Spec Bases as the difference, in percent span, between the SAL and the trip setpoint; each bracketed (percent span) TA computation shows this |TS – SAL| difference relative to the channel's 100% Span value (expressed in engineering units). The CSA is the statistical combination of measurement uncertainty components applicable to an instrument channel between its sensor and trip bistable.

The acceptability of the above-listed Tech Spec trip setpoint values is demonstrated based upon a direct comparison of each trip setpoint against its corresponding SAL; i.e., after accounting for applicable channel measurement uncertainties (CSA), the trip setpoint is enveloped (in terms of relative direction/magnitude, given its particular ESFAS function) by its corresponding SAL value.

This conclusion can be further demonstrated by application of the acceptance criteria contained in NSSS Licensing Report Subsection 6.7.3 and in HNP current licensing/design bases; i.e., the available TA exceeds the applicable CSA for each ESFAS function, given the availability of positive operating margin between respective TA and CSA values.

Calculation HNP-I / INST-1010, which includes proposed Tech Spec Table 3.3-4 changes, has been submitted in response to the April 12, 2001 NRC RAI (reference: CP&L response by letter SERIAL: HNP-01-081, dated May 18, 2001).

Except for the High Containment Pressure SAL (per Note 1 above), the above-specified trip setpoint and SAL values are delineated within UFAPPD, Table 2.18 (ESF Actuation Setpoints).

Safety/Relief Valve Functions:

Nominal setpoint values and associated uncertainties, assumed for normal operation and for purposes of safety analyses, applicable to pressurizer safety valves, PORVs, and steam generator safety valves have also been summarized below. Where multiple setpoints are listed, the worst-case condition to be modeled for a specific safety analysis case is assumed. Furthermore, the listed (+/-) tolerances have been conservatively applied to safety analyses inputs as specified in Enclosure 2. In addition, valve capacities are modeled in safety analyses based upon total capacities summarized in FSAR Table 15.0.3-5.

TS Trip Setpoint Requirement	**Safety Analyses Value**
Tech Spec 3.4.2.2 -- Pressurizer Safety Valves	
2485 psig ± 1% (Lift Setting); (3) total valves	2485 psig +2%** / -1% (Open Setpoint); 2335 psig (Close Setpoint) ** -- +2% uncertainty includes a +1% TS tolerance and a 1% set pressure shift.
Pressurizer Power Operated Relief Valves [PORVs]	
(1) Compensated PORV: 100 psid (2) Noncompensated PORVs: 2335 psig Note: Above setpoints not specified in TS; refer to FSAR Table 15.0.3-5.	Compensated PORV: 100 psid (Open Setpoint); 80 psid (Close Setpoint) Uncompensated PORVs: 2335 psig (Open Setpoint); 2315 psig (Close Setpoint)
Tech Spec Table 3.7-2 -- Main Steam Safety Valves	
Safety Valves (1 per SG loop at each setting), with respective Lift Settings: 1170 psig ± 1%; 1185 psig ± 1%; 1200 psig ± 1%; 1215 psig ± 1%; 1230 psig ± 1%.	Safety Valves, with the following respective Open Setpoints: 1170 psig ± 1%; 1185 psig ± 1%; 1200 psig ± 1%; 1215 psig ± 1%; 1230 psig ± 1%.

The acceptability of Tech Spec values is demonstrated since each safety analysis value, with its associated uncertainty, is equal to (or greater than, in the case of Pressurizer Safety Valves) its respective Tech Spec requirement.

These safety analyses values are contained in Table 2.8 (Pressurizer Safety Valves), Table 2.9 (Pressurizer Control), and Table 2.13 (Main Steam System Safety Valves) of the UFAPPD (see Enclosure 4).

NRC Question 13

As stated in the SRP, one of the acceptance criteria for the transient analysis is related to the calculated DNBRs. The staff finds that the analyses in Section 6.2* do not provide calculated DNBRs during a transient for most of the events. The licensee is requested to list the events that result in a decrease in DNBRs and provide figures for these events to show the calculated DNBRs during transients. For cases (such as the locked rotor event) that are predicted to result in fuel rod damage because of the low calculated DNBRs, the licensee is requested to discuss the methods and input assumptions (such as pin census data and peak factors) used to determine the percentage of the damaged fuel rods and confirm the acceptance of the calculational methods and results.

CP&L Response

Provided below is a list of the requested DNBR tables and plots, which are provided in Enclosure 5:

15.1.2 DNBR plots attached (ANF-RELAP, Not XCOBRA-IIIC) from calc for this event (Increase in Feedwater Flow)

15.1.3 No plots for this event (Increase in Secondary Steam Flow). XCOBRA-IIIC run at minimum ANF-RELAP indicated DNBR condition.

15.1.5 No plots for this event (Steam Line Break). XCOBRA-IIIC run at peak post-scram power indicated by ANF-RELAP.

15.2.3 DNBR plots attached (ANF-RELAP, not XCOBRA-IIIC) from calc for this event (Turbine Trip).

15.2.6 No plots for this event (Loss of Normal AC). Same analysis as 15.2.7 (Loss of Normal Feedwater) by choice of inputs.

15.2.7 No plots for this event (Loss of Normal Feedwater). Bounded by 15.2.3 (Turbine Trip) for MDNBR considerations.

15.2.8 No plots for this event (Feedwater Line Break). This is an over-pressurization event, not a DNBR challenging event.

15.3.2 DNBR plot attached (ANF-RELAP, not XCOBRA-IIIC) from calc for this event (Complete Loss of Reactor Coolant Flow)

15.3.3 DNBR plots attached (both ANF-RELAP and XCOBRA-IIIC) from calc for this event (Locked Rotor).

To determine the number of fuel failures, having determined that the worst-case assembly DNBR was less than the 95/95 HTP correlation limit, the first assumption is that all rods in such an affected assembly would fail. Next, a series of XCOBRA-IIIC computer runs were executed with reduced limiting peaking radial peaking factors to determine the maximum allowable peaking such that the hot assembly MDNBR just meets the 95/95 HTP correlation limit. The percentage of core-wide fuel failure was determined by assuming that all fuel assemblies with a radial peaking greater than this maximum allowable peaking will fail. Examination of the SGR/PUR "equilibrium cycle" full core radial peaking factors showed that only one assembly (the hot assembly) exceeded the maximum allowable peaking determined from the process above. However, to provide flexibility for future reloads, the failed fuel assembly will be assumed to be subject to octagonal symmetry, and a total of 8 assemblies will be reported to fail. Dividing 8 by 157, the result is the reported 5.096% (5.1%) fuel failure due to DNBR.

15.4.1 DNBR Table attached (from XCOBRA-IIIC) as well as ANF-RELAP T/H condition plots from calculation for this event (Uncontrolled RCCA Withdrawal at low power)

15.4.2 DNBR Plot attached (deterministic, not statistical) from the Engineering report for this event (Uncontrolled RCCA Withdrawal at Power).

The deterministic results were evaluated separately by the approved, FRA-ANP Statistical Setpoint/Transient Methodology (EMF-92-081(P)(A), Revision 1 accepted by NRC letter dated February 10, 2000) to determine the MDNBR result "of record," which met acceptance criteria for no fuel failures.

15.4.3 No DNBR plots for this event (RCCA misoperation).

The deterministic results for Dropped Rod were evaluated separately by the approved, FRA-ANP Statistical Setpoint/Transient Methodology (EMF-92-081(P)(A), Revision 1, accepted by NRC letter dated February 10, 2000) to determine the MDNBR result "of record", which met acceptance criteria for no fuel failures. Single rod withdrawal was evaluated statistically after modifying the 15.4.2 bank withdrawal results by an event specific peaking augmentation factor. Thus the plot for event 15.4.2 would nominally apply to this event for the general shape of the curve. The statically mis-aligned rod case is, by definition, a static case, so no plot is provided.

Regarding the fuel failure prediction for the single rod withdrawal, a similar process as described above for the Locked Rotor event was applied. An iterative set of XCOBRA-IIIC calculations was performed. The peaking on the limiting assembly was lowered until the 95/95 HTP correlation limit was reached. It was found that only the limiting assembly was peaked higher than the maximum allowed peaking factor thus determined. Therefore, only this assembly fails as a consequence of this event. Dividing 1 by 157 yields the 0.64% reported fuel failure rate. Unlike the locked rotor, which is a core-wide event, this single rod withdrawal event is local in nature, so there was no need to apply octant symmetry and report failure of a larger than calculated number of assemblies.

15.4.6 No plots for this specific event (CVCS malfunction causing boron dilution). The Disposition of Events declared this event's system responses to be bounded by the Event 15.4.2 (uncontrolled RCCA withdrawal), so that event's plot (see Event 15.4.2 plots described above) of deterministic DNBR can be considered a bounding DNBR trend for this event.

15.4.7 No plots for this event (Inadvertent Loading And Operation Of A Fuel Assembly In An Improper Location). This is a static event, so no plots are provided.

For a normal reload cycle, this event would be analyzed to predict fuel failures based on peaking factors expected to be experienced by the Misload Event. For this special case of the SGR/PUR project, the event was analyzed to generate the maximum allowable measured F-Delta-H (for DNBR failure) and F-Q (for centerline melt) that would provide a "threshold" beyond which fuel failures might be predicted. Coupled with the fuel failure assumption in the Dose Consequences section 2.22 of the BOP Licensing Report, this provides a basis to evaluate each subsequent actual reload calculation's peaking factor results predicted for that cycle, against the peaking factors used in the SGR/PUR misloaded core analyses to set the threshold for fuel failure. Should a particular fuel cycle generate a peaking factor result that exceeds this threshold, the amount of fuel failure (determined in a similar manner as described above for Locked Rotor and/or single rod withdrawal) would be compared to the "dose consequence analysis of record" fuel failure assumption for this event.

15.4.8 No plots provided for this event (RCCA Ejection Accident).

Tables of XCOBRA-IIIC calculated DNBRs around the times of the limiting MDNBR for various conditions) are provided.

15.5.1 A pre-Amendment 45 copy of the DNBR plot in FSAR Figure 15.5.1-1 is provided for this event (Inadvertent Operation of ECCS During Power Operation). For SGR-Only (the October 4, 2000 submittal) the submittal concludes that the existing analysis of record remains bounding and applicable. Because of various dispositions and minor revisions of this original analysis, the analysis DNBR and other plotted results were removed from the HNP FSAR in Amendment 45.

For Power Uprate, a DNBR plot was provided in Licensing Report Figure 6.2.26-6, enclosed to the December 14, 2000 submittal (not resubmitted herein) for this event (Inadvertent Operation of ECCS During Power Operation).

15.6.1 No plot provided for this event (Inadvertent Opening of Pressurizer Safety or Relief Valve). Table is provided in Enclosure 5.

Table 7.8 from the FRA-ANP calculation is provided to show the XCOBRA-IIIC DNBR trend in the region of minimum DNBR occurrence.

NRC Question 14

Section 6.2.2.* states that the increased feedwater flow event is analyzed to ensure that "adequate margin to SAFDLs [specified acceptable fuel design limits] is maintained, and that protection against steam generator overfill is maintained." The staff finds that no figure is presented to show that the calculated DNBRs do not exceed the SAFDLs during the transient. The licensee is requested to provide the figure showing the calculated DNBRs. (This request is applied to all the transients that result in decreased DNBRs - see RAI 11.) The staff also finds that no sufficient information is presented for the assumptions used in the analysis to address the SG overfill issue. Specifically, for the case initiated from a full opening of a feedwater isolation valves without the isolation valve reclosure because of a single failure consideration, the licensee is requested to identify the safety related equipment that are credible to isolate the feedwater in order to prevent SG overfill. If the licensee needs to credit non-safety related systems or components (such as the feedwater control valves or feedwater pumps) to isolate or terminate the feedwater, the licensee should show that the non-safety related system or component is reliable for feedwater isolation and provide a TS limiting condition for operation (LCO) to meet the requirements specified in (c)2(ii)(C) of 10 CFR 50.36. (This request is applied to all the transients that credited the non-safety related equipment for consequence mitigation - see RAI 4.)

CP&L Response

The licensee is requested to provide the figure showing the calculated DNBRs (for the increased feedwater flow event) . . .

See previous question #13 (DNBR plot provided in Enclosure 6 for event 15.1.2, increased feedwater flow).

. . . the licensee is requested to identify the safety related equipment that are credible to isolate the feedwater in order to prevent SG overfill.

Should the sequence of events proposed by the NRC occur, HNP credits the main feedwater flow control and flow control bypass valves to be available to manually isolate and terminate the feedwater. This is consistent with the current licensing basis and the HNP SER (NUREG-1038). The following discussion is extracted from the HNP Design Basis Document for Condensate, Main Feedwater, Condensate Polishers, Feedwater Drains and Vents Systems (DBD-112, Rev 14):

> The main feedwater flow control valves and flow control bypass valves are Seismic Category I, Quality Group D. These valves are classified as Quality Group D since they are contained in influent lines and are capable of being isolated from the reactor coolant pressure boundary by an additional valve which has high leak tight integrity (e.g., MFIV) (Ref. Reg. Guide 1.26). These valves are connected to Non-seismic Category I piping in the Turbine Building and serve as redundant feedwater line isolation designed to fail in the closed position, which per the SER, serves

as an acceptable backup. The fact that the main feedwater isolation valves are designed and installed to seismic Category I, Quality Group B, to ensure feedwater isolation in an accident situation, satisfies the requirements GDC 2 and Position C.1 of Reg. Guide 1.29.

The feedwater control valves and the feedwater control bypass valves are not installed in safety grade piping, but they are important to safety because they act as a backup to the main feed isolation valves during a steam line rupture (FSAR 15.1.5.1c). These valves are not expected to remain functional during and/or after a seismic event since failures of secondary system piping and earthquakes are not required by the NRC to occur simultaneously with each other; i.e., loss of non-safety equipment due to an SSE event is not assumed coincidental with a spontaneous steam line break accident. Reliance on the non-safety grade valves in the postulated accident evaluation (FSAR 15.1.5.1c) is permitted based on the reliability of these valves. The rationale for dependence on these "non-safety grade" feedwater control valves is that they are high quality components since they are built to ASME Section III, Class 3, Seismic Category I requirements. Based on the above discussion and information as contained in NUREG-0138,
"Staff Discussion Memorandum from Director, NRR to NRR Staff," the feedwater control valves and feedwater bypass control valves are not required to be seismically qualified in their installed condition.

Therefore, should the event occur as proposed by the NRC, HNP has adequate assurance that highly reliable alternative means of isolating feedwater will be available and effective in preventing overfill. This confidence is reflected in the analytical methods and procedures applied to this event for Chapter 15 accident analysis. Full opening of FW isolation valve (without the isolation valve reclosure due to a single failure consideration) is not considered to be applicable to this event (per FRA-ANP's Analysis Guidelines).

. . . the licensee should show that the non-safety related system or component is reliable for feedwater isolation and provide a TS limiting condition for operation (LCO) to meet the requirements specified in (c)2(ii)(C) of 10 CFR 50.36.

Note that item (c)2(iii) of 10CFR50.36 states, in part, the following:

"A licensee is not required to propose to modify technical specifications that are included in any license issued before August 18, 1995 to satisfy the criteria in paragraph (c)2(ii) of this section."
Accordingly, modifications to the HNP Technical Specifications to satisfy the criteria in paragraph (c)2(ii) of 10CFR50.36 are not required, since the HNP Operating License was issued prior to August 18, 1995.

NRC Question 15

Section 6.2.3* states that for the increased steam flow event, two cases are analyzed: one for minimum neutronics feedback (beginning-of-cycle (BOC) conditions) and the other for maximum neutronics feedback (end-of-cycle (EOC) conditions). Both cases are evaluated with automatic rod control. The licensee is requested to provide an analysis to show that the cases with automatic rod control are more limiting than the cases without automatic rod control. Also, provide the values of the moderator temperature and Doppler feedback coefficients assumed in the analysis for the BOC and EOC cores and confirm that the analytical values are bounded by the TS values.

CP&L Response

Requested information to be provided in a subsequent submittal.

NRC Question 16

Section 6.2.5* includes a discussion of the SLB analysis. The licensee states that the previously NRC-approved methodology (EMF-84-093) was used to perform the SLB events. The staff recognizes that the referenced SLB methodology was generically approved by the NRC. The staff also notes that the limitations of use of the methodology were identified in the NRC SER. The licensee should discuss the values used for the input parameters in the SLB analysis for HNP applications and confirm that it complies with the limitations for the input parameters or assumptions related to worst stuck control element assembly assumption, moderator reactivity coefficient, break size and location, blowdown fluid quality, single failure consideration, auxiliary feedwater flow and temperature. (The compliance with SER restrictions is applied to all the transient - see RAI 1.)

CP&L Response

The response to staff RAI question #2 provides the input parameters used for the SLB event. HNP SGR/PUR specific input was developed and used in accordance with the EMF-84-093(P)(A) approved methodology and included SER (see response to staff RAI question #1 for specific identification of these references). The SER identifies the following conservative input assumptions:

1. control rod with maximum worth stuck in a fully withdrawn position to maximize radial peaking,
2. end-of-cycle core conditions to yield the most negative moderator coefficient,
3. steam is blown down through the break to maximize the cooldown,
4. break occurs at the location that maximizes the break area (upstream of the MSIV), and
5. limiting single failure is one of two HHSI pumps to delay the transport of boron to the core.

The analyses supporting the HNP SGR/PUR have incorporated each of these conservative input assumptions. In addition, the effects of auxiliary feedwater were treated conservatively by assuming all available flow being directed to the faulted steam generator at a minimum temperature. This conservative treatment of the auxiliary feedwater results in a maximum cooldown of the RCS and ignores the isolation logic (i.e., feed-only-intact steam generator) at HNP.

The input assumptions and restrictions identified in the SER are implemented into the analyses by FRA-ANP's controlling work practices and analysis guidelines.

NRC Question 17

Table 6.2.8-4* lists the allowable high flux trip setpoints as a function of the number of inoperable main steam safety valves (MSSVs). The trip setpoints were credited in the analysis for transient such as the turbine trip event. In accordance with the requirements of item (c)2(ii)(C) of 10 CFR 50.36, The licensee is requested to provide a TS LCO for these high flux trip setpoints. (The licensee states (on page Enclosure 1-23 to a letter dated October 4, 2000) that proposed TS 3.7.1.1 includes the revised maximum power range neutron flux high setpoint with inoperable MSSVs. The proposed TS 3.7.1.1. is not available for the staff to review.)

CP&L Response

The licensee is requested to provide a TS LCO for these high flux trip setpoints.

Note that item (c)2(iii) of 10CFR50.36 states the following:

"A licensee is not required to propose to modify technical specifications that are included in any license issued before August 18, 1995 to satisfy the criteria in paragraph (c)2(ii) of this section."

Accordingly, modifications to the HNP Technical Specifications to satisfy the criteria in paragraph (c)2(ii) of 10CFR50.36 are not required, since the HNP Operating License was issued prior to August 18, 1995.

The proposed TS 3.7.1.1. is not available for the staff to review.

The proposed revision to Tech Spec 3.7.1.1 (TS Table 3.7-1, TS page 3/4 7-2) was provided in Enclosure 5 of the October 4, 2000 submittal.

NRC Question 18

Section 6.2.13* presents a discussion of the feedwater line break (FLB) accident analysis. The results show that for both FLB cases with and without offsite power available, the pressurizer becomes solid during the event. The safety relief valves are assumed to repeatedly open and close for an extended period of time in the water blowdown environment. TMI action Item II.D.1 requires that all RCS safety, relief, and blocked valves be tested to confirm the valve operability under expected operating conditions for design-basis transients and accidents. Accordingly, the licensee is requested to provide analysis or test data, or reference the NRC approval letter to show that (1) the safety relief valves (SRVs) can be operable (opening and closing on demand) under the water environment, and (2) the SRVs are reliable for repeated opening and closing during a transient for an extended period of time. Also, confirm that the value of initial pressurizer water level used in the pressurizer-overfill analysis maximizes the calculated pressurizer water level and is conservative as compared to the TS value.

CP&L Response

Requested information to be provided in a subsequent submittal.

NRC Question 19

Section 6.2.15* indicates that for the complete loss of forced reactor coolant flow event, two cases were analyzed: 100 percent power with a moderator temperature coefficient (MTC) of 0.0 pcm/ °F and 70 percent with MTC of +5.0 pcm/ °F. The licensee is requested to compare the analytical values of MTC with the TS values and confirm that the analytical values of MTC are bounded by the TS values.

Table 6.2.15-1*, Event Summary for Loss of Forced Reactor Coolant Flow, indicates that pressurizer PORV is credited in the transient analysis for the event initiated from the full-power conditions. The licensee should reference a TS LCO for the PORV to show its compliance with item (c)2(ii)(C) of 10 CFR 50.36 requirements. (This RAI is applied to transient analysis for all the cases - see RAI 4.)

CP&L Response

For HNP, the MTC limits have been relocated to the Core Operating Limits Report (COLR), which is controlled by Harris Plant Procedure PLP-106. The analytical values specified are, in fact, the positive MTC limits specified in the COLR. Therefore, the analysis is confirmed to bound the range of COLR allowable conditions.

Licensing Report Table 6.2.15-1 provides the sequence of events for an analysis that determines the minimum DNBR for the Loss of Forced Reactor Coolant Flow. The pressurizer PORVs are assumed to be operable since the PORVs minimize the increase in Reactor Coolant Pressure. The minimum DNBR decreases with decreasing pressure.

NRC Question 20

Section 6.2.19*, Uncontrolled Rod Cluster Control Assembly [RCCA] Withdrawal at Power, indicates that a sensitivity study is performed to evaluate the effects of power level, reactivity insertion rate and reactivity feedback on the results of the transient. The licensee is requested to list all the cases (specifying initial conditions of power level, reactivity insertion rate and reactivity feedback representing the BOC and EOC cores) that are analyzed. Summarize the results of analyses and show that the limiting case is the full-power case at EOC core conditions with reactivity insertion rate of 27.6 pcm/sec as presented in section 6.2.19*.

CP&L Response

In Licensing Report Section 6.2.19.2*, it is stated that power levels of 60% and 100% were considered. Also stated is that reactivity insertion rates up to 100 pcm/sec were considered. BOC cases with minimum reactivity feedback (positive MTC and least negative Doppler) and EOC cases with maximum feedback (large negative MTC and most negative Doppler). From the FRA-ANP calculation, the following table provides a case list summary including the specific numerical values considered:

Limiting Cases	DNB Result Plot (XCOBRA-IIIC)	Initial Power (% of 2900 MWt)	Reactivity Insertion Rate (pcm/sec)	BOC		EOC	
				MTC	Doppler	MTC	Doppler
1) BOC, 0.73 pcm/sec	See Plot "Figure 10.1" from FRA-ANP calculation. MDNBR: 1.153	102	a) 15 cases between 0 and 1 pcm/sec b) 9 cases between 1 and 10 pcm/sec c) 8 cases between 10 and 30 pcm/sec	+5.0	-0.8	-50.0	-2.76

Limiting Cases	DNB Result Plot (XCOBRA-IIIC)	Initial Power (% of 2900 MWt)	Reactivity Insertion Rate (pcm/sec)	BOC MTC	BOC Doppler	EOC MTC	EOC Doppler
2) EOC, 27.6 and 28.0 pcm/sec	See Plot "Figure 10.2" from FRA-ANP calculation MDNBR 1.143	102	a) 4 cases between 1 and 20 pcm/sec b) 12 cases between 20 and 30 pcm/sec c) 5 cases between 30 and 100 pcm/sec	+5.0	-0.8	-50.0	-2.76
3) BOC, 2.9 pcm/sec	See Plot "Figure 10.3" from FRA-ANP calculation MDNBR 1.198	60	a) 13 cases between 0 and 4 pcm/sec b) 4 cases between 4 and 20 pcm/sec c) 2 cases between 20 and 50 pcm/sec	+5.0	-0.8	-50.0	-2.76
4) EOC, 92.0 pcm/sec	See Plot "Figure 10.4" from FRA-ANP calculation MDNBR 1.193	60	a) 8 cases between 10 and 80 pcm/sec b) 6 cases between 80 and 90 pcm/sec c) 5 cases between 90 and 100 pcm/sec	+5.0	-0.8	-50.0	-2.76

NRC Question 21

Section 6.2.20* indicates that the calculated minimum DNBR for the withdrawal of single full-length RCCA event is less than the safety limit. As a result, a total of one assembly (0.64 percent of total fuel rods) is predicted to fail. The licensee is requested to provide a figure showing the calculated DNBRs during the transient. Discuss the analytical methods, input parameters and assumptions used to determine the number of failed fuel rods, and show that the methods used for the analysis are acceptable and the input parameters and assumptions are conservative with respect to the fuel failure calculations.

CP&L Response

The withdrawal of a single full-length RCCA event was analyzed as follows: The system response is essentially the same as the Uncontrolled Bank Withdrawal event at power (Event 15.4.2). The single RCCA withdrawal is distinguished from the withdrawal of an RCCA bank by a severe radial power redistribution. Therefore, the limiting system response from the Bank Withdrawal Event 15.4.2 (determined by the ANF-RELAP code, and identified to be limiting by deterministic evaluation of DNBR by separate XCOBRA-IIIC calculations) was evaluated. A radial power peaking factor on F-Delta-H, specific to the augmented peaking associated with the single RCCA withdrawal event, is included in the MDNBR calculation to account for radial power redistribution effects typical of this event.

Since the system response and the general progression of the event are derived from the Bank withdrawal, Licensing Report Figure 6.2.19-1 provides a "figure showing the calculated DNBRs" during this transient. The application of augmented peaking (for the single RCCA withdrawal) and other DNBR significant system variables (power, temperature, etc) at the minimum DNBR conditions shown on that Figure 6.2.19-1 yielded the prediction that MDNBR would be less than the safety limit in one fuel assembly. Conservatively, every rod in that fuel assembly was therefore assumed to fail, yielding the 0.64 (1 divided by 157 total assemblies in the core) percent of total fuel rod failure prediction.

NRC Question 22

Section 6.2.23* states that for the inadvertent boron dilution event, the analysis shows that there is adequate time for the operator to manually terminate the source of dilute flow during all modes of operation. However, no information is provided for the method used and the assumptions made in the analysis. The licensee is requested to confirm that (1) the method (especially, the boron mixing model applying to the condition without the reactor coolant pump running) used for the boron dilution analysis is acceptable, (2) the initial RCS water volumes and dilution flow rates assumed in the analysis for each mode of operation are conservative with respect to the calculated operator action time to terminate the diluted water flow, and (3) the staff's concern regarding nonconservative inputs for the deboration event analysis documented in NRC Information Notice 93-32 is satisfactorily addressed.

CP&L Response

Confirm the method used for the boron dilution analysis is acceptable . . .

The current licensing basis methodology for HNP is the "instantaneous mixing model" (see Section 15.4.6 of NUREG-1038, "Safety Evaluation Report Related to the Shearon Harris Nuclear Power Plant Units 1 and 2 [SER]"). This methodology models the RCS and RHR as a mixing volume, and assumes immediate continuous dilution of the entire volume, which produces an exponential decrease in the boron concentration. It is recognized that this cannot occur since the dilution flow requires a finite time to mix throughout the volume, and this time is dependent upon the actual values of RCS loop flow, RHR flow and dilution flow. The present and proposed SGR/PUR analysis basis for HNP is provided by the current fuel vendor, FRA-ANP. The FRA-ANP methodology for analyzing the boron dilution for PWRs (ANF-84-73 Revision 5 Appendix B(P)(A)) states:

> For operation of the shutdown cooling system (SDCS), primary coolant flow rates may be insufficient to assure a completely mixed primary coolant volume. If a completely mixed coolant volume cannot be assumed, then a dilution front approach is employed to evaluate the adequacy of the margin to criticality.

The alternate Dilution Front Model methodology has been considered, but not implemented for the HNP SGR/PUR.

CP&L reviewed some recent comparisons of the perfect mixing model's predictions to scale model dilution tests conducted at the LOFT facility for conditions equivalent to 3000 gpm RHR flow and 300 gpm dilution flow, and concluded that the model is a reasonable approximation of the actual boron transient. The data show that instantaneous mixing model may be non-conservative in predicting core boron concentration early in the transient, but that the model becomes more accurate and conservative as time passes.

The HNP flux doubling alarm licensing basis (for Mode 5, the Mode with the lowest mixing volume) puts the time of criticality, relative to start of the dilution event, well beyond the 15 minute time period when the instantaneous mixing model may be non-conservative in predicting core boron concentration. With the flux doubling alarm requirement, criticality occurs in the 30 to 45 minute time frame. Data evaluated during a recent review of this modeling assumption shows that the perfect mixing model is a very good representation of the LOFT experimental data in this time period. Mode 4 and other potentially limiting conditions are bounded by this lowest mixing volume Mode 5 condition.

Further, the referenced LOFT studies specifically evaluated the instantaneous mixing model predictions against the experimental results and found that the predicted time to criticality was about 30% conservative. This was explained as due to dilution of stagnant volumes that are conservatively neglected in the instantaneous mixing model. This study stated that "the functional form of the data is very well described by the perfect mixing model."

Based on these reviews of the Harris licensing basis, and the overall model performance of the current fuel vendor (FRA-ANP) models against LOFT benchmarks, CP&L concludes that the perfect mixing boron dilution methods continue to be acceptable to perform conservative analyses of this event.

FRA-ANP has sent an interim 10 CFR Part 21 evaluation to the NRC ("Interim Report of Evaluation of a Deviation Pursuant to 10 CFR 21.21(a)(2)," NRC:01:017, April 27, 2001) dealing with the ANF-84-73 Revision 5 Appendix B(P)(A)) methodology's potentially inadequate definition of the approach used to determine which of the two models should be applied for modes with RHR cooling. The interim evaluation specifically continues to support the use of the instantaneous mixing model for current analyses (including HNP SGR/PUR). The report further states that the criterion used to draw this conclusion is being further evaluated to determine its adequacy.

Confirm that the initial RCS water volumes and dilution flow rates assumed in the analysis for each mode of operation are conservative with respect to the calculated operator action time to terminate the diluted water flow . . .

Specific analyses were performed for Mode 1 through 5 to determine if the acceptance criteria for the event (that criticality would not occur within the 15 minutes required for operator action to identify and terminate the dilution event) would be met. See Table 2.17 in the UFAPPD (Enclosure 3 to this RAI response) for the RCS volumes and flows considered. The minimum time required to reach criticality occurred in Mode 1 (full power), and was 16.6 minutes.

For Mode 6, no analysis was performed, since administrative controls are in place at the Harris plant to preclude a boron dilution from occurring during a refueling operation. These controls require that all valves that connect to systems that may inject unborated water to be locked in the closed position.

Confirm that the staff's concern regarding non-conservative inputs for the deboration event analysis documented in NRC Information Notice 93-32 is satisfactorily addressed . . .

IN 93-32 specifically addressed generic inverse count rate ratio (ICRR) plots and non-conservative (lack of) modeling of uncertainties in Boron Dilution Mitigation Systems (BDMS) neutron monitoring system, which uses a flux doubling setpoint to trigger an automatic mitigation of the dilution event. The Harris plant does not have an automatic BDMS, but does use a flux doubling alarm to initiate the 15 minute operator action time in Mode 5.

Investigation of this issue in 1992 and 1993 at Harris determined that using the current actual plant ICRR plots did not lead to a violation of the commitment in the SER (NUREG-1038) to maintain 15 minutes from alarm to criticality to allow for operator action. Corrective actions (including specifically accounting for the curvature of the ICRR plots and including uncertainty in the analysis of Shutdown Margin requirements) have been implemented to prevent an occurrence of the concerns that initiated the IN 93-32.

NRC Question 23

Section 6.2.26* submitted on December14, 2000, presents the results of analysis for the RCS inventory increase event resulting from inadvertent operation of the emergency core cooling system (ECCS). The event is initiated from inadvertent actuation of charging pumps. In the HNP plant, the ECCS contains the safety injection (SI) pumps, safety injection tanks (SITs) and charging pumps that inject water into the RCS. Inadvertent operations of any of these ECC subsystems may increase RCS inventory. The licensee should expand the discussion of Section 6.2.26 to address the effects of the inadvertent operations of the SI pumps and SITs on the increased RCS inventory event.

CP&L Response

At HNP, the charging pumps also function as the (High Head) Safety Injection pumps. Since this does not represent a separate injection path or event, Section 6.2.26 need not be expanded. If any change is necessary, a re-iteration of definitions previously provided could be included.

At HNP, the safety injection tanks are referred to as the Engineered Safety Feature (ESF) Accumulators. The IOECCS event 15.5.1 is defined in the FSAR as IOECCS at Power. At power conditions, the RCS pressure exceeds the nitrogen blanket pressure/motive force available to the ESF Accumulators (665 psig maximum), so this injection path at power is not credible.

Similarly, the Low Head Safety Injection pumps (i.e., the RHR pumps at HNP), have discharge pressures (below 200 psig) that are significantly below the at-power operating pressure of 2250 psia. This injection path is not credible at power.

NRC Question 24

Section 6.2.27*, CVCS [chemical and volume control system] Malfunction that Increases Reactor Coolant Inventory, states that the potential for water relief through the pressurizer through safety valves is addressed in Event 15.5.1 and the challenge to SAFDL is addressed in Event 15.4.6. The staff notes that the Event 15.5.1 is the inadvertent operation of the ECCS during power operation and Event 15.4.6 is the CVCS malfunction that results in a decrease in the boron concentration event. The referenced cases are caused by different initiators, need different safety systems to mitigate the consequences, may result in different system and thermal-hydraulic responses, and have different safety concerns. The licensee should provide a technical basis to justify that the increased reactor coolant inventory event due to CVCS malfunction is adequately represented by the analysis for Event 15.5.1 and Event 15.4.6, or provide the results of analysis for this event for the staff to review. Also, the licensee states that for Modes 4 through 6, at least one pressurizer PORV (or vent) is available for pressure relief. The licensee should reference the TS for PORVs to satisfy the requirements specified in item (c)2(ii)(C) of 10 CFR 50.36.

CP&L Response

Requested information to be provided in a subsequent submittal.

NRC Question 25

Section 6.3.1* states that the major operator actions for SG tube rupture (SGTR) recovery provided in the licensee's EOPs Path-2 are explicitly modeled in the SG overfill analysis. Provide justification to show that the licensee's STGR recovery procedures are acceptable for determining the operation actions and the associated action times.

CP&L Response

Section 6.3.1 of the HNP SGR submittal states that the SGTR analyses were performed using the analysis methodology developed in WCAP-10698 and Supplement 1 to WCAP-10698. The methodology was developed by the SGTR Subgroup of the Westinghouse Owner's Group (WOG) and was approved by the NRC in SERs dated December 17, 1985 and March 30, 1987. The operator actions assumed are consistent with the WOG SGTR methodology. Simulator tests indicate that the revised operator action times are appropriate.

NRC Question 26

Section 6.3.2* states that the block valve downsteam of the PORV is credited in the analysis to isolate the PORV from the SG with a ruptured tube. The licensee is requested to discuss the reliability of the block valves to function under the expected transient conditions and address the acceptability of the valves for the accident mitigation. Also, provide a TS LCO for the block valves to satisfy the requirements specified in item (c)2(ii)(C) of 10 CFR 50.36.

As stated in Section 6.3.2*, the licensee determines that an operator can locally close the block valve for the PORV on the affected SG within 20 minutes following the SG PORV failure in the open position. The licensee is requested to discuss the method used to determine the action time for the operator to close the block valve and show that the method is acceptable and the proposed action time of 20 minutes is available.

CP&L Response

Requested information to be provided in a subsequent submittal.

NRC Question 27

The licensee's uprated power and steam generator replacement application will increase the operating power limit by 4.5 percent and change the SG heat capacity. The changes in the SG design and operating conditions may result in changes to the setpoint of the low temperature overpressure protection (LTOP) system. The licensee is requested to provide a setpoint analysis for the LTOP system in accordance with guidance specified in Section II.B of SRP 5.2.2 and show that either the current LTOP setpoint remains valid or propose a new setpoint with an asssociated TS.

CP&L Response

A description of existing analyses and evaluations for the Low Temperature Overpressure Protection System (LTOPS), also known as Cold Overpressure Mitigation Systems (COMS), for current plant operation is described in HNP FSAR section 5.2.2.11, "RCS Operation During Low Temperature Operation." The design bases mass input and heat input events used in the analyses are also listed in HNP Technical Specification (TS) Bases 3/4 .4.9 for "Low Temperature Overpressure Protection."

The existing LTOPS Mass Input transients for current plant operation are based on a conservative hand calculation with plant specific parameters. The existing LTOPS Heat Input transients for current plant operation are based on a conservative 1977 generic Westinghouse Owners Group (WOG) model.

For the HNP SGR/PUR project, these analyses were revised. Westinghouse performed a transient analysis to determine the Reactor Coolant System (RCS) pressure overshoot effects during an LTOPS event. This analysis is briefly described in Section 4.3.1.5 of the Licensing Report (Enclosure 6 of licensee letter dated October 4, 2000). The heat capacity of both the RCS and SGs are changed for SGR/PUR. The SG heat transfer surface area has increased, which potentially increases the RCS peak pressure during a LTOPS heat input event. In addition, the RCS volume has also increased.

For the HNP SGR/PUR, the design bases mass input and heat input events used in the analyses were the same as those listed in the current HNP Technical Specification (TS) Bases 3/4 .4.9 for "Low Temperature Overpressure Protection". A Westinghouse plant specific model was created with plant specific parameters and an LTOPS analysis was performed to determine the RCS peak pressure overshoot for a variety of setpoints, mass input flowrates and RCS temperatures. The use of a plant specific model and input parameters assisted in minimizing some of the overly conservative aspects of the older WOG model. The revised model and the increase in RCS volume have mitigated the potentially adverse impact of an increased SG heat transfer surface area. The results of the Westinghouse LTOPS / COMS analyses were reported in terms of RCS pressure overshoot (pressure increase above the LTOPS setpoint) for a range of setpoints.

The description of analyses and evaluations performed for Reactor Vessel Integrity (RVI), including LTOPS setpoints, is in section 5.1.2 of the LR. Specifically, section 5.1.2.1 acknowledges the potential impacts to RVI created by the SGR/PUR project. Section 5.1.2.2 (last paragraph) of the LR briefly describes the analysis that was performed for the determination of the LTOPS setpoints. The Westinghouse pressure overshoot results were considered, along with other plant features, such as dynamic pressure drop, instrumentation locations and uncertainties, etc, to determine the actual RCS peak pressure during a postulated LTOP event for specific LTOPS setpoints. The enable (arming) temperature and margin to the RCS Pressure-Temperature (P-T) limits were also evaluated. Section 5.1.2.4 (last paragraph) of the Licensing Report provides a discussion of the results of the LTOPS setpoint analysis which indicates that no changes to the setpoints or enable temperature were required since the peak pressure during a design basis LTOPS event, assuming the current setpoints remained below the current HNP TS RCS P-T limits issued in TS Amendment # 100.

NRC Question 28

Page Enclosure 1-11 to a letter dated October 4, 2000, provides a basis to support the proposed TS change for the required water volume in the boric acid tank. It states that "Based on the analysis results, the minimum contained volume during shutdown specified for the boric acid tank in Technical Specification 3.1.2.5 is increased from 6650 to 7150." Discuss the referenced analysis used to draw the above conclusion on water volume in boric acid tank and show that both the analytical method and results are acceptable.

CP&L Response

The analysis to support SGR and PUR Technical Specification 3.1.2.5 began with an equilibrium fuel loading pattern, design boron concentrations and fixed poison assumptions. From the equilibrium core, design shutdown boron concentration requirements were calculated for beginning-of-life (BOL) and end-of-life (EOL). Boron concentration requirements were translated into boration injection volume requirements, based on maintaining current refueling water storage tank (RWST) and boric acid tank (BAT) minimum boron concentration requirements and increasing the reactor coolant system (RCS) volume. Increasing the RCS volume directly impacts the boration/dilution process proportional to the increase in RCS volume. The increase in RCS volume is directly attributed to replacing the D4 steam generators with the Delta 75 steam generators. The injection volume requirements were converted into tank volumes using recently calculated uncertainty values. In each step, values were conservatively rounded to provide additional margin.

NRC Question 29

Item G of Branch Technical Position (BTP) RSB 5.1 requires that a seismic Category I auxiliary feedwater supply be provided with sufficient inventory to permit operation at hot shutdown conditions for least 4 hours, followed by a cooldown to the condition permitting operation of the residual heat removal system. The auxiliary feedwater needed for the cooldown shall be based on the longest coldown time needed with either onsite or offsite power available and with the worst single failure. The licensee is requested to provide a discussion to address its compliance with the requirements specified in item G of BTP RSB 5-1. The discussion should include information to show that the analytical models and methods are acceptable; the assumptions used are consistent with the BTP RSB 5-1 and are conserative to maximize the required auxiliary feedwater supply; and the analytical results are bounded by the TS values for the auxiliary feedwater supply.

Since the auxiliary feedwater supply is credited for event mitigation, a TS is required for the auxiliary feedwater system to specify the required water volume (item (c)2(ii)(C) of 10 CFR 50.36). The licensee indicates (on page Enclosure 1-25 to a letter dated October 4, 2000) that TS Bases 3.7.1.3 specifies the volume of 270,000 gallons for the condensate storage tank that is the primary source of supply for auxiliary feedwater system. According to 10CFR 50.36, the Bases for specifications are not part of the TS. Therefore, the staff determines that TS Bases 3.7.1.3 is not a TS and it alone is not adequate to satisfy the TS requirements for the auxiliary feedwater water supply.

CP&L Response

As described in the HNP FSAR Section 1.8 compliance to Regulatory Guide 1.139, "Guidance For Residual Heat Removal", paragraph 1, "Shearon Harris is a Class 2 plant as defined by the implementation section of BTP RSB 5-1. The safe shutdown basis is hot standby. Thus, Shearon Harris does not fully comply with the functional requirements of Regulatory Guide 1.139." Also, HNP FSAR section 1.8, paragraph 1.2 states that " A sufficient Seismic Category I supply of deaerated auxiliary feedwater to permit four hours operation at hot standby plus cooldown to Residual Heat Removal System initiation conditions is provided by the condensate storage tank. A back up Seismic Category I source for the Auxiliary Feedwater System is the station Service Water System.

In addition, HNP FSAR Section 5.4.7.2.8 provides a cold shutdown methodology per the requirements of SRP 5.4.7 (RSB BTP 5-1). More specifically, FSAR section 5.4.7.2.8.2.a.(1), fifth paragraph, states that "Condensate Storage Tank - Upon depletion of the primary source of auxiliary feedwater in the seismic Category I condensate storage tank, a backup source of auxiliary feedwater is the UHS via either train of the SWS."

For SGR/PUR, the above FSAR statements remain applicable. Therefore, compliance with item G of BTP RSB 5.1 is not required as permitted by Table 1 of BTP RSB 5-1.

Notwithstanding the above statements, HNP Technical Specification (TS) 3.7.1.3 does in fact, already specify a condensate storage tank (CST) to be operable with a contained volume of at least 270,000 gallons of water, equivalent to a 62% indicated level. It is the Bases for this Technical Specification that is being proposed for revision to reflect that this CST water volume will support hot standby for 6 hours versus the current 12 hours hot standby.

Listing of Methodologies and Computer Codes Used for Chapter 15 Accident and Transient Analyses

FSAR Event Designation	Event Name	Methodology Topical Reports / NRC Acceptance Letters	Computer Codes used in Methodologies
15.1	**INCREASE IN HEAT REMOVAL BY SECONDARY SYSTEM**		
15.1.1	Decrease in Feedwater Temperature	1. EMF-89-151(P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992. Acceptance letter from NRC dated March 16, 1992 included in (P)(A) version of report. 2. XN-NF-82-21(P)(A) Revision 1, Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, Exxon Nuclear Company, September 1983. Acceptance letter from NRC dated August 26, 1983 included in (P)(A) version of report. 3. XN-75-21(P)(A) Revision 2, XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation, Exxon Nuclear Company, January 1986. Acceptance letter from NRC dated March 13, 1985 included in (P)(A) version of report.	1. ANF-RELAP 2. XCOBRA-IIIC

FSAR Event Designation	Event Name	Methodology Topical Reports / NRC Acceptance Letters	Computer Codes used in Methodologies
15.1.2	Increase in Feedwater Flow	1. EMF-89-151(P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992. Acceptance letter from NRC dated March 16, 1992 included in (P)(A) version of report. 2. XN-NF-82-21(P)(A) Revision 1, Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, Exxon Nuclear Company, September 1983. Acceptance letter from NRC dated August 26, 1983 included in (P)(A) version of report. 3. XN-75-21(P)(A) Revision 2, XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation, Exxon Nuclear Company, January 1986. Acceptance letter from NRC dated March 13, 1985 included in (P)(A) version of report.	1. ANF-RELAP 2. XCOBRA-IIIC

FSAR Event Designation	Event Name	Methodology Topical Reports / NRC Acceptance Letters	Computer Codes used in Methodologies
15.1.3	Increase in Steam Flow	1. EMF-89-151(P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992. Acceptance letter from NRC dated March 16, 1992 included in (P)(A) version of report. 2. XN-NF-82-21(P)(A) Revision 1, Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, Exxon Nuclear Company, September 1983. Acceptance letter from NRC dated August 26, 1983 included in (P)(A) version of report. 3. XN-75-21(P)(A) Revision 2, XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation, Exxon Nuclear Company, January 1986. Acceptance letter from NRC dated March 13, 1985 included in (P)(A) version of report.	1. ANF-RELAP 2. XCOBRA-IIIC
15.1.4	Inadvertent Opening of a Steam Generator Relief or Safety Valve	None	

FSAR Event Designation	Event Name	Methodology Topical Reports / NRC Acceptance Letters	Computer Codes used in Methodologies
15.1.5	Steam Line Break	1. EMF-84-093 (P)(A), Revision 1, Steam Line Break Methodology for PWR's, February, 1999. Acceptance letter from NRC dated February 16, 1999 included in the (P)(A) version of the report. 2. XN-CC-28(A) Revision 3, XTG: A Two Group Three-Dimensional Reactor Simulator Utilizing Coarse Mesh Spacing, Exxon Nuclear Company, September 1975. Acceptance letter from NRC dated September 27, 1975 included in the (A) version of the report. 3. XN-NF-82-21 (P)(A) Revision 1, Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, Exxon Nuclear Company, September 1983. Acceptance letter from NRC dated August 26, 1983 included in (P)(A) version of report. 4. XN-75-21(P)(A) Revision 2, XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation, Exxon Nuclear Company, January 1986. cceptance letter from NRC dated March 13, 1985 included in (P)(A) version of report.	1. ANF-RELAP 2. XCOBRA-IIIC 3. XTG

FSAR Event Designation	Event Name	Methodology Topical Reports / NRC Acceptance Letters	Computer Codes used in Methodologies
15.2	**DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM**		
15.2.1	Steam Pressure Regulator Malfunction	N/A	
15.2.2	Loss of External Load	None	
15.2.3	Turbine Trip	1. EMF-89-151(P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992. Acceptance letter from NRC dated March 16, 1992 included in (P)(A) version of report. 2. XN-NF-82-21(P)(A) Revision 1, Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, Exxon Nuclear Company, September 1983. Acceptance letter from NRC dated August 26, 1983 included in (P)(A) version of report. 3. XN-75-21(P)(A) Revision 2, XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation, Exxon Nuclear Company, January 1986. Acceptance letter from NRC dated March 13, 1985 included in (P)(A) version of report.	1. ANF-RELAP 2. XCOBRA-IIIC
15.2.4	Inadvertent Closure of MSIV's	None	
15.2.5	Loss of Condenser Vacuum	None	

FSAR Event Designation	Event Name	Methodology Topical Reports / NRC Acceptance Letters	Computer Codes used in Methodologies
15.2.6	Loss of Non-emergency AC Power	EMF-89-151(P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992. Acceptance letter from NRC dated March 16, 1992 included in (P)(A) version of report.	ANF-RELAP
15.2.7	Loss of Normal Feedwater	EMF-89-151(P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992. Acceptance letter from NRC dated March 16, 1992 included in (P)(A) version of report.	ANF-RELAP
15.2.8	Feedline Break	EMF-89-151(P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992. Acceptance letter from NRC dated March 16, 1992 included in (P)(A) version of report.	ANF-RELAP

FSAR Event Designation	Event Name	Methodology Topical Reports / NRC Acceptance Letters	Computer Codes used in Methodologies
15.3	**DECREASE IN REACTOR COOLANT SYSTEM FLOW**		
15.3.1	Partial Loss of Forced Reactor Coolant Flow	None	
15.3.2	Complete Loss of Forced Reactor Coolant Flow	1. EMF-89-151(P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992. Acceptance letter from NRC dated March 16, 1992 included in (P)(A) version of report. 2. XN-NF-82-21(P)(A) Revision 1, Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, Exxon Nuclear Company, September 1983. Acceptance letter from NRC dated August 26, 1983 included in (P)(A) version of report. 3. XN-75-21(P)(A) Revision 2, XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation, Exxon Nuclear Company, January 1986. Acceptance letter from NRC dated March 13, 1985 included in (P)(A) version of report.	1. ANF-RELAP 2. XCOBRA-IIIC

FSAR Event Designation	Event Name	Methodology Topical Reports / NRC Acceptance Letters	Computer Codes used in Methodologies
15.3.3	RCP Shaft Seizure (Locked Rotor)	1. EMF-89-151(P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992. Acceptance letter from NRC dated March 16, 1992 included in (P)(A) version of report. 2. XN-NF-82-21(P)(A) Revision 1, Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, Exxon Nuclear Company, September 1983. Acceptance letter from NRC dated August 26, 1983 included in (P)(A) version of report. 3. XN-75-21(P)(A) Revision 2, XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation, Exxon Nuclear Company, January 1986. Acceptance letter from NRC dated March 13, 1985 included in (P)(A) version of report.	1. ANF-RELAP 2. XCOBRA-IIIC

FSAR Event Designation	Event Name	Methodology Topical Reports / NRC Acceptance Letters	Computer Codes used in Methodologies
15.3.4	RCP Shaft Break	1. EMF-89-151(P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992. Acceptance letter from NRC dated March 16, 1992 included in (P)(A) version of report. 2. XN-NF-82-21(P)(A) Revision 1, Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, Exxon Nuclear Company, September 1983. Acceptance letter from NRC dated August 26, 1983 included in (P)(A) version of report. 3. XN-75-21(P)(A) Revision 2, XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation, Exxon Nuclear Company, January 1986. Acceptance letter from NRC dated March 13, 1985 included in (P)(A) version of report.	1. ANF-RELAP 2. XCOBRA-IIIC

FSAR Event Designation	Event Name	Methodology Topical Reports / NRC Acceptance Letters	Computer Codes used in Methodologies
15.4	**REACTIVITY AND POWER DISTRIBUTION ANOMALIES**		
15.4.1	Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition	1. EMF-89-151(P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992. Acceptance letter from NRC dated March 16, 1992 included in (P)(A) version of report. 2. XN-NF-82-21(P)(A) Revision 1, Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, Exxon Nuclear Company, September 1983. Acceptance letter from NRC dated August 26, 1983 included in (P)(A) version of report. 3. XN-75-21(P)(A) Revision 2, XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation, Exxon Nuclear Company, January 1986. Acceptance letter from NRC dated March 13, 1985 included in (P)(A) version of report.	1. ANF-RELAP 2. XCOBRA-IIIC

FSAR Event Designation	Event Name	Methodology Topical Reports / NRC Acceptance Letters	Computer Codes used in Methodologies
15.4.2	Uncontrolled RCCA Bank Withdrawal at Power	1. EMF-89-151(P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992. Acceptance letter from NRC dated March 16, 1992 included in (P)(A) version of report. 2. XN-NF-82-21(P)(A) Revision 1, Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, Exxon Nuclear Company, September 1983. Acceptance letter from NRC dated August 26, 1983 included in (P)(A) version of report. 3. XN-75-21(P)(A) Revision 2, XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation, Exxon Nuclear Company, January 1986. Acceptance letter from NRC dated March 13, 1985 included in (P)(A) version of report. 4. EMF-92-081(P), Revision 1, Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors, December 1998. Acceptance Letter from the NRC dated February 10, 2000.	1. ANF-RELAP 2. XCOBRA-IIIC

FSAR Event Designation	Event Name	Methodology Topical Reports / NRC Acceptance Letters	Computer Codes used in Methodologies
15.4.3	RCCA Misoperation 1) Dropped Rod/Bank 2) Single Rod Withdrawal 3) Statically Misaligned RCCA	1. XN-CC-28(A) Revision 3, XTG: A Two Group Three-Dimensional Reactor Simulator Utilizing Coarse Mesh Spacing, Exxon Nuclear Company, September 1975. Acceptance letter from NRC dated September 27, 1975 included in the (A) version of the report. 2. EMF-89-151(P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992. Acceptance letter from NRC dated March 16, 1992 included in (P)(A) version of report. 3. XN-NF-82-21(P)(A) Revision 1, Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, Exxon Nuclear Company, September 1983. Acceptance letter from NRC dated August 26, 1983 included in (P)(A) version of report. 4. XN-75-21(P)(A) Revision 2, XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation, Exxon Nuclear Company, January 1986. Acceptance letter from NRC dated March 13, 1985 included in (P)(A) version of report. 5. EMF-92-081(P), Revision 1, Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors, December 1998. Acceptance Letter from the NRC dated February 10, 2000.	1. ANF-RELAP 2. XCOBRA-IIIC 3. XTG
15.4.4	Startup of an Inactive RCP at an Incorrect Temperature	None	
15.4.5	A Malfunction or Failure of the Flow Controller in a BWR Loop that Results in an Increased Reactor Coolant Flow Rate	N/A	

FSAR Event Designation	Event Name	Methodology Topical Reports / NRC Acceptance Letters	Computer Codes used in Methodologies
15.4.6	CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant	ANF-84-73 Revision 5 Appendix B(P)(A)), "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events, " July 1990. Acceptance letter from NRC dated July 17, 1990 included in the (P)(A) version of report.	Manual Calculation
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	1. XN-NF-82-21(P)(A) Revision 1, Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, Exxon Nuclear Company, September 1983. Acceptance letter from NRC dated August 26, 1983 included in (P)(A) version of report. 2. XN-75-21(P)(A) Revision 2, XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation, Exxon Nuclear Company, January 1986. Acceptance letter from NRC dated March 13, 1985 included in (P)(A) version of report. 3. XN-CC-28(A) Revision 3, XTG: A Two Group Three-Dimensional Reactor Simulator Utilizing Coarse Mesh Spacing, Exxon Nuclear Company, September 1975. Acceptance letter from NRC dated September 27, 1975 included in the (A) version of the report.	1. XCOBRA-IIIC 2. XTG

FSAR Event Designation	Event Name	Methodology Topical Reports / NRC Acceptance Letters	Computer Codes used in Methodologies
15.4.8	Spectrum of Rod Cluster Control Assembly Ejection Accidents	1. EMF-89-151(P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992. Acceptance letter from NRC dated March 16, 1992 included in (P)(A) version of report. 2. XN-NF-82-21(P)(A) Revision 1, Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, Exxon Nuclear Company, September 1983. Acceptance letter from NRC dated August 26, 1983 included in (P)(A) version of report. 3. XN-75-21(P)(A) Revision 2, XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation, Exxon Nuclear Company, January 1986. Acceptance letter from NRC dated March 13, 1985 included in (P)(A) version of report. 4. XN-NF-78-44(A), A Generic Analysis of the Control Rod Ejection Transient for Pressurized Water Reactors, Exxon Nuclear Company, October 1983. 5. XN-CC-28(A) Revision 3, XTG: A Two Group Three-Dimensional Reactor Simulator Utilizing Coarse Mesh Spacing, Exxon Nuclear Company, September 1975. Acceptance letter from NRC dated September 27, 1975 included in the (A) version of the report.	1. ANF-RELAP 2. XCOBRA-IIIC 3. XTG
15.5	**INCREASE IN REACTOR COOLANT INVENTORY**		

FSAR Event Designation	Event Name	Methodology Topical Reports / NRC Acceptance Letters	Computer Codes used in Methodologies
15.5.1	Inadvertent Operation of the ECCS During Power Operation	EMF-89-151(P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992. Acceptance letter from NRC dated March 16, 1992 included in (P)(A) version of report.	ANF-RELAP
15.5.2	CVCS Malfunction that Increases RCS Inventory	None	

FSAR Event Designation	Event Name	Methodology Topical Reports / NRC Acceptance Letters	Computer Codes used in Methodologies
15.6	**DECREASE IN REACTOR COOLANT INVENTORY**		
15.6.1	Inadvertent Opening of a Pressurizer Safety or PORV	1. EMF-89-151(P)(A), ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events, Advanced Nuclear Fuels Corporation, May 1992. Acceptance letter from NRC dated March 16, 1992 included in (P)(A) version of report. 2. XN-NF-82-21(P)(A) Revision 1, Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations, Exxon Nuclear Company, September 1983. Acceptance letter from NRC dated August 26, 1983 included in (P)(A) version of report. 3. XN-75-21(P)(A) Revision 2, XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation, Exxon Nuclear Company, January 1986. Acceptance letter from NRC dated March 13, 1985 included in (P)(A) version of report.	1. ANF-RELAP 2. XCOBRA-IIIC
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	None	Spreadsheet

FSAR Event Designation	Event Name	Methodology Topical Reports / NRC Acceptance Letters	Computer Codes used in Methodologies
15.6.3	Radiological Consequences of Steam Generator Tube Rupture	1. WCAP-10698-P-A, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," Lewis, Huang, Behnke, Fittante, Gelman, August 1987. Approval letter from NRC dated March 30, 1987. 2. Supplement 1 to WCAP-10698-P-A, "Evaluation of Offsite Radiation Doses for a Steam Generator Tube Rupture Accident," Lewis, Huang, Rubin, March 1986. Approval letter from NRC dated December 17, 1985.	1. LOFTTR2 (an updated version of the LOFTTR1 program used in the WCAP 10698 references cited at left) 2. TITAN5 (Radiological Consequences – No topicals submitted, nor are there any NRC generic approval letters for this specific code and methods. The methods and code, however, have been used in several submittals (by Westinghouse and other licensees) and the results of these analyses have been acceptable to the NRC.
15.6.4	Radiological Consequences of a Main Steam Line Failure Outside Containment	N/A	

FSAR Event Designation	Event Name	Methodology Topical Reports / NRC Acceptance Letters	Computer Codes used in Methodologies
15.6.5	Loss of Coolant Accidents 1) Large Break LOCA	EMF-2087(P)(A), SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications, Siemens Power Corporation, June 1999. Approval letter from NRC dated June 15, 1999.	SEM/PWR-98 Including: 1. RODEX2 2. RELAP4-EM 3. CONTEMPT/LT-22 4. REFLEX 5. TOODEE2
	2) Small Break LOCA	1. XN-NF-82-49(P)(A), Revision 1, Exxon Nuclear Company Evaluation Model EXEM PWR Small Break LOCA Model, April, 1989. Acceptance letter from NRC dated July 12, 1988 included in the (P)(A) version of the report. 2. XN-NF-82-49(P)(A), Revision 1, Supplement 1 and Correspondence, Exxon Nuclear Company Evaluation Model Revised EXEM PWR Small Break Model, December, 1994. Acceptance letter from NRC dated October 3, 1994 included in (P)(A) version of report.	1. ANF-RELAP 2. RODEX2 3. TOODEE2

FSAR Event Designation	Event Name	Methodology Topical Reports / NRC Acceptance Letters	Computer Codes used in Methodologies
15.7	**RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT**		
15.7.1	Radioactive Waste Gas System Leak or Failure	None	Manual Calculation
15.7.2	Liquid Waste System Leak or Failure	None	Manual Calculation
15.7.3	Postulated Radioactive Releases Due to Liquid Tank Failure	None	Manual Calculation
15.7.4	Fuel Handling Accidents: Inside Containment Inside Fuel Handling Building	No topicals submitted, nor are there any NRC generic approval letters for these specific codes and methods. The methods and codes, however , have been used in several submittals (both by CP&L and by other licensees) and the results of these analyses have been acceptable to the NRC.	1. FISSION 2. CONTROOM 3. MicroShield, Version 4.10
15.7.5	Spent Fuel Cask Drop Accidents	N/A	N/A

Key Inputs for Chapter 15 Events

Tables extracted from Engineering Report EMF-2377, Rev. 1

Input Provided in Response to staff RAI question #2

SIEMENS

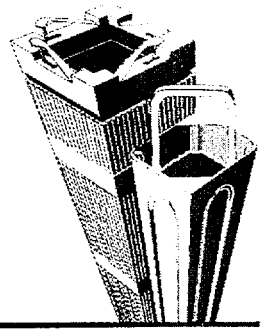
EMF-2377
Revision 1

Carolina Power and Light Company

Harris Nuclear Plant

FSAR Chapter 15
Engineering Report for Harris Nuclear Plant Steam Generator Replacement/Uprating

July 2000



Siemens Power Corporation

Table 6.1.5-1 Input Parameters and Biasing for Control Rod Ejection

Parameter	HFP Cases	HZP Cases
Core power	Rated + 2%	1 kW
RCS pressure	Nominal	Nominal
Pressurizer level	Nominal at HFP	Nominal at HZP power
Core average temperature	Nominal at HFP	Nominal at HZP power
RCS flow rate	Tech. Spec. minimum	Tech. Spec. minimum
Steam generator pressure	Nominal	Nominal at HZP power
Initial feedwater flow rate	Nominal	~ 0
Feedwater temperature	Nominal	Nominal
Steam generator liquid level	Nominal	Nominal
Cycle exposure	BOC and EOC	BOC and EOC
Moderator Temperature Coefficient	Tech. Spec. limit for BOC Bounding least negative for EOC	Tech. Spec. limit for BOC Bounding least negative for EOC
Doppler coefficient	0.8 * BOC 0.8 * EOC (temperature-dependent function)	0.8 * BOC 0.8 * EOC (temperature-dependent function)
Delayed neutron fraction, β	Bounding minimum for exposure	Bounding minimum for exposure
β/ℓ	Nominal for exposure	Nominal for exposure
U^{238} capture-to-fission ratio	Nominal	Nominal
Reactor trip reactivity insertion	Minimum allowed shutdown margin and the most reactive rod stuck out of the core	Minimum allowed shutdown margin and the most reactive rod stuck out of the core
Ejected rod worth, pcm	50	500
Pellet-to-cladding heat transfer coefficient	Maximum bounding HFP value for exposure	Maximum bounding HFP value for exposure
Steam generator tube plugging	Maximum	Maximum
Rod position controller	Manual	Manual
Pressurizer heaters	Disabled	Disabled
Pressurizer spray	Available	Available
Pressurizer PORVs	Available	Available
Main feedwater	Auto	Auto
Auxiliary feedwater	Available	Available

Table 6.2.2-1 Input Parameters and Biasing for Increase in Feedwater Flow

Parameter	HFP Cases	HZP Case
Core power	Rated + 2%	10^{-9} rated
RCS pressure	Nominal	Nominal
Pressurizer level	Nominal	Nominal
Core average temperature	Nominal at HFP	Nominal at HZP
RCS flow rate	Tech. Spec. minimum	Tech. Spec. minimum
Steam generator pressure	Nominal at HFP	Nominal at HZP
Initial feedwater flow rate	Nominal at HFP	0
Feedwater temperature	Nominal	Nominal
Steam generator liquid level	Nominal	Nominal
Cycle exposure	BOC and EOC	EOC
Moderator Temperature Coefficient	Tech. Spec. limits	Tech. Spec. limit
Doppler coefficient	0.8 * BOC 0.8 * EOC	0.8 * EOC
Delayed neutron fraction, β	Nominal for exposure	Nominal
β/ℓ	Nominal for exposure	Nominal
U^{238} capture-to-fission ratio	Nominal	Nominal
Reactor trip reactivity insertion	Minimum allowed shutdown margin and the most reactive rod stuck out of the core	Minimum allowed shutdown margin and the most reactive rod stuck out of the core
Pellet-to-cladding heat transfer coefficient	Mean	Mean
PORV setpoint	Nominal - tolerance	Nominal - tolerance
Pressurizer SRV setpoint	Nominal - tolerance	Nominal - tolerance
MSSV setpoints	Nominal + tolerance	Nominal + tolerance
Steam generator tube plugging	Nominal	Nominal
Rod position controller	Auto for BOC Manual and auto for EOC	Manual
Pressurizer heaters	Disabled	Disabled
Pressurizer spray	Available	Available
Main feedwater	Faulted	Faulted
Auxiliary feedwater	Available	Available

Table 6.2.3-1 Input Parameters and Biasing for Increase in Steam Flow

Parameter	Minimum Feedback Case	Maximum Feedback Case
Core power	Rated + 2%	Rated + 2%
RCS pressure	Nominal	Nominal
Pressurizer level	Nominal	Nominal
Core average temperature	Nominal	Nominal
RCS flow rate	Tech. Spec. minimum	Tech. Spec. minimum
Steam generator pressure	Nominal	Nominal
Initial feedwater flow rate	Nominal	Nominal
Feedwater temperature	Nominal	Nominal
Steam generator liquid level	Nominal	Nominal
Cycle exposure	BOC	EOC
Moderator Temperature Coefficient	Tech. Spec. limit	Tech. Spec. limit
Doppler coefficient	0.8 * BOC	0.8 * EOC
Delayed neutron fraction, β	Nominal	Nominal
β/ℓ	Nominal	Nominal
U^{238} capture-to-fission ratio	Nominal	Nominal
Reactor trip reactivity insertion	Minimum allowed shutdown margin and the most reactive rod stuck out of the core	Minimum allowed shutdown margin and the most reactive rod stuck out of the core
Pellet-to-cladding heat transfer coefficient	Mean	Mean
Steam generator tube plugging	Nominal	Nominal
Rod position controller	Auto	Auto
Pressurizer heaters	Disabled	Disabled
Pressurizer spray	Available	Available
Pressurizer PORVs	Available	Available
Main feedwater	Auto	Auto
Auxiliary feedwater	Available	Available

Table 6.2.8-1 Input Parameters and Biasing for Turbine Trip

Parameter	Limiting Primary Overpressurization Case	Limiting Secondary Overpressurization Case	Limiting MDNBR Case
Core power	Rated + 2%	Rated + 2%	Rated + 2%
RCS pressure	Nominal	Nominal	Nominal
Pressurizer level	Nominal+uncertainty	Nominal+uncertainty	Nominal
Core average temperature	Minimum (580.8 °F)	Nominal	Nominal
RCS flow rate	Tech. Spec. minimum	Tech. Spec. minimum	Tech. Spec. minimum
Steam generator pressure	Nominal	Nominal	Nominal
Initial feedwater flow rate	Nominal	Nominal	Nominal
Feedwater temperature	Nominal	Nominal	Nominal
Steam generator liquid level	Nominal	Nominal	Nominal
Cycle exposure	BOC	BOC	BOC
Moderator Temperature Coefficient	Tech. Spec. limit	Tech. Spec. Limit	Tech. Spec. limit
Doppler coefficient	0.8 * BOC	0.8 * BOC	0.8 * BOC
Delayed neutron fraction, β	Minimum	Minimum	Minimum
β/ℓ	Nominal	Nominal	Nominal
U^{238} capture-to-fission ratio	Nominal	Nominal	Nominal
Reactor trip reactivity insertion	Minimum allowed shutdown margin and the most reactive rod stuck out of the core	Minimum allowed shutdown margin and the most reactive rod stuck out of the core	Minimum allowed shutdown margin and the most reactive rod stuck out of the core
Pellet-to-cladding heat transfer coefficient	Mean	Mean	Mean
Steam generator tube plugging	Maximum	Minimum	Maximum
Pressurizer SRV setpoint	Nominal+tolerance	Nominal+tolerance	Nominal+tolerance
Pressurizer PORV setpoints	Disabled	Nominal-tolerance	Nominal-tolerance
MSSV setpoints	Nominal+tolerance	Nominal+tolerance	Nominal+tolerance
Rod position controller	Manual	Manual	Manual
Pressurizer heaters	Available	Available	Available
Pressurizer spray	Disabled	Available	Available
Main feedwater	Auto	Auto	Auto
Auxiliary feedwater	Disabled	Disabled	Disabled

Table 6.2.11-1 Input Parameters and Biasing for Loss of AC Power

Parameter	Value
Core power	Rated + 2%
RCS pressure	Nominal
Pressurizer level	Nominal+uncertainty
Core average temperature	Nominal at HFP
RCS flow rate	Tech. Spec. minimum
Steam generator pressure	Nominal at HFP
Initial feedwater flow rate	Nominal at HFP
Feedwater temperature	Nominal
Steam generator liquid level	Nominal
Cycle exposure	BOC
Moderator Temperature Coefficient	Tech. Spec. limit
Doppler coefficient	0.8 * BOC
Delayed neutron fraction, β	Nominal for exposure
β/ℓ	Nominal for exposure
U^{238} capture-to-fission ratio	Bounding maximum
Reactor trip reactivity insertion	Minimum allowed shutdown margin and the most reactive rod stuck out of the core
Pellet-to-cladding heat transfer coefficient	Mean
PORV setpoint	Nominal – tolerance
MSSV setpoints	Nominal + tolerance
Steam generator tube plugging	Minimum
Rod position controller	Manual
Pressurizer heaters	Available
Pressurizer spray	Available
Main feedwater	Faulted
Auxiliary feedwater	Available (one motor-driven pump)

Table 6.2.12-1 Input Parameters and Biasing for Loss of Normal Feedwater

Parameter	Value
Core power	Rated + 2%
RCS pressure	Nominal
Pressurizer level	Nominal+uncertainty
Core average temperature	Nominal at HFP
RCS flow rate	Tech. Spec. minimum
Steam generator pressure	Nominal at HFP
Initial feedwater flow rate	Nominal at HFP
Feedwater temperature	Nominal
Steam generator liquid level	Nominal
Cycle exposure	BOC
Moderator Temperature Coefficient	Tech. Spec. limit
Doppler coefficient	0.8 * BOC
Delayed neutron fraction, β	Nominal for exposure
β/ℓ	Nominal for exposure
U^{238} capture-to-fission ratio	Bounding maximum
Reactor trip reactivity insertion	Minimum allowed shutdown margin and the most reactive rod stuck out of the core
Pellet-to-cladding heat transfer coefficient	Mean
PORV setpoint	Nominal – tolerance
MSSV setpoints	Nominal + tolerance
Steam generator tube plugging	Minimum
Rod position controller	Manual
Pressurizer heaters	Available
Pressurizer spray	Available
Main feedwater	Faulted
Auxiliary feedwater	Available (one motor-driven pump)

Table 6.2.15-1 Input Parameters and Biasing for Loss of Forced Reactor Coolant Flow

Parameter	Value
Core power	Rated + 2%
RCS pressure	Nominal
Pressurizer level	Nominal
Core average temperature	Nominal
RCS flow rate	Tech. Spec. minimum
Steam generator pressure	Nominal
Initial feedwater flow rate	Nominal
Feedwater temperature	Nominal
Steam generator liquid level	Nominal
Cycle exposure	BOC
Moderator Temperature Coefficient	Tech. Spec. limit
Doppler coefficient	0.8 * BOC
Delayed neutron fraction, β	Nominal
β/ℓ	Nominal
U^{238} capture-to-fission ratio	Nominal
Reactor trip reactivity insertion	Minimum HFP scram worth and the most reactive rod stuck out of the core
Pellet-to-cladding heat transfer coefficient	Mean
Steam generator tube plugging	Maximum
Pressurizer PORV setpoint	Nominal-tolerance
Pressurizer SRV setpoint	Nominal-tolerance
MSSV setpoints	Nominal+tolerance
Rod position controller	Manual
Pressurizer heaters	Disabled
Pressurizer spray	Available
Main feedwater	Isolated at event initiation
Auxiliary feedwater	Available

Table 6.2.16-1 Input Parameters and Biasing for Locked Rotor

Parameter	MDNBR Case	Overpressurization Case
Core power	Rated + 2%	Rated + 2%
RCS pressure	Nominal	Nominal
Pressurizer level	Nominal	Nominal
Core average temperature	Nominal	Minimum (580.8 °F)
RCS flow rate	Tech. Spec. minimum	Tech. Spec. minimum
Steam generator pressure	Nominal	Nominal
Initial feedwater flow rate	Nominal	Nominal
Feedwater temperature	Nominal	Nominal
Steam generator liquid level	Nominal	Nominal
Cycle exposure	BOC	BOC
Moderator Temperature Coefficient	Tech. Spec. limit	Tech. Spec. limit
Doppler coefficient	0.8 * BOC	0.8 * BOC
Delayed neutron fraction, β	Nominal	Nominal
β/ℓ	Nominal	Nominal
U^{238} capture-to-fission ratio	Nominal	Nominal
Reactor trip reactivity insertion	Minimum allowed shutdown margin and the most reactive rod stuck out of the core	Minimum allowed shutdown margin and the most reactive rod stuck out of the core
Pellet-to-cladding heat transfer coefficient	Mean	Mean
Pressurizer PORV setpoint	Nominal	Disabled
Pressurizer SRV setpoint	Nominal - tolerance	Nominal + tolerance
MSSV setpoints	Nominal + tolerance	Nominal + tolerance
Steam generator tube plugging	Maximum	Maximum
Rod position controller	Manual	Manual
Pressurizer heaters	Disabled	Available
Pressurizer spray	Available	Disabled
Main feedwater	Auto	Auto
Auxiliary feedwater	Available	Available

Table 6.2.17-1 Input Parameters and Biasing for RCP Shaft Break

Parameter	MDNBR Case	Overpressurization Case
Core power	Rated + 2%	Rated + 2%
RCS pressure	Nominal	Nominal
Pressurizer level	Nominal	Nominal
Core average temperature	Nominal	Minimum (580.8 °F)
RCS flow rate	Tech. Spec. minimum	Tech. Spec. minimum
Steam generator pressure	Nominal	Nominal
Initial feedwater flow rate	Nominal	Nominal
Feedwater temperature	Nominal	Nominal
Steam generator liquid level	Nominal	Nominal
Cycle exposure	BOC	BOC
Moderator Temperature Coefficient	Tech. Spec. limit	Tech. Spec. limit
Doppler coefficient	0.8 * BOC	0.8 * BOC
Delayed neutron fraction, β	Nominal	Nominal
β/ℓ	Nominal	Nominal
U^{238} capture-to-fission ratio	Nominal	Nominal
Reactor trip reactivity insertion	Minimum allowed shutdown margin and the most reactive rod stuck out of the core	Minimum allowed shutdown margin and the most reactive rod stuck out of the core
Pellet-to-cladding heat transfer coefficient	Mean	Mean
Pressurizer PORV setpoint	Nominal	Disabled
Pressurizer SRV setpoint	Nominal - tolerance	Nominal + tolerance
MSSV setpoints	Nominal + tolerance	Nominal + tolerance
Steam generator tube plugging	Maximum	Maximum
Rod position controller	Manual	Manual
Pressurizer heaters	Disabled	Available
Pressurizer spray	Available	Disabled
Main feedwater	Auto	Auto
Auxiliary feedwater	Available	Available

Table 6.2.18-1 Input Parameters and Biasing for Uncontrolled Bank Withdrawal From a Subcritical or Low Power Condition

Parameter	Value
Core power	10^{-9} * rated
RCS pressure	Nominal
Pressurizer level	Nominal at HZP
Core average temperature	Nominal at HZP
RCS flow rate	Tech. Spec. minimum
Steam generator pressure	Nominal at HZP
Initial feedwater flow rate	~ None
Feedwater temperature	Nominal
Steam generator liquid level	Nominal
Cycle exposure	BOC
Moderator Temperature Coefficient	Tech. Spec. limit for HZP
Doppler coefficient	0.8 * BOC (temperature-dependent function)
Delayed neutron fraction, β	Maximum at BOC
β/ℓ	Nominal at BOC
U^{238} capture-to-fission ratio	Nominal
Reactor trip reactivity insertion	Minimum allowed shutdown margin and the most reactive rod stuck out of the core
Differential bank worth	Bounding maximum of two sequential banks being withdrawn in overlap
Pellet-to-cladding heat transfer coefficient	Maximum bounding value for BOC exposure
Rod position controller	Manual
Pressurizer heaters	Disabled
Pressurizer spray	Available
Pressurizer PORVs	Available
Main feedwater	Auto
Auxiliary feedwater	Available
Number of operational RCPs	Two

Table 6.2.19-1 Input Parameters and Biasing for Uncontrolled Bank Withdrawal at Power

Parameter	100% Power Cases	60% Power Cases
Core power	Rated + 2%	60% of rated
RCS pressure	Nominal	Nominal
Pressurizer level	Nominal at HFP	Nominal at 60% power
Core average temperature	Nominal at HFP	Nominal at 60% power
RCS flow rate	Tech. Spec. minimum	Tech. Spec. minimum
Steam generator pressure	Nominal	Nominal at 60% power
Initial feedwater flow rate	Nominal	Nominal at 60% power
Feedwater temperature	Nominal	Nominal
Steam generator liquid level	Nominal	Nominal
Cycle exposure	BOC and EOC	BOC and EOC
Moderator Temperature Coefficient	Tech. Spec. limits	Tech. Spec. limits
Doppler coefficient	Bounding least negative at BOC and bounding most negative at EOC	Bounding least negative at BOC and bounding most negative at EOC
Delayed neutron fraction, β	Nominal for exposure	Nominal for exposure
β/ℓ	Nominal for exposure	Nominal for exposure
U^{238} capture-to-fission ratio	Nominal	Nominal
Reactor trip reactivity insertion	Minimum allowed shutdown margin and the most reactive rod stuck out of the core	Minimum allowed shutdown margin and the most reactive rod stuck out of the core
Differential bank worth	Spectrum	Spectrum
Pellet-to-cladding heat transfer coefficient	Mean	Mean
Steam generator tube plugging	Maximum	Maximum
Rod position controller	Manual	Manual
Pressurizer heaters	Available	Available
Pressurizer spray	Available	Available
Pressurizer PORVs	Available	Available
Main feedwater	Auto	Auto
Auxiliary feedwater	Available	Available

Table 6.2.20-1 Input Parameters and Biasing for Dropped RCCA

Parameter	Value
Core power	Rated + 2%
RCS pressure	Nominal
Pressurizer level	Nominal
Core average temperature	Nominal
RCS flow rate	Tech. Spec. minimum
Steam generator pressure	Nominal
Initial feedwater flow rate	Nominal
Feedwater temperature	Nominal
Steam generator liquid level	Nominal
Cycle exposure	BOC and EOC
Moderator Temperature Coefficient	Tech. Spec. limit
Doppler coefficient	Bounding for exposure
Delayed neutron fraction, β	Nominal for exposure
β/ℓ	Nominal for exposure
U^{238} capture-to-fission ratio	Nominal
Reactor trip reactivity insertion	Minimum allowed shutdown margin and the most reactive rod stuck out of the core
Differential bank worth	Spectrum
Dropped rod worth	Spectrum
Rod shadowing factor	Spectrum
Pellet-to-cladding heat transfer coefficient	Mean
PORV setpoint	Nominal
Pressurizer SRV setpoint	Nominal-tolerance
MSSV setpoints	Nominal+tolerance

Table 6.2.20-1 Input Parameters and Biasing for Dropped RCCA
(continued)

Parameter	Value
Steam generator tube plugging	Maximum
Rod position controller	Auto
Pressurizer heaters	Disabled
Pressurizer spray	Available
Main feedwater	Auto
Auxiliary feedwater	Available
Turbine runback	Available

Table 6.2.28-1 Input Parameters and Biasing for Inadvertent Opening of a Pressurizer Safety Valve or PORV

Parameter	Value
Core power	Rated + 2%
RCS pressure	Nominal
Pressurizer level	Nominal
Core average temperature	Nominal
RCS flow rate	Tech. Spec. minimum
Steam generator pressure	Nominal
Initial feedwater flow rate	Nominal
Feedwater temperature	Nominal
Steam generator liquid level	Nominal
Cycle exposure	BOC
Moderator density reactivity	Consistent with BOC Tech. Spec. limit
Doppler coefficient	0.8 * BOC
Delayed neutron fraction, β	Bounding minimum
β/ℓ	Nominal
U^{238} capture-to-fission ratio	Nominal
Reactor trip reactivity insertion	Minimum allowed shutdown margin and the most reactive rod stuck out of the core
Pellet-to-cladding heat transfer coefficient	Mean
Rod position controller	Manual
Pressurizer heaters	Disabled
Pressurizer spray	Available
Pressurizer PORVs	Available
MSSVs	Available
Main feedwater	Auto
Auxiliary feedwater	Available

Enclosure 4 to SERIAL: HNP-01-078

Harris Nuclear Plant SGR/PUR
Uprate Fuel Analysis Plant Parameters Document
(UFAPPD)

Provided in Response to staff RAI question #3

VERIFY FOR OUTSTANDING CHANGES BEFORE USE

Revision 3

Harris Nuclear Plant SGR/PUR
Uprate Fuel Analysis Plant Parameters Document

Revision Date: 12/15/99

VERIFY FOR OUTSTANDING CHANGES BEFORE USE

CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT

HARRIS NUCLEAR PLANT SGR/PUR UPRATE FUEL ANALYSIS PLANT PARAMETERS DOCUMENT

REVISION	DATE	PREPARED BY	INDEPENDENT REVIEW BY	APPROVED BY
0	03/30/99	Joel Handschuh (Co-Sign L. A. Martin)	William R. Ziegler 03/31/99 (Co-sign by L. A. Martin)	Leo A. Martin 04/01/99
1	04/19/99	William R. Ziegler (Co-Sign L. A. Martin)	Wallace R. Ponder 4-21-99	Leo A. Martin 04/21/99
2	11/15/99	William R. Ziegler	Wallace R. Ponder 12/01/99	Leo A. Martin 12/01/99
3	01/12/00	William R Ziegler	1/20/00	1/20/00

Harris Nuclear Plant, SGR/PUR
Uprate Fuel Analysis Plant Parameters Document

Nature of Changes

Relative to DBD-314, Revision 3, *Harris Nuclear Plant Parameters Document*, the following changes were made for SGR/PUR Uprate Fuel Analysis Plant Parameters Document:

Item	Paragraph or Page(s)	Description and Justification
1.	Throughout	Parameters associated with Power Uprating and the new Steam Generators have been modified to reflect these changes.
2.	Throughout	Comments and notes were added to several tables to update the references and provide clarification for several parameters. The tables were re-structured to include pre-uprate and uprate parameters, and to provide space for identification of documents which are known to use/reference the particular parameter values.
3.	Throughout	References were renumbered.
4.	Throughout	Comments were moved to a separate section at the end of the document, and separated by a table number identifier.
5.	Throughout	Siemens Analysis Assumptions were removed.
6.	Throughout	The contents of the Cycle 6 PPD, Siemens Document EMF-93-033, were examined and elements removed between Cycle 6 and Cycle 9 required to support Power Uprating and new Steam Generators were re-implemented into this document.
7.	Table 2.1	Power inputs were modified to the Uprate values. Vessel Average Temperature at HFP was increased from 580.8 to 588.8°F
8.	Table 2.1	Nominal RCS Coolant Flow values are modified. Core Bypass Flow at Hot conditions was increased from 6.3% to 7.1%
9.	Table 2.1	Steam Generator Conditions were modified to reflect the replacement Steam Generators
10.	Table 2.2	Trip Setpoints for High Negative Flux Rate and Low-Low Steam Generator Level were changed.
11.	Table 2.2	Analysis Values for High Negative Flux Rate, Low Primary Coolant Flow, Low-Low Steam Generator Level were changed.

Harris Nuclear Plant, SGR/PUR
Uprate Fuel Analysis Plant Parameters Document

Nature of Changes (continued)

Relative to DBD-314, Revision 3, *Harris Nuclear Plant Parameters Document*, the following changes were made for SGR/PUR Uprate Fuel Analysis Parameters Document:

Item	Paragraph or Page(s)	Description and Justification
12.	Table 2.2	Revision 2 added OPDT/OTDT analysis values resulting from Westinghouse, Siemens and CP&L collaboration. The reference document was changed from the preliminary information transmittal to refer to the transmittal letter from CP&L to SPC of the "approved for analysis values."
13.	Table 2.2	Revision 2 added P7 Reactor Trip Interlock and High Pressurizer Level setpoint information.
14.	Table 2.3	Peaking Factor Limits were reduced by 4.5% to compensate for the Power Uprating
15.	Table 2.4	In Revision 1, the EOL MTC limit was changed to reflect interim Siemens Neutronics Inputs to Safety results.
16.	Table 2.6	A dimensional tolerance was added to the pump weir max. height
17.	Table 2.7	Several dimensions were modified.
18.	Table 2.8	Loop Seal Purge Time was increased based on CP&L Calculation, Ref. (75).
19.	Table 2.8	Revision 2 provided a more conservative SRV capacity to match the CP&L specification of a similar change for Cycle 10 analyses.
20.	Table 2.10 and Table 2.11	Steam Generator data was replaced to reflect the change from the Westinghouse Model D4 Generator to the Westinghouse Model Delta 75. Revision 2 changed the blowdown value supplied by Revisions 0 and 1 of the UFAPPD. Revision 3 added some Delta 75 physical dimensions requested by SPC to Table 2.10, and a note was added about blowdown isolation.
21.	Table 2.15	Revision 0 provided changed AFW min and max flows to be conservative for uprate conditions. Revision 1 added a more conservative maximum AFW flow value to this table.
22.	Table 2.15	Revision 2 added Volumes of hot/ambient water that must be swept out prior to injection of cold AFW, and additional minimum flowrates.

Harris Nuclear Plant, SGR/PUR
Uprate Fuel Analysis Plant Parameters Document

Nature of Changes (continued)

Relative to DBD-314, Revision 3, *Harris Nuclear Plant Parameters Document*, the following changes were made for SGR/PUR Uprate Fuel Analysis Parameters Document:

Item	Paragraph or Page(s)	Description and Justification
23.	Table 2.16	RCS Average Temperature was changed to increase the positive uncertainty. Revision 3 added PORV setpoint uncertainty.
24.	Table 2.17	Steam Generator Level setpoints are modified to represent the SGR. Revision 2 provided revised RCS volumes for Boron Dilution.
25.	Table 2.18	A conservative value of ECCS response time was chosen. The alternate miniflow valve response time is provided as a new item in this Table. The HHSI pump head curves listed are the degraded curves used by Siemens in the analysis.
26.	Table 2.18	Revision 2 added Steam generator High-High level Response time
27.	Table 2.19	A conservative maximum response time is chosen for the analysis value
28.	Table 2.22	Conservative inputs for the Maximum Spray Flowrate and response time are chosen.
29.	Table 2.23	A conservative minimum response time for Fan Coolers was chosen
30.	Table 2.25	Additional information on Containment Initial Pressure were added to this Table.
31.	Table 2.26	Revision 2 added ARC deadband, proportional range rod speed, and lockup setpoint.
32.	Table 2.29	Revision 2 added new Table for the parameters associated with SGR "only" option.

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1. Introduction

This report documents the principal parameters that represent the uprated Harris Nuclear Plant (HNP) plant configuration following Steam Generator Replacement/Power Uprate project (SGR/PUR). The primary purpose of this document is to provide a mechanism to identify plant configuration parameters which should be used in the generic Chapter 15 safety analyses by the fuel vendor for the SGR/PUR. Changes in the plant configuration should be reflected in this document to assure that the plant safety analyses are consistent with the SGR/PUR expected plant configuration.

The plant configuration parameters listed herein are those parameters used in the Chapter 15 analyses performed by Siemens Power Corporation (SPC) which might be expected to change before the end of plant life. These parameters include pump and other fluid hydraulic characteristics, protection and control setpoints, gains, time constants, etc., mitigating equipment response times, instrument or setpoint uncertainties, procedural requirements, Technical Specification limits, and valve capacities. Other Uprate Plant Parameters which are not expected to change in the life of the plant such as piping and equipment cross sectional areas and volumes can be found in SPC calculation notebooks (see CP&L ESR 95-00512 for a cross-reference listing of such references used by SPC in Reference 66, among others).

This report is applicable to the Uprate Analysis Plant Parameters for SGR/PUR of HNP. Its scope is based on combining or merging the list of parameters presented in earlier versions for Cycle 6 (EMF-93-033) and Cycle 9 (EMF-2002).

2. Uprate Analysis Plant Parameters

Table 2.1 Plant Operating Parameters

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Power						
Core Thermal	2775 MWt	2900 MWt	1.1.5 & others	TS Section1.28	DBD-314, Rev 3, Table 2.1, & others	Ref. (62), (63)
Net Heat Input	12 MWt	12.4 MWt	1.1.5 & others			Ref. (62), (79)
NSSS Power	2787 MWt	2912.4 MWt	1.1.5 & others			Ref. (62)
Vessel Average Temperature						
HFP	580.8°F	588.8°F	Several	TS Table 2.2-1	DBD-314, Rev 3, Table 2.1	Ref. (62), (63)
Low T_{AVG} Contingency		580.8°F				Ref. (62)
EOC T_{AVG} Coastdown Allowance	7.0°F					See Note 2.1.3
Nominal Lower Limit	≥ 557.0°F	≥ 557.0°F	Chapter 15 & others		DBD-314, Rev. 3, Table 2.1	Ref. (1), Page 25 of 98, Note 2.1.1
Minimum Analytical Temperature for Criticality	≥ 551.0°F	≥ 551.0°F		TS Section 3.1.1.4	DBD-314, Rev. 3, Table 2.4	No Change
Primary System Pressure (Pressurizer Master Controller Setpoint)	2250 psia	2250 psia	Many		UAPD Table 2.9 SD-100.03, Table 6.1) DBD-314 Rev. 3 Table 2.7	Ref. (1), sh. 35 of 98,Ref (62), (65)

Table 2.1 Plant Operating Parameters

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
RCS Coolant Flow						
Minimum Technical Specification Requirement	293,540 gpm	293,540 gpm		TS Section 3.2.5	DBD-314 Rev. 3 Table 2.1, EST-708, EST-709	Ref. (62)
Nominal (3% SG Tube Plugging)	305,478 gpm	304,800 gpm	Table 4.4.3-2			Ref. (62), Ref.(65) See Attachment B, Table 2
Nominal (0% SG Tube Plugging) Estimated		306,600 gpm				Ref. (62), Ref.(65) See Attachment B, Table 2
Mechanical design Flow	321,300 gpm	321,300 gpm		TS Figure 3.2-3	DBD-314 Rev. 3 Table 2.1, EST-708, EST-709	Ref. (62)
Bypass Flow						
Core						
Cold	7.1%	7.1%	4.4.1.3 & Table 4.4.3-3			Ref. (3), (62)
Hot	6.3%	7.1%	4.4.1.3 & Table 4.4.3-3			Ref. (62)
Baffle/Barrel Region and Core Cavity Region	1.6%	1.6%	Table 4.4.3-3			Ref. (3)

Table 2.1 Plant Operating Parameters

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Outlet Nozzle						
cold	1.0%	1.0%	Table 4.4.3-3		SD-100.02	Ref. (3)
hot	0.23%	1.0%	Table 4.4.3-3			Ref. (62)
Guide Tube	2.5%	2.5%	Table 4.4.3-3			Ref. (3)
Head Cooling	2.0%	2.0%	Table 4.4.3-3		SD-100.02	Ref. (3)
Steam Generator Secondary Pressure (psia) @ 100% Power, Tech Spec Minimum Flow Conditions (psia)	Cycle 4 (T_{avg} = 588.8°F) Measured: A = 950.17 B = 946.32 C = 950.37	T_{avg} 588.8°F 989 (0% Tube Plugging) 985 (3% Tube Plugging) T_{avg} 580.8°F 920 (0% Tube Plugging) 916 (3% Tube Plugging)				Ref. (63), Ref.(62) Note 2.1.2
Total Steam Flow @ 100% Power, Tech Spec Minimum Flow (Mlbm/hr)	Cycle 4 Measured: 12.17 Mlbm/hr	T_{avg} 588.8°F 12.85 T_{avg} 580.8°F 12.81				Ref.(62) Note 2.1.2
Feedwater Temperature @ 100% Power	Cycle 4 Measured: 432 °F	Range: 435°F - 440°F				Ref.(62) Note 2.1.2

Table 2.2 Reactor Trip Setpoints and Response Times

Uprate Analysis Parameter	Pre-Uprate Value			Uprate Value			Where Used and/or Reported in Uprate Documents			Uprate Source Document
	Trip Setpoint	Tech Spec Total Allowance	Analysis Value / Response Time	Trip Setpoint	Tech Spec Total Allowance	Analysis Value / Response Time	FSAR	Tech Spec	Other Documents	
Power Range, Neutron Flux										
High Setting	≤ 109% of RTP	+(0.075) (120% of RTP)	118% of RTP / <0.5 sec	≤ 109% of RTP	+(0.075) (120% of RTP)	118% of RTP / <0.5 sec		Tab. 2.2-1	SD-103	Ref. (66), Note 2.2.1
Low Setting	≤ 25% of RTP	+(0.083) (120% of RTP)	35% of RTP / <0.5 sec	≤ 25% of RTP	+(0.083) (120% of RTP)	35% of RTP / <0.5 sec		Tab. 2.2-1	SD-103	Ref. (66), Note 2.2.1
High Negative Flux Rate	≤ 5% of RTP with a time constant ≥ 2 sec	-(0.016) (120% of RTP)	-6.92% of RTP with a time constant ≥ 2 sec / <0.5 sec	≤ 5% of RTP with a time constant ≥ 2 sec		-8.00% of RTP with a time constant ≥ 2 sec / <0.5 sec	Table 15.0.6-2	Tab. 2.2-1	SD-103	Ref. (66), Note 2.2.1
High Pressurizer Pressure	≤ 2385 psig	+(0.075) (800 psi)	2445 psig/ < 2.0 sec	≤ 2385 psig	+(0.075) (800 psi)	2445 psig < 2.0 sec		Tab. 2.2-1	SD-103, SD-100.03, AOP-001, APP-ALB-009 APP-ALB-012	Ref. (66), Note 2.2.1

Table 2.2 Reactor Trip Setpoints and Response Times (continued)

Uprate Analysis Parameter	Pre-Uprate Value			Uprate Value			Where Used and/or Reported in Uprate Documents			Uprate Source Document
	Trip Setpoint	Tech Spec Total Allowance	Analysis Value / **Response Time**	Trip Setpoint	Tech Spec Total Allowance	Analysis Value / **Response Time**	FSAR	Tech Spec	Other Documents	
Low Pressurizer Pressure Lead Time Constant, τ_4 (See Note 2.2.2) Lag Time Constant, τ_5 (See Note 2.2.2)	≥ 1960 psig 2 sec 1 sec	- (0.05) (800 psi)	1920 psig / **< 2.0 sec** 1.8 sec 1.1 sec	≥ 1960 psig 2 sec 1 sec	- (0.05) (800 psi)	1920 psig/ **< 2.0 sec** 1.8 sec 1.1 sec		Tab. 2.2-1	SD-103, SD-100.03, AOP-001, APP-ALB-009 APP-ALB-019 AOP-016	Ref. (66), Note 2.2.1
Pressurizer Water Level - High	≤ 92% of instrument span	≤ 93.8% of instrument span	≤ 93.8% of instrument span	≤ 92% of instrument span	≤92.5% of instrument span	≤ 93.8% of instrument span		Table 2.2-1		Ref. (94) See Note 2.2.5
Low Primary Coolant Flow	≥ 90.5% of full flow	-(0.029) (120%)	87% of full flow / **≤ 1.0 sec**	≥ 90.5% of full flow		85% of full flow / **≤ 1.0 sec**		Table 2.2-1	SD-103	Ref. (66), Note 2.2.1
Low-Low Steam Generator Level— Loss of Feedwater Transient	≥ 38.5% of Narrow Range Span	-(0.192) (100%)	19.3% span/ **≤ 3.5 sec**	≥ 25.0% of Narrow Range Span		16.1% span / **≤ 3.5 sec**		Table 2.2-1	SD-103	Ref. (66), Note 2.2.1 **Ref. (5)**
Low-Low Steam Generator Level— FLB and SLB	≥ 38.5% of Narrow Range Span	-(0.192) (100%)	19.3% span/ **≤ 3.5 sec**	≥ 25.0% of Narrow Range Span		0.0 % span / **≤ 3.5 sec**		Table 2.2-1	SD-103	Ref. (66), Note 2.2.1 **Ref. (5)**

Table 2.2 Reactor Trip Setpoints and Response Times (continued)

Uprate Analysis Parameter	Pre-Uprate Value			Uprate Value			Where Used and/or Reported in Uprate Documents			Uprate Source Document
	Trip Setpoint	Tech Spec Total Allowance	Analysis Value / Response Time	Trip Setpoint	Tech Spec Total Allowance	Analysis Value / Response Time	FSAR	Tech Spec	Other Documents	
Undervoltage - Reactor Coolant Pumps	≥ 5148 volts		Note 2.2.3. ≤ 1.5 sec	> 5148 volts		Note 2.2.3 ≤ 1.5 sec		Table 2.2-1	SD-103, APP-ALB-010 APP-ALB-011 AOP-028	Ref. (66),(70), Note 2.2.1 Ref. (5)
Underfrequency - Reactor Coolant Pumps	≥ 57.5 Hz	-(0.05) (10 Hz)	57.0 Hz ≤ 0.6 sec	> 57.5 Hz	-(0.05) (10 Hz)	57.0 Hz ≤ 0.6 sec		Table 2.2-1	SD-103, APP-ALB-010 APP-ALB-011 AOP-029	Ref. (66),(71), Note 2.2.1 Ref. (84) Ref. (5)
Reactor Trip Interlocks										
Reactor Trip Interlock, P7 (P7 is fed by both P10 and P13)										
NI Power Indication, P10	≤ 10% of RTP	N/A	T.S. Allowable Value ≤12.1% of RTP	10% of RTP	N/A	T.S. Allowable Value ≤10.6% of RTP		Table 2.2-1	Ref. (84)	Ref. (93)
Turbine Impulse Pressure Equivalent Power Indication, P13	≤ 10% of RTP	N/A	T.S. Allowable Value ≤12.1% of RTP	10% of RTP	N/A	T.S. Allowable Value ≤10.6% of RTP		Table 2.2-1	Ref. (84)	Ref. (93)
Single Loop Loss-of-Flow Trip Block, P-8	≤ 49% of RTP	+(0.075) (120% of RTP)	58% of RTP	≤ 49% of RTP	+(0.075) (120% of RTP)	58% of RTP		Table 2.2-1	SD-103	Ref. (66) and Ref. (93)

Table 2.2 Reactor Trip Setpoints and Response Times (continued)

Uprate Analysis Parameter	Pre-Uprate Value			Uprate Value			Where Used and/or Reported in Uprate Documents			Uprate Source Document
	Trip Setpoint	Tech Spec Total Allowance	Analysis Value / **Response Time**	Trip Setpoint	Tech Spec Total Allowance	Analysis Value / **Response Time**	FSAR	Tech Spec	Other Documents	
Over Temperature ΔT Response Time			4.75 sec RTD lag time and 1.25 sec delay Footnote (1) in Attach. 1 to Ref. (5)			4.75 sec RTD lag time and 1.25 sec delay Footnote (1) in Attach. 1 to Ref. (5)				Ref.(5)
ΔT_0										
K_1	1.17	+(0.087) (150%)	1.30	1.185		1.320				Ref. (89)
K_2	0.0224/°F		0.0224/°F	0.0224/°F		0.0224/°F				Ref. (89)
K_3	0.001072/ psi		0.001072/ psi	0.0012/ psi		0.0012/ psi				Ref. (89)
τ_1	8 sec		7.2 sec	0.0 sec		0.0 sec				Ref. (89)
τ_2	3 sec		3.3 sec	0.0 sec		0.0 sec				Ref. (89)
τ_3	0 sec		0 sec	4.0 sec		4.0 sec				Ref. (89)
τ_4	20 sec		18 sec	22.0 sec	- 10%	19.8 sec				Ref. (89)
τ_5	4 sec		4.4 sec	4.0 sec	+10%	4.4 sec				Ref. (89)
τ_6	0 sec		0 sec	0.0 sec		0.0 sec				Note 2.2.4
$f_1(\Delta I)$	0 when +12% ≥ ΔI ≥ -21.6%		0 when +12% ≥ ΔI ≥ -21.6%	0 when +12% ≥ ΔI ≥ -21.6%		0 when +12% ≥ ΔI ≥ -21.6%				Ref. (2)

Table 2.2 Reactor Trip Setpoints and Response Times (continued)

Uprate Analysis Parameter	Pre-Uprate Value			Uprate Value			Where Used and/or Reported in Uprate Documents			Uprate Source Document
	Trip Setpoint	Tech Spec Total Allowance	Analysis Value / **Response Time**	Trip Setpoint	Tech Spec Total Allowance	Analysis Value / **Response Time**	FSAR	Tech Spec	Other Documents	
Over Temperature ΔT (Continued)										
$f_1(\Delta I)$	2.36% per neg. % ΔI; 1.57% per pos. %ΔI		2.36% per neg. % ΔI; 1.57% per pos. %ΔI	1.75% per neg. % ΔI; 1.60% per pos. %ΔI		1.75% per neg. % ΔI; 1.60% per pos. %ΔI				Ref. (2)
Over Power ΔT										
ΔT_0										
K_4	1.079	+(0.047) (150%)	1.150	1.120		1.180				Ref. (89)
K_5	0.02/°F for increasing average temperature; 0.0 for decreasing average temperature		0.02/°F for increasing average temperature; 0.0 for decreasing average temperature	0.02/°F for increasing average temperature; 0.0 for decreasing average temperature		0.02/°F for increasing average temperature; 0.0 for decreasing average temperature				Ref. (89)
K_6	0.002/°F for T > T"; 0.0 for T <T"		0.002/°F for T > T"; 0.0 for T <T"	0.002/°F for T > T"; 0.0 for T <T"		0.002/°F for T > T"; 0.0 for T <T"				Ref. (89)
τ_7	10 sec		9 sec	13 sec	- 10%	11.7 sec				Ref. (89)
$f_2(\Delta I)$	0 for all ΔI		0 for all ΔI	0 for all ΔI		0 for all ΔI				Ref. (2)

Table 2.2 Reactor Trip Setpoints and Response Times (continued)

Uprate Analysis Parameter	Pre-Uprate Value			Uprate Value			Where Used and/or Reported in Uprate Documents			Uprate Source Document
	Trip Setpoint	Tech Spec Total Allowance	Analysis Value / **Response Time**	Trip Setpoint	Tech Spec Total Allowance	Analysis Value / **Response Time**	FSAR	Tech Spec	Other Documents	
Control Rod Drop Time	≤ 2.7 seconds at full flow and operating temperature from beginning of rod motion to dashpot entry		≤ 2.7 seconds at full flow and operating temperature from beginning of rod motion to dashpot entry	≤ 2.7 seconds at full flow and operating temperature from beginning of rod motion to dashpot entry		≤ 2.7 seconds at full flow and operating temperature from beginning of rod motion to dashpot entry	15.0.5	Tech. Spec., Section 3.1.3.4	SD-104 EST-724 EST-704 DBD-314	No Change
Over Temperature ΔT and Over Power ΔT Trips Adjusted for Post Accident "Harsh Environment" Conditions?			No			No			TEM:98:392	No Change

Table 2.3 Heat Flux and Power Distribution Limits

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Core Heat Generated in Fuel, %	97.4	97.4	Table 4.1.1-1 Table 1.3.1-1			Ref. (10) Table 15.0.3-1
$F_{\Delta H}$ @ RTP	1.73	1.66	Table 4.3.2-2	T.S. 3.2.3	COLR (PLP-106)	Ref. (63) Note 2.3.1
F_Q @ RTP	2.52	2.41	Table 4.3.2-2	T.S. 3.2.2 T.S. 4.2.2.2.2.b	COLR (PLP-106)	Note 2.3.2

Table 2.4 Neutronic Parameters

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Moderator Temperature Coefficient						
Positive MTC Limit	≤ 5.0 pcm/°F for power levels up to 70% RTP with a linear ramp to 0 pcm/°F at 100% RTP	≤ 5.0 pcm/°F for power levels up to 70% RTP with a linear ramp to 0 pcm/°F at 100% RTP			COLR (PLP-106), Attachment 9 of Ref. (5), EST-703, EST-718, DBD-314	No Change
Negative MTC Limit	less negative than -45.0 pcm/°F	less negative than -50.0 pcm/°F				See Note 2.4.1
Required Shutdown Margin						
Operating Modes 1 and 2	1770 pcm	1770 pcm		T.S. 3.1.1.1	OST-1036, GP-002/004/005/006 EST-701/707, AOP-002, DBD-314	No Change
Operating Modes 3-5	Figure 1 in Attachment 9 to Ref. (5)	Figure 1 in Attachment 9 to Ref. (5)			COLR (PLP-106)	Figure 1 in Attachment 9 to Ref. (5)

Table 2.4 Neutronic Parameters (continued)

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Boron-10 Abundance						
Nominal maximum in RCS	19.78 atom % (natural)	19.78 atom % (natural)			POWERTRAX EXSPACK DBD-314	Ref. (64)
Nominal depletion rate	$\frac{-0.65 \text{ atom\% B-10}}{120 \text{ EFPD}}$	$\frac{-0.68 \text{ atom\% B-10}}{120 \text{ EFPD}}$				Ref. (75)

Table 2.5 Fuel and Core Parameters

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Total number of fuel assemblies	157	157				No Change
Total number of fuel rods	41448	41448				No Change
Assembly pitch	8.466 in.	8.466 in.				No Change
Thimble Plugs installed?	Yes	Yes			HW/98-032	No Change
Diameter of Control Rods						
Nominal	0.381 in.	0.381 in.				No Change
Maximum (at weld)	0.385 in.	0.385 in.				No Change
Control Bank Spacing	128 steps tip to tip	128 steps tip to tip				No Change

Table 2.6 Reactor Coolant Pump Parameters

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Rated Head						
Hot	276 feet	276 feet	Table 5.4.1-1		DBD-314 Rev. 3 Table 2.5, SD-100.01	Ref. (12), sh. 1, Ref. (13), Section 1, pg. 14
Cold	294 feet	294 feet			DBD-314 Rev. 3 Table 2.5	Ref. (12), sh. 1
Rated Flow						
Hot	103,400 gpm	103,400 gpm	Table 5.4.1-1		DBD-314 Rev. 3 Table 2.5 SD-100.01	Ref. (12), sh. 1, Ref. (13), Section 1, pg. 14
Cold	97,300 gpm	97,300 gpm			DBD-314 Rev. 3 Table 2.5	Ref. (12), sh. 1
Rated speed						
Hot	1186 rpm	1186 rpm	5.4.1.5.1 Table 5.4.1-1		DBD-314 Rev. 3 Table 2.5	Ref. (13), Section 1, pg. 164
Cold	1182 rpm	1182 rpm			DBD-314 Rev. 3 Table 2.5	Idem.
Brake Horsepower						
Hot (Nominal)	6700 bhp	6700 bhp	Table 5.4.1-1		DBD-314 Rev. 3 Table 2.5	Ref. (12), sh. 1, Ref. (13), Section 1, pg. 164
Cold (Nominal)	8870 bhp	8870 bhp	Table 5.4.1-1		DBD-314 Rev. 3 Table 2.5	Idem.

Table 2.6 Reactor Coolant Pump Parameters (continued)

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Torque						
Hot	29,670 ft-lbf/RCP	29,670 ft-lbf/RCP			DBD-314 Rev. 3 Table 2.5	Note 2.6.1
Cold	39,413 ft-lbf/RCP	39,413 ft-lbf/RCP			DBD-314 Rev. 3 Table 2.5	Note 2.6.2
Rated Fluid Conditions						
Hot	571.1°F, 2250 psia specific gravity = 0.747	571.1°F, 2250 psia specific gravity = 0.747			DBD-314 Rev. 3 Table 2.5	Ref. (13), Section 1, pg. 120
Cold	70.0°F, 500 psia specific gravity = 1.001	70.0°F, 500 psia specific gravity = 1.001			DBD-314 Rev. 3 Table 2.5	Ref. (13), Section 1, pg. 121
Specific speed	N_s = 5200	N_s = 5200			DBD-314 Rev. 3 Table 2.5	Note 2.6.3.
Total Reactor Coolant Pump and Motor Inertia	95,000 lb-ft^2	95,000 lb-ft^2	Table 5.4.1-1		DBD-314 Rev. 3 Table 2.5	Assumed to be the nominal value. Ref. (12), sh 1, Ref. (13), Section 1, pg. 14.
Pump weir maximum height	5.0 in	5.0 ± 0.1 in			DBD-314 Rev. 3 Table 2.5	Ref. (78)

Table 2.7 Pressurizer Parameters

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Total volume	1400 ft^3	1400 ft^3	Table 5.4.10-1		DBD Rev. 3, Table 2.7	Ref. (37)
Shell I.D.	84 in	84 in			DBD-314 Rev. 3 Table 2.7	Ref. (67), Ref. (68)
Upper level tap from upper dome	6.0 in	5.8 in			DBD-314 Rev. 3 Table 2.7	Ref. (67), Ref. (68), Ref. (69)
Lower level tap from lower dome	6.0 in	6.0 in			DBD-314 Rev. 3 Table 2.7	Ref. (67), Ref. (68), Ref. (69)
Upper dome height	28.04 in	28.24 in			DBD-314 Rev. 3 Table 2.7	Ref. (69)
Lower dome height	30.39 in	30.39 in			DBD-314 Rev. 3 Table 2.7	Ref. (69)

Table 2.8 Pressurizer Safety Valves

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Pressure Setpoint						
Open	2485 psig +2% / -1%	2485 psig +2% / -1%	Table 5.4.13-1	Tech. Spec. 3.4.2.2	DBD-314 Rev 3, Table 2.6	Tech. Spec., Section 3.4.2.2, Ref. (15), See Note 2.8.1 below
Close	2350 psia	2350 psia			DBD-314 Rev 3, Table 2.6	Ref. (14), See Note 2.8.2 below
Loop Seal Purge Time	1.11 seconds	1.13 seconds				Ref. (75)
Maximum Flow Capacity per valve		475,000 lbm/hr of saturated steam at 2500 psia.				Ref. (91)— See Note 2.8.3
Minimum Flow Capacity per valve	380,000 lbm/hr of saturated steam at valve setpoint	380,000 lbm/hr of saturated steam at valve setpoint	Table 5.4.13-1	Tech. Spec. Bases 3/4.4.2	DBD-314 Rev 3, Table 2.6	Tech. Spec. Bases 3/4.4.2, pg. B 3/4 4-1.

Table 2.9 Pressurizer Control

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Pressurizer Pressure Controller (PC-444A):						
Proportional gain, K_{21}	5 psi/psi	5 psi/psi				Ref. (1), sh. 35 of 98
Reset time constant, τ_{21}	1000 sec	1000 sec				Idem.
Pressure Setpoint, P_{ref}	2235 psig	2235 psig				Idem.
Pressurizer Power Operated Relief Valves:						
Dynamically compensated pressure valve (PCV-444B, a.k.a. 1RC-114):						
Pressure differential setpoint ($P - P_{ref}$):						
Open	100 psid	100 psid	Table 15.0.3-5			Ref. (1), sh. 37 & 38 of 98,
Close	80 psid	80 psid	Table 15.0.3-5			Ref. (1), sh. 37 of 98
Maximum flow capacity/valve	236,000 lbm/hr @ 2155 psia and saturation temp.	236,000 lbm/hr @ 2155 psia and saturation temp.				Ref. (36), Table 4.6.1-1b Note 2.9.1
Opening time (maximum)	2.0 sec	2.0 sec	Table 15.0.3-5			Conservative to value presented in Ref. (38)

Table 2.9 Pressurizer Control (continued)

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Dynamically compensated pressure valve (PCV-444B, a.k.a. 1RC-114): (Continued)						
Closing time (maximum)	2.0 sec	2.0 sec				Conservatively long relative to value presented in Ref. (38)
Non-compensated valve (PCV-445A, a.k.a. 1RC-118 and PCV-445B, a.k.a. 1RC-116):						
Pressure setpoint						
Open	2335 psig	2335 psig	Table 5.4.10-2 Table 15.0.3-5		SD-100.03	Ref. (1), sh. 37 & 38 of 98
Close	2315 psig	2315 psig	Table 15.0.3-5		SD-100.03	Ref. (1), sh. 37 of 98
Maximum flow capacity/valve	236,000 lbm/hr @ 2155 psia and saturation temp.	236,000 lbm/hr @ 2155 psia and saturation temp.				Ref. (36), Table 4.6.1-1b See note 2.9.1
Opening time (maximum)	2.0 sec	2.0 sec	Table 15.0.3-5			Conservative relative to value presented in Ref. (38)
Closing time (maximum)	2.0 sec	2.0 sec				Conservatively long relative to value presented in Ref. (38)

Table 2.9 Pressurizer Control (continued)

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Pressurizer Control:						
Level Control:						
Programmed level, no load	25% of level span @ T_{avg} = 557°F	25% of level span @ T_{avg} = 557°F			SD-100.03	Ref. (1), sh. 40 of 98
Programmed level, full load	60% of level span @ T_{avg} = 588.8°F	60% of level span @ T_{avg} = 588.8°F	5.4.10.3.2		SD-100.03	Idem
	51.2% of level span @ T_{avg} = 580.8°F	51.2% of level span @ T_{Tavg} = 580.8°F				Idem
Low-Low level heater cutout	17% of level span	17% of level span			SD-100.03	Ref. (1), sh. 40 of 98
Pressure Control:						
Setpoint at which spray is full on	2310 psig	2310 psig	Table 5.4.10-2		SD-100.03	Ref. (1), sh. 37 of 98
Setpoint at which spray initiates	2260 psig	2260 psig	Table 5.4.10-2		SD-100.03	Ref. (1), sh. 37 of 98

Table 2.9 Pressurizer Control (continued)

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Pressure Control (Continued):						
Spray valve controller proportional gain	2% valve lift/psi	2% valve lift/psi			SD-100.03	Ref. (1), sh. 38 of 98
Maximum Pressurizer Spray Flow Capacity	350 gpm/valve @ 2485 psig and 650°F	350 gpm/valve @ 2485 psig and 650°F	Table 5.1.0-1			Ref. (37) Ref. (87)
Setpoint for proportional heaters						
Full on	2220 psig	2220 psig	5.4.10 Table 5.4.10-2			Ref. (1), sh. 37 of 98
Full off	2250 psig	2250 psig				Ref. (1), sh. 37 of 98
Setpoint for backup heaters						
Full on						
pressure	2210 psig	2210 psig	Table 5.4.10-2			Ref. (1), sh. 37 of 98
Pressurizer level	5% above programmed level	5% above programmed level				Ref. (1), sh. 40 of 98
full off	2218 psig	2218 psig				Ref. (1), sh. 37 of 98

Table 2.9 Pressurizer Control (continued)

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Backup heater capacity	≥ 1024 kW	≥ 1024 kW				Ref. (26), Ref. (27), and Ref. (28)
Proportional heater capacity	≥ 377 kW	≥ 377 kW				Ref. (25)

Table 2.10 Steam Generator Parameters

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Type of Steam Generator	Westinghouse Model D4	Westinghouse Model Delta 75			SD-100.01 (Diagram) SD-126.02	Ref. (4) Ref. (80)
Tube Plugging (%)						
Maximum Value	6%	3%				Ref. (62)
Main Feedwater/Auxiliary Feedwater Flow Split						
Analysis Value	80/20	100/0				Ref. (80), Section 3.3.1.2 Note 2.10.1
Secondary Side Fluid Mass						
Nominal HFP (0% tube plugging, thermal design fouling factor, and thermal design primary flow rate-92,600gpm)	101,500 lbm/SG	108,474 lbm/SG				Ref. (4)
HZP (0% tube plugging, thermal design fouling factor, and thermal design primary flow rate-92,600gpm)	162,000 lbm/SG	161,575 lbm/SG				Ref. (4)
Steam Generator Blowdown (Maximum Continuous Tubesheet)		28,000 lbm/hr				Ref. (99) Note 2.10.2

Table 2.10 – Steam Generator Parameters (Continued)

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Replacement Steam Generator Physical Parameters						
Length of average SG tube, not including the portion of the tube which penetrates the tubesheet	value used = 795.8 in.	794.364 in				Ref. (75) & Ref.(4)
Elevation of the mid-deck plate	value used = 569.2 in. above the tubesheet, deducing that the mid-deck plate is 4 in. below the top of the swirl vane separators which is 573.2 in.	569.2 in				Ref. (75) & Ref.(4)
Total volume of downcomer from tubesheet to mid-deck plate	value used = 1725 ft3	1698.1 ft3				Ref. (75) & Ref.(4)
Volume of SG riser (boiler region) between tubesheet and top of tube bundle	value used = 1770 ft3	1769.6 ft3				Ref. (75) & Ref.(4)
Volume of SG riser between top of SG tubes and mid-deck plate	value used = 669 ft3	691.4 ft3				Ref. (75) & Ref.(4)

Table 2.11 Steam Generator Water Level Control

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Feedwater Control Valves:						
Closing stroke time	8 sec	8 sec				Ref. (18)
Maximum C_v	1127 (Cycle 6 through Cycle 9) 950 (Cycle 10)	950				Ref. (91) See Note 2.11.1
Feedwater / Level Control:						
Gain coefficient for compensated level error, K_{30}	1.5 V/V					
Gain coefficient for flow compensation, K_{31}	0.4 V/V					
Nominal Level Control Setting	66% span	57% span				Ref. (1), sh. 41 of 98 Ref. (74)
Level controller lag time constant, t_{30}	5 sec					
Level compensation reset time constant, τ_{31}	420 sec					
Flow compensation reset time constant, τ_{33}	60 sec					

Table 2.12 Main Steam Isolation Valves

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Maximum MSIV Stroke Time						
Closing Time	5 sec	5 sec	Table 10.3.1-3, 14.2.7, 14.2.12.1.50 Table 15.0.3-5	Tech. Spec. Section 4.7.1.5. Note 2.12.1	SD-126.01	Tech. Spec. Section 4.7.1.5. Note 2.12.1
Signal Delay	2 sec	2 sec	Table 15.0.3-5			Note 2.12.1
Total Response Time	7 sec	7 sec	Table 6.2.1-9, 15.1.5-4, Table 15.1.5-3			Ref. (5) Note 2.12.1

Table 2.13 Main Steam System Safety Valves

Uprate Analysis Parameter	Pre - Uprate Value		Uprate Value		Where Used and/or Reported in Uprate Documents			Uprate Source Document
	Opening Setpoint	Capacity at 3% Accumulation	Opening Setpoint	Capacity at 3% Accumulation	FSAR	Tech Spec	Other Documents	
MS-43, -44, -45	1170.0 psig ± 1%	881,980 lbm/hr	1170.0 psig ± 1%	881,980 lbm/hr		T.S. Table 3.7-2	SD-126.01	Ref. (16) Note 2.13.1
MS-46, -47, -48	1185.0 psig ± 1%	893,160 lbm/hr	1185.0 psig ± 1%	893,160 lbm/hr		T.S. Table 3.7-2	SD-126.01	Ref. (16) Note 2.13.1
MS-49, -50, -51	1200.0 psig ± 1%	904,330 lbm/hr	1200.0 psig ± 1%	904,330 lbm/hr		T.S. Table 3.7-2	SD-126.01	Ref. (16) Note 2.13.1
MS-52, -53, -54	1215.0 psig ± 1%	915,500 lbm/hr	1215.0 psig ± 1%	915,500 lbm/hr		T.S. Table 3.7-2	SD-126.01	Ref. (16) Note 2.13.1
MS-55, -56, -57	1230.0 psig ± 1%	926,670 lbm/hr	1230.0 psig ± 1%	926,670 lbm/hr		T.S. Table 3.7-2	SD-126.01	Ref. (16) Note 2.13.1

From Ref. (16), Accumulation = 3%

From Ref. (17), Blowdown = 8%

Table 2.14 Main Feedwater System Isolation Valves

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
MFW control valves (1FW-133, 1FW-191, 1FW-249 also known as FCV-478, FCV-488, FCV-498) and Isolation valves (1FW-159, 1FW-217, 1FW-277)						
Closing Time	8 sec for MFW control valve & isolation valves	8 sec for MFW control valve & isolation valves				Ref. (18)
Maximum Stroke Time	10 sec	10 sec				Note 2.14.1
MFW bypass control valves (1FW-140, 1FW-198, 1FW-256 also known as FCV-479, FCV-489, FCV-499) ~~and Isolation valves (1FW-307, 1FW-313, 1FW-319)~~. Note: Valves -307, -313, and -319 are planned to be deleted for SGR/PUR conditions per Ref. (86)						
Closing Time	10 sec for MFW bypass control & isolation valves	10 sec for MFW bypass control valves				Ref. (18)
Maximum Stroke Time	12 sec	12 sec				Note 2.14.1
Signal Delay	$\leq$ 2 sec	$\leq$ 2 sec				No Change- Note 2.14.1

Table 2.15 Auxiliary Feedwater System

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Minimum Delivery (Total Flow) of one AFW Pump to 3 SG's	417 gpm	417 gpm				Ref. (43)
Total maximum AFW flow (Faulted condition, low back pressure, pumps at runout conditions)	approx. 2600 gpm	3000 gpm				See Note 2.15.2
Minimum AFW flow for Safety Analyses	400 gpm	390 gpm (one MD pump to 3 SG's) 760 gpm (two MD pumps to 2 SG's) 1400 gpm (2 MD + 1 TD pump to 1 SG)				See Note 2.15.2 Ref. (98) and Note 2.15.3 Ref (98) and Note 2.15.4
Volume of Water swept out prior to injection of cold AFW	N/A	9 ft^3 (Hot) 50ft^3-9 ft^3 = 41 ft^3 (Ambient)	Should be added to 6.2.1.4			Ref. (67) Ref. (95)
Temperature						
Maximum	120°F	120°F				Ref. (22)
Minimum	40°F	40°F				No Change
Auxiliary Feedwater Isolation Time	41 sec	41 sec				Ref. (5), Attachment 2, sh. 3 See Note 2.15.1
Maximum Response Time to initiate AFW	61.5 sec	61.5 sec				Ref. (5), Attachment 2, sh. 3 See Note 2.15.1

Table 2.16 Primary System Parameter Measurement and Control Uncertainties

Uprate Analysis Parameter	Pre Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Reactor Power	± 2% at full power	± 2% at full power		3.2.3	DBD-314 Table 2.14	Ref. (66) & Ref. (103) See Note 2.16.3
Pressurizer Pressure	+38/-50 psi	+38/-50 psi See Note 2.16.1		3.2.5	DBD-314 Table 2.14	Ref. (66) & Ref. (100) See Note 2.16.3
Pressurizer PORV Bistable Setpoint Tolerance	N/A	+/- 4 psig				Ref (107)
Pressurizer Level Control Uncertainty	Not Modeled	+/- 0.75% of instrument span			VNG;99:200 (dated 08/19/99)	Ref. (104) Note 2.16.4
RCS Average Temperature	+5.3/-6.8°F	+6.0/-6.8°F	Not Listed	3.2.5	DBD-314 Table 2.14	Ref. (66) & Ref. (101) See Note 2.16.3
RCS Flow, Including 0.1% flow for feedwater venturi fouling	± 2.2%	± 2.2%	4.4.4.5	3.2.3	DBD-314 Table 2.14	Ref. (66) & Ref. (102) See Note 2.16.3
Startup Tolerance on RCS Average Temperature					DBD 314 Table 2.14	Ref. (66)
nominal value for transition from Mode 3 to Mode 2	± 2°F	± 2°F				
sync to grid in the nominal power range of 12% to 16%	+6.5/-6.8°F	+6.5/-6.8°F				Note 2.16.2

Table 2.16 Primary System Parameter Measurement and Control Uncertainties (Continued)

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
$F_{\Delta H}$ measurement uncertainty	4%	4%		Tech. Spec., Section 3.2.3	DBD-314 Table 2.14	Ref. (66)
F_Q measurement uncertainty	5%	5%		Tech. Spec., Section 4.2.2.2	DBD-314 Table 2.14	Ref. (66)
F_Q engineering and manufacturing tolerance	3%	3%		Tech. Spec., Section 4.2.2.2	DBD-314 Table 2.14	Ref. (66)

Table 2.17 Boron Dilution Parameters

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Cold Shutdown (Mode 1)						
Minimum RCS mixing volume (Mid Loop, RHR Cooling)	3754 ft^3	4224 ft^3	15.4.6.2			Pre-Uprate: Ref. (76), Ref. (77), Note 2.17.1 Uprate: Ref. (105), Case 19
Dilution flow	132 gpm	132 gpm	15.4.6.2			See Note 2.17.2, One makeup water pump
Required shutdown margin	Figure 1 in Attachment 9 to Ref. (5)	Figure 1 in Attachment 9 to Ref. (5)			COLR (PLP-106)	Figure 1 in Attachment 9 to Ref. (5)
High Flux at Shutdown Alarm Setpoint	2 times above background levels	2 times above background levels				Plant Operations assures that the setpoint is maintained in Mode 5. Ref. (55), (56) are the setpoint surveillance. Ref. (39) is the General Plant Procedure which initiates the setpoint change.
Hot Shutdown (Mode 4)						
Minimum RCS mixing volume—No RCP's, Single RHR, No Pzr	4918 ft^3	4642 ft^3	15.4.6.2			Pre-Uprate: Ref. (76), Ref. (77), Note 2.17.1 Uprate: Ref. (105), Case 15
Minimum RCS mixing volume—One RCP, Single RHR, No Pzr	7450 ft^3	9241 ft^3	15.4.6.2			Pre-Uprate: Ref. (76), Ref. (77), Note 2.17.1 Uprate: Ref. (105), Case 11

Table 2.17 Boron Dilution Parameters (continued)

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Hot Shutdown (Mode 4, Continued)						
Dilution flow	321 gpm	321 gpm	15.4.6.2			Pump curves in section S1e of (Ref. 58), a reasonably high value for the combination of two reactor makeup water pumps.
Required shutdown margin	Figure 1 in Attachment 9 of Ref (5)	Figure 1 in Attachment 9 of Ref (5)			COLR (PLP-106)	Figure 1 in Attachment 9 of Ref (5)
Hot Standby (Mode 3)						
RCS mixing volume, No Pzr, No RHR	7450 ft^3	8828 ft^3	15.4.6.2			Pre-Uprate: Ref. (76), Ref. (77), Note 2.17.1 Uprate: Ref. (105), Case 6
Dilution flow	321 gpm	321 gpm	15.4.6.2			Ref. (58) Two reactor makeup water pumps
Required shutdown margin	Figure 1 in Attachment 9 of Ref (5)	Figure 1 in Attachment 9 of Ref (5)			COLR (PLP-106)	Figure 1 in Attachment 9 of Ref (5)
Startup (Mode 2)						
RCS mixing volume, No Pzr	7450 ft^3	9085 ft^3	15.4.6.2			Ref. (76), Ref. (77), Note 2.17.1 Uprate: Ref. (105), Case 4

Table 2.17 Boron Dilution Parameters (continued)

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Startup (Mode 2, Continued)						
Dilution flow	337 gpm	337 gpm	15.4.6.2			Ref. (11), Conservative charging flow.
Required shutdown margin	1770 pcm	1770 pcm	15.4.6.2	T.S. 3.1.1.1		Tech. Spec. 3.1.1.1
Full Power (Mode 1)						
RCS mixing volume, No Pzr	7450 ft^3	9085 ft^3	15.4.6.2			Pre-Uprate: Ref. (76), Ref. (77), Note 2.17.1 Uprate: Ref. (105), Case 2
Dilution flow	337 gpm	337 gpm	15.4.6.2			Ref. (11) Conservative charging flow
Required shutdown margin	1770 pcm	1770 pcm	15.4.6.2	T.S. 3.1.1.1		Tech. Spec. 3.1.1.1

Table 2.18 ESF Actuation Setpoints

Uprate Analysis Parameter	Pre - Uprate Value			Uprate Value			Where Used and/or Reported in Uprate Documents			Uprate Source Document
	Trip Setpoint	Tech. Spec. Total Allowance	Analysis Value	Trip Setpoint	Tech. Spec. Total Allowance	Analysis Value	FSAR	Tech Spec	Other Documents	
Safety Injection on High Containment Pressure	≤ 3 psig	+(0.027) (55 psi)	4.5 psig	≤ 3 psig	+(0.027) (55 psi)	4.5 psig		Tech. Spec. Table 3.3-4	DBD-314, Rev 3, Table	Ref. (66)
Containment Spray on Containment Pressure - High	≤ 10 psig	+(0.036) (55 psi)	12.0 psig	≤ 10 psig	+(0.036) (55 psi)	12.0 psig		Tech. Spec. Table 3.3-4	DBD-314, Rev 3, Table	Ref. (66)
Safety Injection on Low Pressurizer Pressure	≥ 1850 psig	-(0.188) (800 psi)	1699.6 psig	≥ 1850 psig	-(0.188) (800 psi)	1699.6 psig		Tech. Spec. Table 3.3-4	DBD-314, Rev 3, Table	Ref. (66)
Safety Injection on Low Steam Line Pressure	≥ 601 psig See Note 2.18.1 below	-(0.177) (1300 psi)	370.9 psig	≥ 601 psig See Note 2.18.1 below	-(0.177) (1300 psi)	370.9 psig		Tech. Spec. Table 3.3-4	DBD-314, Rev 3, Table	Ref. (66)
Auxiliary Feedwater on SG Water Level--Low-Low (Loss of Normal Feedwater)	≥ 38.5% of Narrow Range Span	-(0.192) (100%)	19.3% span	**≥ 25.0% of Narrow Range Span**		**16.1% of Narrow Range span**		Tech. Spec. Table 3.3-4	DBD-314, Rev 3, Table	Ref. (66)
Auxiliary Feedwater on SG Water Level--Low-Low (Main Feed Line Break)	≥ 38.5% of Narrow Range Span	-(0.192) (100%)	19.3% span	**≥ 25.0% of Narrow Range Span**		**0.0% of Narrow Range span**		Tech. Spec. Table 3.3-4	DBD-314, Rev 3, Table	Ref. (66)

Table 2.18 ESF Actuation Setpoints (continued)

Uprate Analysis Parameter	Pre - Uprate Value			Uprate Value			Where Used and/or Reported in Uprate Documents			Uprate Source Document
	Trip Setpoint	Tech. Spec. Total Allowance	Analysis Value	Trip Setpoint	Tech. Spec. Total Allowance	Analysis Value	FSAR	Tech Spec	Other Documents	
Turbine Trip and Main Feedwater Isolation on SG Water Level--High-High	≤ 82.4% of Narrow Range Span	+(0.15) (100%)	97.4% span	≤ 78.0% of Narrow Range Span Response Time: 2.5sec		100.0% of Narrow Range span Response Time: 2.5sec	Table 15.0.6-2	Tech. Spec. Table 3.3-4	DBD-314, Rev 3, Table	Ref. (74) - Trip Setpoint Ref. (66) - Analysis Value Ref (5) Response Time
Auxiliary Feedwater Isolation on High Steam Line Differential Pressure coincident with Steam Line Isolation	≤ 100 psi	+(0.05) (1300 psi)	165 psi	≤ 100 psi	+(0.05) (1300 psi)	165 psi		Tech. Spec. Table 3.3-4	DBD-314, Rev 3, Table	Ref. (66)
Main Steam Line Isolation on Low Steam Line Pressure	≥ 601 psig See Note 1 below	(0.177) (1300 psi)	370.9 psig	≥ 601 psig See Note 1 below	(0.177) (1300 psi)	370.9 psig		Tech. Spec. Table 3.3-4	DBD-314, Rev 3, Table	Ref. (66)

Table 2.19 ESF High Head Safety Injection

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Maximum ECCS Response Time (to get water into the core)	27 sec	29 sec	6.3.2-13a 6.3.3-8			Diesel generator starting and sequence loading delays included, Ref. (5), Attachment 2. See Notes 2.19.1 and 2.19.2. See Note 2.19.4 for uprate value.
Maximum RWST to CSIP opening time (to get boron into the core)	27 sec	27 sec				Not included in ECCS response time. Diesel generator starting and sequence loading delays included. Ref. (5), Attachment 2 footnote 10. See Note 2.19.1.
Maximum VCT to CSIP closing time (to get boron into the core)	37 sec	37 sec				Not included in ECCS response time. Diesel generator starting and sequence loading delays included. Ref. (5), Attachment 2, footnote 10. See Note 2.19.1
Alternate Miniflow Valve (1CS-746 and 1CS-752) open/close times		15 sec			PLP-106	Not included in ECCS response time. Ref. (5), Attachment 2 footnote 12.
Boron Injection Tank Volume	900 gal	900 gal				Ref. (7) See note 2.19.5 for boration/dilution assumptions about BIT and ECCS piping.

Table 2.19 ESF High Head Safety Injection (Continued)

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Non-Borated Water Volume Injected (Including the CSIP pump volume, and non-borated flow during closure time of VCT outlet valves) before borated water arrives at CSIP	Not Modeled	42 ft**3				Ref. (75) -- Calculation HNP-F/NFSA-0034, Rev 2. Approval to use/release value of CSIP fluid volume in Ref. (96)

Table 2.19 ESF High Head Safety Injection (continued)

The degraded pump head curves for use in the Safety Analysis are listed below. (Ref. 81 and 82) This table represents the minimum allowable or acceptable performance (Ref. 81 and 82). Ref.(32) discusses variation in "B" Charging pump flows, and was an original DBD-314 reference.

Flow (gpm)	Pump Head (psid)	Pump Head (ft)
0.	2446.	5714.7
50.	2446.	5714.7
100.	2446.	5714.7
150.	2403.	5614.75
200.	2380.	5514.8
250.	2318.	5414.7
300.	2253.	5264.7
350.	2146.	5014.7
400.	2018.	4714.7
450.	1890.	4414.7
500.	1718.	4014.7
550.	1590.	3714.7
600.	1376.	3214.75
650.	1205.	2814.7
700.	1034.	2414.7

at a fluid density of 61.635 lb/ft^3

Table 2.19 ESF High Head Safety Injection (continued)

The hydraulic characteristics of the HHSI system in the LBLOCA SIS model are benchmarked to match the total pump flow and flow distribution specified in Technical Specification Section 4.5.2.h.1. For charging/safety injection pump lines with one pump running, the sum of the injection line flow rates, excluding the highest flow rate, is ≥ 348 gpm and the total flow rate is ≤ 625 gpm (See Note 2.19.3).

In the SBLOCA model, the hydraulic characteristics of the HHSI system are benchmarked to give a maximum flow rate equal to the Technical Specification minimum flow rate, i.e., 174 gpm to each of the 3 loops. This charging/safety pump flow distribution gives a total SI flow of 522 gpm and conservatively minimizes the SI flow that would reach the reactor vessel.

Safety Injection Flow Rate for MFLB Analysis

The following SI flow table was used in the Westinghouse Uprate SGTR analysis. It is based on a Westinghouse calculation and modified to reflect the new impeller per CP&L letter HW/99-104, 6/24/99, item 22. The Westinghouse Reference document is CQL-99-129, 10/14/99, item #5 IAM for SGTR (proprietary to Westinghouse) which provides the SI flow for Maximum Safeguards/ Minimum Flow Resistances (where high flow rate is conservative). The values used in the SGTR analysis are listed below:

Pressure (psia)	Flow (gpm)
614 .7	822.8
1014.7	744.7
1214.7	703.4
1514.7	639.2
1714.7	592.0
1914.7	537.6
2014.7	508.7
2114.7	473.9
2214.7	436.5
2314.7	396.0
2414.7	351.0
2514.7	299.5

The Siemens analysis of MFLB was reviewed in a Technical Review meeting in October, 1999. Based on that review, there were a number of CP&L "challenges" to assumptions used in the analysis. SI Flowrate from Westinghouse SGTR event analysis is provided here in case resolution of these challenges involves use of maximum SI flow (vs. the minimum SI flow on the previous page).

Table 2.20 ESF Low Head Safety Injection

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Maximum Response Time	27 sec	29 sec				Diesel generator starting and sequence loading delays included, Ref. (5), Attachment 2. See Note 2.20.1 and 2.20.2
Maximum flow split in any loop	≤ 50%	≤ 50%				Ref. (6)

Table 2.20 ESF Low Head Safety Injection (continued)

The pump head curve for RHR Pump 1A-SA [Ref. (35)] was used in LBLOCA analysis. This pump head curve is listed below:

Flow (gpm)	Discharge Pressure (psig)	Suction Pressure (psig)
1513.0	170.0	42.3
1947.0	168.0	41.4
2371.0	163.0	40.3
2964.0	156.0	39.2
3267.0	150.0	38.6
3742.0	140.0	37.1
4320.0	126.0	35.6

Pump performance is specified by Tech. Spec. 4.5.2.h.2. For RHR pump lines, with one pump running, the sum of the injection line flow rates is greater than or equal to 3663 gpm.

Technical Specification 4.5.2.f.2 provides a minimum pump pressure differential of 100 psid at a flow rate of at least 3663 gpm.

The above pump head curve [Ref. (35)] is adjusted in the Safety Analysis to be consistent with Technical Specification 4.5.2.f.2. The overall LHSI model is benchmarked against the total flow rate specified in Technical Specification 4.5.2.h.2.

Table 2.21 ESF Accumulators

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Nitrogen Blanket Pressure						
Minimum	585 psig	585 psig				Tech. Spec. 3.5.1.d
Maximum	665 psig	665 psig				Idem.
Maximum Operating Temperature	130°F	130°F				Conservative relative to Tech. Spec., Section 3.6.1.5
Borated Water Volume (each of three)						
Maximum	96% indicated level (1029.4 ft^3) See Note 2.21.1.	96% indicated level (1029.4 ft^3) See Note 2.21.1.				Tech. Spec. 3.5.1.b, Ref. (51)
Minimum	66% indicated level (7440 gal. or 994.6 ft^3)See Note 2.21.1.	66% indicated level (7440 gal. or 994.6 ft^3)See Note 2.21.1.				Idem.

Table 2.22 ESF Containment Cooling, Containment Spray

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Flowrate						
Maximum	4745 gpm @ 0 psig	5000 gpm @ 0 psig				2375 gpm from Train A plus 2370 gpm from Train B on page 43 of 43 of Ref. (52) See Note 2.22.2
Response Time						
Minimum	29.3 sec	0 sec				See Note 2.22.1 and 2.22.2. Page 84 of 125 of Ref. (53)

Table 2.23 ESF Containment Cooling, Reactor Containment Fan Coolers/Air Recirculation Cooling Units (HVN)

Figure 2.1 depicts Heat Removal Rate per Fan vs. Containment Temperature from page S2-34 of Ref. (59). The Heat Removal Rates shown here are conservatively high relative to the corresponding values in Section X-6 of Ref. (54).

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Response Time						
Minimum	21.08 sec	**0 sec**	6.2.1-167			See Note 2.23.1

Table 2.24 ESF Refueling Water Storage Tank

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Boron Concentration						
Maximum	2600 ppm	2600 ppm		Tech. Spec., Section 3.5.4		Tech. Spec., Section 3.5.4
Minimum	2400 ppm	2400 ppm		Tech. Spec., Section 3.5.4		Tech. Spec., Section 3.5.4
Temperature						
Maximum	125°F	125°F		Tech. Spec., Section 3.5.4		Tech. Spec., Section 3.5.4
Minimum	40°F	40°F		Tech. Spec., Section 3.5.4		Tech. Spec., Section 3.5.4

Table 2.25 ESF Reactor Containment Building

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Net Free Volume						
Nominal	2.266×10^6 ft^3	2.266×10^6 ft^3				Ref. (92)
Maximum	2.344×10^6 ft^3	2.344×10^6 ft^3				Ref. (75) -- Calculation HNP-F/NFSA-0034, Rev 2.
Containment Pressure Initial Range		**-1.0" water gage to 1.6 psig**		3.6.1.4		Tech Spec 3.6.1.4
Containment Pressure Uncertainty		**1.32 psig**		Bases 3/4.6.1.4		Bases 3/4.6.1.4
Heat Sinks	FSAR Table 6.2.1-63 (as referenced in EMF-93-033(P))	Unchanged	6.2.1		Bechtel markups of Calculation3-A-05-001 for SGR project	Ref. (90) and Calculation HNP-F/NFSA-0034, Rev 2.
Containment Temperature						
Minimum (winter)	80°F	80°F				Ref. (49) Ref. (85)

Table 2.26 Automatic Rod Control System

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Automatic Rod Control enabled?	Yes, if conservative for event	Yes, if conservative for event				Ref.(10), Note 2.26.2
ARC Programmed Deadband Width	+/- 1.5°F	+/- 1.5°F				Ref. (1), sh. 26
Maximum rod speed	72 steps/min.	72 steps/min.				Ref. (1), sh. 26
Proportional Range rod speed	32 steps/min./°F	32 steps/min./°F				Ref. (1), sh. 34
Minimum non-zero rod speed	8 steps/min.	8 steps/min.				Ref. (1), sh. 26
ARC Lockup Setpoint	+/- 0.5°F	+/- 0.5°F				Ref. (1), sh. 26
Temperature Channel:						
Temperature gain	See Note 2.26.1 below	0.318 °F/ % Pwr				Ref. (1), sh. 25
Low limit	557°F	557°F				Idem.
Lead time constant, t_3	40 sec	40 sec				Idem.

Table 2.26 Automatic Rod Control System (continued)

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Lag time constant						
τ_2	30 sec	30 sec				Idem.
τ_4	10 sec	10 sec				Idem.
τ_5	5 sec	5 sec				Idem.
Power Mismatch Channel:						
Impulse unit time constant, τ_1	40 sec	40 sec				Ref. (1), sh. 26, Ref. (66)
Impulse unit low gain	0.3°F/%	0.3°F/%				Idem.
Impulse unit high gain	1.5°F/%	1.5°F/%				Idem.
Switch point for non-linear gain, K_1	±1% error	±1% error				Idem.
Variable gain at 100%	1.0	1.0				Idem.
Variable gain below 50%	2.0	2.0				Idem.
Variable gain between 50% and 100%	Inversely proportional to turbine power	Inversely proportional to turbine power				Idem.

Table 2.27 Turbine Runback Control System

When either permissive and interlock circuit C-3 or permissive and interlock circuit C-4 are actuated, automatic turbine power runback will take place at the rate of a 5% decrease every 30 seconds until these circuits are no longer actuated (Ref. (1), sheet 10 of 98).

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
Over-temperature ΔT Turbine Runback (Ref. (1), sheet 15 of 98):						
ΔT turbine runback setpoint = ΔT reactor trip setpoint - 3%						
Turbine runback time delay relay						
on	1.5 seconds	1.5 seconds				Ref. (1) sh.15
off	28.5 seconds	28.5 seconds				Ref. (1) sh.15
Turbine load reference reduction rate	200%/minute	200%/minute				Ref. (1) sh.15
Over-power ΔT Turbine Runback (Ref. (1), sheet 17 of 98):						
ΔT turbine runback setpoint = ΔT reactor trip setpoint - 1.9%						
Turbine runback time delay relay						
on	1.5 seconds	1.5 seconds				Ref. (1) sh.15
off	28.5 seconds	28.5 seconds				Ref. (1) sh.15
Turbine load reference reduction rate	200%/minute	200%/minute				Ref. (1) sh.15

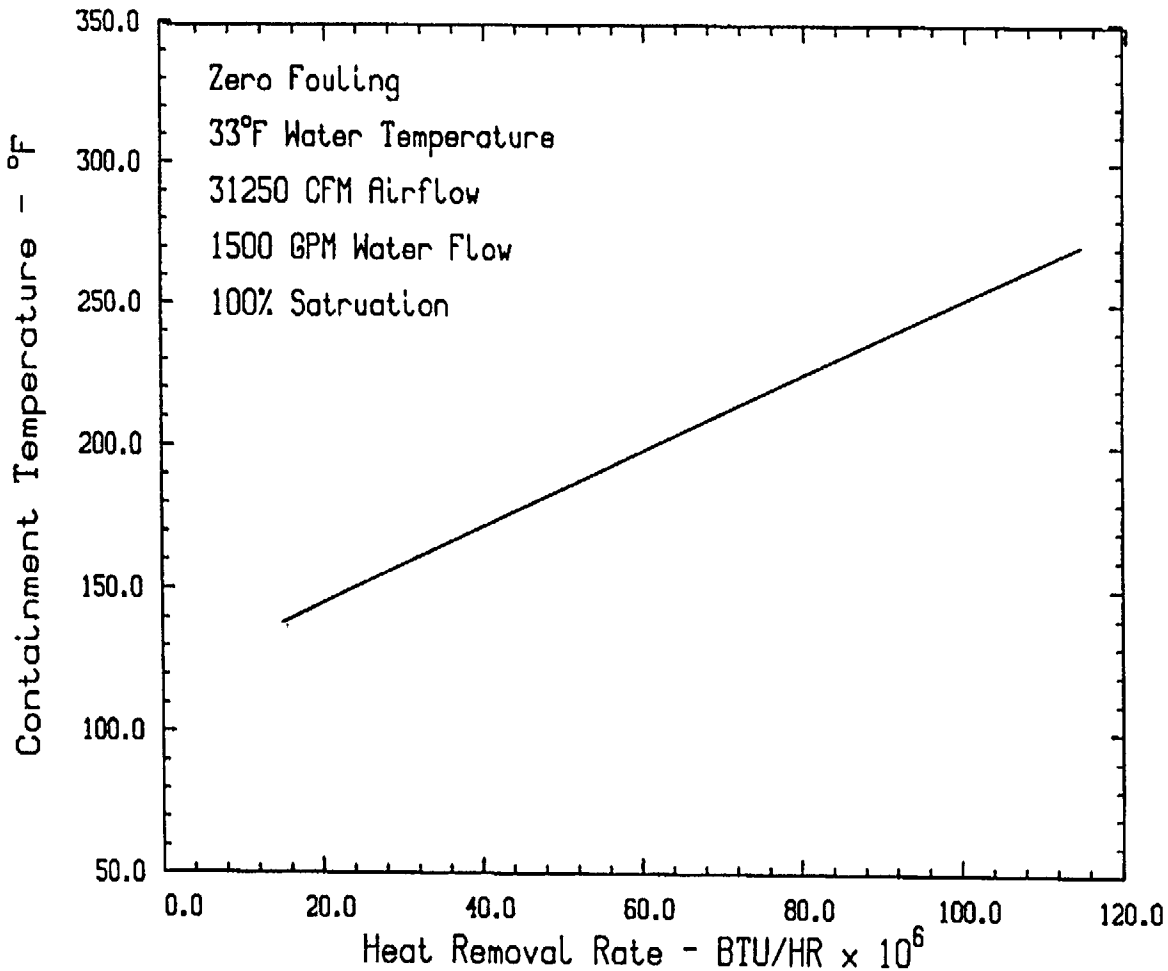
Table 2.28 Operator Actions

Uprate Analysis Parameter	Pre - Uprate Value	Uprate Value	Where Used and/or Reported in Uprate Documents			Uprate Source Document
			FSAR	Tech Spec	Other Documents	
RCP Trip Criteria						Ref. (41)
If both of the following occur, then stop all RCP's:						
SI flow greater than 200 gpm						
RCS pressure less than 1360 psig						
Intact SG Levels						Ref. (42)
Control feed flow to maintain all intact levels between 10% and 50%.						

Table 2.29 Steam Generator Replacement ONLY Parameters Which Differ From SGR/PUR Values

SGR Only Analysis Parameter	SGR/PUR Value	SGR Only Value	Where Used and/or Reported in Uprate Documents			SGR Only Source Document
			FSAR	Tech Spec	Other Documents	
Table 2.1						
Power	2900 MWt	2775 MWt				Ref. (97)
Net Heat Input	12.4 MWt	12.4 MWt				Ref. (97)
NSSS Power	2912.4 MWt	2787.4 MWt				Ref. (97)
Steam generator Secondary Pressure (psia) @ 100% power, Tech Spec Minimum Flow, FW Temp = 440 degrees	T_{avg} 588.8°F 989 (0% Tube Plugging) 985 (3% Tube Plugging) T_{avg} 580.8°F 920 (0% Tube Plugging) 916 (3% Tube Plugging)	T_{avg} 588.8°F 993 (0% Tube Plugging) 978 (10% Tube Plugging) T_{avg} 572°F 852 (0% Tube Plugging) 838 (10% Tube Plugging)				Ref. (97)
Total Steam Flow (10^6 Lbm/hr)	T_{avg} 588.8°F 12.85 T_{avg} 580.8°F 12.81	T_{avg} 588.8°F 12.30 T_{avg} 572°F 12.23				Ref. (97)
Table 2.2						
ΔT_0	Normalized to 2900 MWt ΔT	Normalized to 2775 MWt ΔT				Ref. (75)

Figure 2.1 Heat Removal Rate of Emergency Cooler Unit



3. References

1. EMDRAC 1364-053872, Revision 18, *Precautions, Limitations and Setpoints*

2. Siemens Letter VNG:99:037, *Transmittal of Analysis Value Trip Coefficients for the OTΔT and OPΔT Trips,* February 11, 1999.

3. *Shearon Harris Cycle 6: Thermal Hydraulic Compatibility Analysis*, EMF-93-194(P), March 1994. This document is proprietary to Siemens Power Corporation.

4. Westinghouse Report WNEP-9517 Revision 2, *Delta 75 Steam Generator Thermal and Hydraulic Design Data Report for Shearon Harris Nuclear Station,* June 15, 1998.

5. Plant procedure PLP-106, Revision 20, *Technical Specification Equipment List Program and Core Operating Limits Report.*

6. HNP ESR 97-00378, Revision 1, *OST-1088 Low Head RHR Low Flow Through Branch Line to RCS.*

7. EMDRAC 1364-2106, Revision 5, *Boron Injection Tanks Units 1 through 4.*

8. Not Used.

9. *Operations Curve Book* (controlled by plant procedure OMM-010).

10. Siemens Report EMF-93-163, *Shearon Harris Cycle 6: Disposition of FSAR Chapter 15 Events and Analysis of Plant Transients*, March 1994.

11. Plant Change Request - 6831, dated 08/15/94, *1A-SA and 1B-SB CSIP Replacement.*

12. EMDRAC 1364-2010, Revision 12, *Reactor Coolant Pump & Motor Outline,* (Westinghouse Number 115E110).

13. Vendor Manual MRF, Revision 14, *Pumps, Reactor Coolant.*

14. *Data Reports for Pressurizer Safety Relief Valves*, STAIRS DIN 87 315 2980.

15. EMDRAC 1364-096543, Revision 0, *Pressurizer Safety Valve Set Pressure Shift,* (Vendor ID is WCAP-12910, March 1991).This document is Westinghouse Proprietary Class 2.

16. CP&L Specification CAR-SH-M-055, Revision 2, *Main Steam Safety Valves.*

17. Field Change Request - M-1976,dated July 18, 1986, *Reset Main Steam Safety Valve Rings.*

18. Plant Change Request - 4628, Revision 0, *Pre-Heater Bypass and FW Stroke Time.*

19. Memo with attachments, R.T. Meyer (Westinghouse) to T. B. Clements (CP&L), *Carolina Power & Light Shearon Harris Unit 1 Integrated Safety Evaluation for ECCS Inconsistencies,* August 7, 1990 (90CP*-G-0080)—included in Reference 20.

20. Plant Change Request - 5021, Revision1, *ECCS Flow Inconsistencies.*

21. HNP Calculation FW-18, Revision1, *Accident Analysis.*

22. Design Basis Document 114, *Auxiliary Feedwater System.*

23. HNP calculation AF-45, Revision 0, *Evaluation of TDAFW Pump Runout.*

24. Not Used.

25. CP&L Drawing CAR 2166 B-401, Sheet 151, Revision 9, *(Control and Wiring Diagram for) Pressurizer Heater Control Group C.*

26. CP&L Drawing CAR 2166 B-401, Sheet 152, Revision 12, *(Control and Wiring Diagram for) Pressurizer Heater Backup Group A.*

27. CP&L Drawing CAR 2166 B-401, Sheet 153, Revision 13, *(Control and Wiring Diagram for) Pressurizer Heater Backup Group B.*

28. CP&L Drawing CAR 2166 B-401, Sheet 154, Revision 8, *(Control and Wiring Diagram for) Pressurizer Heater Backup Group D.*

29. Plant procedure MST-I0119, *Pressurizer Pressure P-0455.*

30. Plant procedure MST-I0120, *Pressurizer Pressure P-0456.*

31. Plant procedure MST-I0121, *Pressurizer Pressure P-0457.*

32. HNP ESR 97-00620, Revision 1, *'B' CSIP Rotating Element Replacement.*

33. Not Used.

34. Not Used.

35. Vendor Manual BJH, Volume 4, Revision 21, *Pumps, Residual Heat Removal.*

36. EMDRAC 1364-096544, Revision 0, *EPRI PWR Safety and Relief Valve Test Program Safety and Relief Valve Test Report,* (Vendor ID is EPRI NP-2628SR, December 1982).

37. Vendor Manual FAY, Revision 7, *Pressurizer.*

38. Plant procedure OST-1503, Revision 5, *Pressurizer PORV Operability Quarterly Interval Mode 5,* **NOTE: This procedure status is "On-Hold, Do Not Use" as of 03/09/99. Values referenced to this procedure must be validated prior to approval of the SGR/PUR ESR's which depend on the Siemens analysis done under this UFAPPD.**

39. Plant procedure GP-007, Revision 20, *Normal Plant Cooldown (Mode 3 to Mode 5).*

40. Not Used.

41. Plant procedure EOP-Guide-1, Revision 8, *(Emergency Operating Procedure [EOP] Path Guide Procedure) Path-1 Guide.*

42. Plant procedure EOP-EPP-009, Revision 9, *(Emergency Operating Procedure [EOP] End Path Procedure [EPP]) Post LOCA Cooldown and Depressurization.*

43. HNP calculation AF-42, Revision 0, *AFW Minimum Flowrate Study.*

44. Plant procedure MST-I0037, *Delta T/Tavg Loop (T-0412) Calibration.*

45. Not Used.

46. Not Used.

47. Not Used.

48. Not Used.

49. Design Basis Document 136, Revision 5, *Containment Ventilation & Cooling Systems.*

50. Not Used.

51. EMDRAC 1364-096547, Revision 0, excerpt from W letter FCQL-366 (dated 10-7-85).

52. HNP calculation CT-25, Revision 0, *Containment Spray Pump Flow Determination.*

53. HNP calculation CT-28, Revision 1, *To Determine Spray Header Fill-up Time.*

54. HNP calculation 9-RCB-1A, Revision 3, *Reactor Containment Cooling System.*

55. Plant procedure MST-I0050, *Nuclear Instrumentation System Source Range N31 Calibration.*

56. Plant procedure MST-0051, *Nuclear Instrumentation System Source Range N32 Calibration.*

57. Plant procedure GP-008, *Draining the Reactor Coolant System.*

58. Vendor Manual BHP, Revision 18, (Crane Company, Chempump Division) *Pumps.*

59. Vendor Manual ORS, Revision 9, *Instruction Manual for the Containment Fan Coolers.*

60. Not Used.

61. HNP Calculation, 3-E-8-002, Revision 0, *Pressurizer SRV, PORV, Block Valve Operability.*

62. CP&L Letter, *Harris SGR/PUR Fuel Design Information - Final PCWG Parameters,* Serial NF-98A-214, File NF-926, dated 08/04/98; transmitting uprate PCWG information to SPC from Westinghouse letter CQL-98-030, Rev 1 dated 07/08/98.

63. CP&L Letter, *Harris SGR/PUR Setpoint OPDT/OTDT Analysis Design Information*, NF-98A-385, File NF-926, dated 12/08/98; transmitting uprate OPDT/OTDT information to SPC.

64. Textbook, M.M. El-Wakil, *Nuclear Heat Transport*, 1971, page 430.

65. Westinghouse Letter, *Final PCWG Parameters for the SGR/Uprating Analysis and Licensing Project*, CQL-98-030, Rev. 1 dated July 8, 1998.

66. CP&L Letter, *Harris SGR/PUR Fuel Design Information*, Serial NF-98A-0211, File NF-926, dated July 9, 1998.

67. EMDRAC1364-2409 Rev. 1, CP&L Drawing, *Pressurizer General Arrangement*.

68. EMDRAC1364-2440 Rev. 2, CP&L Drawing, *Pressurizer Outline*.

69. Siemens Calculation E-5847-593-003, *Shearon Harris ANF-RELAP Base Mode Development for Cycle 6*, page 6.5-20, 11/4/93.

70. HNP calculation E2-010, Revision 5, *Undervoltage Relays: RCP Motors 1A, 1B, 1C*.

71. HNP calculation E2-011, Revision 4, *Underfrequency Relays: RCP Motors 1A, 1B, 1C*.

72. Not Used.

73. WCAP 14778, dated December 17, 1996, *SGR Delta 75 Engineering Report*, Books 1 and 2.

74. Westinghouse Letter, *Revised RSG Level & Trip Setpoints in Consideration of Moisture Separator Modifications*, CQL-98-050, November 3, 1998.

75. HNP Calculation, HNP-F/NFSA-0034, Revision 3, HNP SGR/PUR Fuel Related Design Input Calculations.

76. Westinghouse Letter, *RCS Boron Dilution Volumes*, 97-CQL-042, June 23, 1997.

77. HNP Calculation, HNP-M-MECH-1020, dated 3/26/98, *Reactor Vessel and Reactor Coolant system Water Volumes for Shearon Harris Nuclear Power Plant*.

78. EMDRAC 1364-1729 Rev 3, CP&L Drawing, *RCP Casing Weld Details 2 SH*.

79. Westinghouse Letter, *RCS Heat Balance, Net RCP Heat Input, and Power Calculations at a Reduced Thot Temperatures*, 94-CQL-050/MIP-CQL-1247, May 17, 1994.

80. Westinghouse Design Specification, *Delta 75 Steam Generator*, Spec 412A86, 06/05/95, Revision 1.

81. CP&L Letter *Emergency Work Authorization for H1C9: Disposition of UFSAR Chapter 15 Events*, Serial NF-98A-371, File NF-404.1104, NF-609.04, dated November 21, 1998.

82. Siemens Letter *Disposition of Degraded Charging/Safety Injection Pump Flow on Harris Chapter 15 Events*, Serial TEM:98:351, CP&L Files NF-2001.9874, NF-404.1104, dated November 22, 1998.

83. EMF-93-033(P), *Plant Parameters for Shearon Harris Nuclear Power Plant*, March 1994, [Harris Cycle 6 PPD].

84. EMDRAC 1364-053067, Revision 3, *Westinghouse Setpoint Methodology for Protection Systems, Shearon Harris*.

85. HNP Calculation, 9-ARCB-1A, Revision 3 3/5/95, *Reactor Containment Cooling System*,.

86. HNP ESR/MOD 98-00537, *Tempering Lines Modification*, currently "Assigned", not complete; no revision number, no date available.

87. HNP ESR/ED 99-00116, Revision 0, 03/22/99, *Review PZR Spray flow value in DBD-314*.

88. EMDRAC 1364-096549, *Westinghouse ITDP Instrument Uncertainty Methodology...*, Revision 0 (Vendor ID is WCAP-12340 dated 09/1989).This document is Westinghouse Proprietary Class 2.

89. CP&L Letter *HNP SGR/PUR CP&L Approval of Final OPΔT/OTΔT Setpoints and Tau's*, Serial HS/99-008, dated August 25, 1999.

90. HNP Calculation 3-A-05-001, *Heat Sinks Calculation for Containment Analysis*, Revision 0, February, 1980

91. HNP Calculation HNP-F/NFSA-0061, Revision 2, 09/03/99, *HNP Cycle 10 Changes to the Plant Parameters Document*

92. CP&L Letter, *Revised Response to WX885 (Maximum Credible Accident Sources) Data Request*, Serial/File HW/99-117, dated July 9, 1999.

93. Westinghouse Letter CQL-99-085, transmitting Calculation, CN-SSO-99-13, Revision 1, *Nuclear Instrumentation System Power Range Protection Functions for Harris Uprate to 2912.4 MWt-NSSS Power*, Westinghouse Proprietary

94. Westinghouse Letter CQL-99-084, transmitting Calculation, CN-SSO-99-5, Revision 1, *Pressurizer Water Level - High Reactor Trip Setpoint Uncertainty for Harris Uprate to 2912.4 MWt-NSSS Power*, Westinghouse Proprietary

95. CP&L Letter HW/99-094, "Analysis Inputs [for Steamline Break Mass & Energy Releases]," dated June 14, 1999.

96. Westinghouse Letter CQL-99-087, "Release of CSIP Pump Fluid Volume to Siemens," dated September 10, 1999

97. Westinghouse Letter CQL-99-125, "Final PCWG Parameters for SGR conditions at Current Power WX893," dated October 12, 1999

98. HNP Calculation AF-23, "AFW System Flowrate Study," Revision 0, dated 05/05/1988.

99. HNP ESR 97-0808, Revision 0, DRAFT Under Review as of 11/12/99, pages 13.425 through 13.428

100. Westinghouse Letter CQL-99-093, transmitting Calculation CN-SSO-99-9, Rev. 0, "Harris Pressurizer Pressure Automatic Control System Uncertainty and Technical Specification DNB Parameter Pressurizer Pressure Limit for Uprate to 2912.4 Mwt - NSSS Power", Westinghouse Proprietary.

101. Westinghouse Letter CQL-99-150, transmitting Calculation CN-SSO-99-10, Rev. 1, "Harris Tavg Automatic Control System Uncertainty and Technical Specification DNB Parameter Tavg Limit for 2912.4 Mwt - NSSS Power," Westinghouse Proprietary.

102. Westinghouse Letter CQL-99-150, transmitting Calculation CN-SSO-99-11, Rev. 1, "Harris Calorimetric and Loop RCS Flow Uncertainty for Uprate to 2912.4 Mwt - NSSS Power," Westinghouse Proprietary.

103. Westinghouse Letter CQL-99-150, transmitting Calculation CN-SSO-99-12, Rev. 1, "Harris Calorimetric Power Measurement Uncertainty for Uprate to 2912.4 Mwt - NSSS Power," Westinghouse Proprietary.

104. Westinghouse Letter CQL-99-083, transmitting Calculation CN-SSO-99-6, Rev. 1, "Pressurizer Water Level Automatic Control System Uncertainty Calculation for Uprate to 2912.4 Mwt - NSSS Power," Westinghouse Proprietary.

105. Westinghouse Letter CQL-99-115, transmitting Calculation CN-FSE-99-108, Rev.1, "RCS Volume and Boron Inventory Analysis [for Uprate & Steam Generator Replacement]," Westinghouse Proprietary.

106. Bechtel Letter BH/99-161, transmitting ESR for approval by CP&L, "Second Submittal of 100% ESR 97-00807 [Revision 0], Large Bore Piping Modifications," November 3, 1999.

107. HNP Calculation SC-N-064, "Pressurizer Pressure Control PT-445," Rev.4 , 09/03/98

108. HNP Drawing 2166-B-430, Sheet 23.1, Rev. 9, "Instrument Schematics and Logic Diagrams; Steam Generator 1A Blowdown to Flash Tank;" and Sheet 23.8, Rev. 3, "Instrument Schematics and Logic diagrams; Steam Generator 1A Blowdown Isolation Valves."

4. Notes

Table 2.1 Notes

2.1.1 Page 25 of 98 of Ref. (1) presents 557°F as the nominal lower limit, independent of the Tavg program.

2.1.2 See Reference 62 for additional combinations of tube plugging and T_{avg}.The FW temperature range reflects two separate Westinghouse predictions of "nominal" conditions. SPC should choose conservative values from this range for specific Chapter 15 safety analysis calculations.

2.1.3 The use of a Low Tavg contingency bounds the EOC coastdown of 7.0°F for the SGR/PUR project. No additional coastdown assumption is required.

Table 2.2 Notes

2.2.1 The Reactor Trip System Response Time is defined as the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage. (Tech. Spec. definition 1.29).

2.2.2 The lead/lag time constants were obtained from Ref. (29), Ref. (30), and Ref. (31). Sheet 20 of 98 of Ref. (1) is an additional reference, but only the MST's present the $\pm$10% tolerance

2.2.3 A specific undervoltage setpoint should not be assumed in the analysis.

2.2.4 Calibration tolerance on the Tech. Spec. values for all first order time constants is $\pm$10%. Pages 134, 135, 138, 158, & 159 of 183 in Ref. (44) are a typical example for RCS loop 1.

2.2.5 Note that in Reference (84), Table 3-19 states that Pressurizer Water Level-High is "not used in safety analysis." If SPC uses this value in Safety Analysis, then CP&L must be notified so that the change to this reference document can be made. The Westinghouse "Tech Spec Allowable" value in "Two Column" format for Uprate is 92.5%. Since the larger value is conserative, and for consistency with the current revision of Ref. (84), the 93.8% Analysis Value continues to be appropriate for analysis purposes.

Table 2.3 Notes

2.3.1 The Tech. Spec. Limits on $F_{\Delta H}$ are calculated using the following equation:

$$F_{\Delta H} \leq F_{\Delta H}^{RTP} * (1 + PF_{\Delta H} * (1 - P))$$

where: P = THERMAL POWER/RATED THERMAL POWER

a. $F_{\Delta H}^{RTP}$ = 1.66 for SPC fuel
b. $PF_{\Delta H}$ = 0.35 for SPC fuel

The $F_{\Delta H}^{RTP}$ value represents a reduction of 4.5% from the current COLR [Section 2.7 of Attachment 9 of Ref. (5)] limit for SPC fuel. $F_{\Delta H}$ as calculated above includes 4% measurement uncertainty (Tech. Spec., Section 3.2.3). The $F_{\Delta H}^{RTP}$ limit for previously burned Westinghouse fuel must be re-calculated prior to use.

2.3.2 The F_Q limits are calculated using the following equations:

$$F_Q(Z) \leq F_Q^{RTP} * K(Z)/P \text{ for } P > 0.5$$

$$F_Q(Z) \leq F_Q^{RTP} * K(Z)/0.5 \text{ for } P \leq 0.5$$

where: P = THERMAL POWER/RATED THERMAL POWER

a. F_Q^{RTP} = 2.41 for SPC fuel

b. K(Z) is specified in Figure 4 of Attachment 9 of Ref. (5).

F_Q^{RTP} limit represents a 4.5% decrease in the current limit found in the COLR (Section 2.6 of Attachment 9 of Ref. (5). F_Q as calculated above includes 3% engineering and manufacturing tolerance and 5% measurement uncertainty (Tech. Spec., Section 4.2.2.2.b). The F_Q^{RTP} limit for previously burned Westinghouse fuel must be re-calculated prior to use.

Table 2.4 Notes

2.4.1 The Minimum EOL MTC (most negative) value change from -45 pcm per °F to the Uprate Value of -50 pcm per °F is necessary due to Siemens calculation E-6924-N06-1 for the Neutronics Input to Safety. This calculation is not yet summarized in a suitable reference document, but a copy of the applicable portion of this proprietary internal Siemens document was supplied to CP&L for review in support of this UFAPPD change.

Table 2.6 Notes

2.6.1 Torque for hot conditions:

$$\text{Torque}_{hot} = \frac{\frac{6700\ \text{hp}}{\text{RCP}} \times 550 \frac{\text{ft-lb}_f}{\text{hp-sec}}}{1186 \frac{\text{rev}}{\text{min}} \times \frac{1\ \text{min}}{60\ \text{sec}} \times \frac{2\pi\ \text{radians}}{\text{rev}}} = \frac{29{,}670\ \text{ft-lb}_f}{\text{RCP}}$$

2.6.2 Torque for cold conditions:

$$\text{Torque}_{cold} = \frac{\frac{8870\ \text{hp}}{\text{RCP}} \times 550 \frac{\text{ft-lb}_f}{\text{hp-sec}}}{1182 \frac{\text{rev}}{\text{min}} \times \frac{1\ \text{min}}{60\ \text{sec}} \times \frac{2\pi\ \text{radians}}{\text{rev}}} = \frac{39{,}413\ \text{ft-lb}_f}{\text{RCP}}$$

2.6.3 Specific Speed:

$$N_s = \frac{\text{rpm} \times (\text{GPM})^{0.5}}{\text{Head}^{0.75}} = \frac{1182 \times (97{,}300)^{0.5}}{(294)^{0.75}} = 5193$$

Table 2.8 Notes

2.8.1 The +2% uncertainty associated with the Pressurizer safety valve setpoint includes a +1% tolerance (Tech. Spec. 3.4.2.2), and a +1% set pressure shift [Ref. (15)].

2.8.2 The close setpoint for the Pressurizer safety valves is based on 5% blowdown [Ref. (14)]. Assumes open pressure as 2475 psia, (1% tolerance) and then 5% blowdown.

2.8.3 Reference 61 documents the maximum flow as 420,000 lbm/hr. Siemens has previously used 455,000 lbm/hr for the evaluation of the Inadvertent Opening of the Pressurizer Safety Value. In Reference (91), CP&L specified a new, more conservative value for Cycle 10 analyses. The Uprate PPD was changed in Revision 2 to reflect this new, Cycle 10 and beyond value.

Table 2.9 Notes

2.9.1 This bounds the flow capacity of 255,600 lbm/hr at 2443 psia and 669°F reported in Ref. (36), Table 4.6.1-2b.

Table 2.10 Notes

2.10.1 The controller logic associated with the change in SG feedwater flow splits will be affected by the results of ESR 97-00807. When complete, that ESR should be compared against the UFAPPD for impact.

2.10.2 Ref. (99) is the Small Bore Piping, Replacement Steam Generator ESR 97-00808, which evaluates the acceptability of the existing piping that might be affected by the RSG effort. This ESR documents that the 1% continuous allowable tube sheet blowdown (approximately 43,000 lb/hr) reported to SPC in Revisions 0 and 1 of the UFAPPD needs to be reduced to the 28,000 lb/hr value shown to minimize the duty on the branch line control and isolation valves associated with the blowdown piping inside containment. While not formally approved by CP&L as of the date of issue of the Revision 2 of the UFAPPD, the reviews completed to date indicate that this portion of the ESR will not change prior to its final approval and issuance.

2.10.3 Upon receipt of an AFW Isolation signal, Blowdown flow will be isolated per Ref. (108). The reference is a typical reference for SG 1A, but the logic is the same for all three SG's.

Table 2.11 Notes

2.11.1 The valve flow capacity is dependent on ΔP. Siemens has been provided with a copy of Reference 91 which shows the valve performance in greater detail than this single scalar value.

Table 2.12 Notes

2.12.1 The ESF Response Time is defined as that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include Diesel Generator and Sequence Loading delays where applicable (Tech. Spec. Definition 1.13).

The total response time is composed of:

Closing Time	5 sec	Tech. Spec., Section 4.7.1.5
Signal Delay	2 sec	Ref. (5) Attach. 2 gives a total response time of ≤ 7 sec., leaving 2.0 sec for the signal delay

Table 2.13 Notes

2.13.1 From Ref. (16), the total relief capacity of all valves for each unit shall not be less than 12,816,196 lbm/hr at maximum accumulated pressure of 1290 psig. Any individual valve shall have a relief capacity no greater than 970,000 lbm/hr at maximum accumulated pressure of 1278 psig.

Table 2.14 Notes

2.14.1 The ESF Response Time is defined as that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressure reach their required values, etc.). Times shall include Diesel Generator and Sequence Loading delays where applicable (Tech. Spec. Definition 1.13).

The total response times are composed of:

Closing Times	8 sec for MFW control and isolation valves	Ref. (18)
	10 sec for MFW bypass control and isolation valves	Idem.
Signal Delay	2 sec	Ref. (5), Attach. 2 gives a total response time of ≤10.0 sec. for Main Feedwater Isolation valves only, leaving 2.0 sec for the signal delay

Table 2.15 Notes

2.15.1 The ESF Response Time is defined as that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressure reach their required values, etc.). Times shall include Diesel Generator and Sequence Loading delays where applicable (Tech. Spec. Definition 1.13).

2.15.2 The values provided for SGR/PUR represent conservative values (lower minimum AFW and higher maximum AFW) relative to values previously given in DBD-314. The Maximum AFW flow is discussed in the Calculations section of the Ref. (75) controlling document for the UFAPPD.

2.15.3 The two pump minimum AFW flow to two SG's is approximated from the range of values given on page 7 of 30 in Ref. 98. It is noted that this Ref. 98 was superceded, in part, by Ref. 43, since Ref. 98 did not include possible pump degradation. The Ref. 43 calculation did not, however re-calculate all the cases in Ref. 98, so Ref. 98 remains the only available reference for two motor driven pumps feeding two generators. The pump degradation (4%) would have the effect of lowering the Ref. 98 values about 10 gpm per pump (based on comparisons to similar cases in both calculations), so the observed values of approximately 850 gpm are reduced to 830 gpm. Ref. 98 was also examined for possible reductions in flow associated with pipe resistance and pump combination effects. Two motor driven pumps to 2 SG's was about 97% of the value obtained from using two times the single pump average. Twice the single pump Analysis Value of 390 gpm times the 97% flow restriction factor yields 756.6 gpm, which was rounded to **760 gpm** for the Analysis Value for 2 MD pumps into two SG's.

2.15.4 The three pump minimum flow to one SG is approximated from the range of values given on page 14 of 30 of Ref. 98. The median value of about 1450gpm is reduced by 10 gpm per motor driven AFW pump for reasons cited in Note 2.15.3, yielding to value of 1430 gpm. Ref.23 was reviewed for potential runout concerns for the Turbine driven AFW pump (which might have raised the minimum three pump flow substantially). It was shown in that calculation, however, that sufficient reasons exist to confirm that the TDAFW pump will not runout uncontrollably during the MFWLB analysis (for which this minimum three pump flow is likely to be used). Additional input by the System Engineer indicated that this value should be reduced further, based on expected increases in flow resistance over the analysis of record. Accordingly, the 1430 gpm value above is conservatively reduced further to **1400 gpm** for specification of this input to SPC.

Table 2.16 Notes

2.16.1 Pressurizer pressure uncertainty analysis value consists of a ±38 psi random component in combination with a -12 psi bias. The -12 psi bias refers to the directional tendency of the Barton pressure transmitter to give an artificially high indication of pressure. Ref. (100) confirms that this analysis value envelops uprated operating conditions.

2.16.2 While indicated temperature may be manually increased up to 4°F above the normal program or target value for this particular step as part of power ascension in Mode 1, the actual or ultimate range is increased to accommodate measurement uncertainty.

2.16.3 WCAP-12340 (Ref. (88)) forms the basis for current HNP Improved Thermal Design procedure (ITDP) uncertainties, and this reference will be updated by Westinghouse as part of the SGR/PUR project. Ref.'s (100) through (103) document ITDP uncertainties for pressurizer pressure, Tavg, RCS loop flow, and reactor power which are applicable for uprate power conditions.

2.16.4 Pressurizer Water Level Control uncertainty analysis value was developed in Ref. (104) specifically for the Siemens uprate reload analyses. This value should be considered in conjunction with the Pressurizer Water Level High Trip Setpoint analysis value shown in Table 2.2

Table 2.17 Notes

2.17.1 Ref. (76) provides the value calculated by Westinghouse for the initial licensing calculations. Ref. (77) is a recent CP&L calculation of the reactor coolant system volumes. When the Ref. (77) mid-loop volume is combined with the Westinghouse RHR volume, a total volume of 3760.6 ft^3 is produced. The use of the Westinghouse/FSAR value is therefore justified. The affect of the SGR will be to increase the mixing volume for higher modes of operation. For the purpose of the Boron Dilution analyzes, the additional volume will be ignored for conservatism.

2.17.2 This is the high letdown flow alarm setpoint of 130 gpm [sheet 74 of 98 of Ref. (1)] increased by approximately 1% as an allowance of instrument accuracy. This is also consistent with a high charging flow alarm setpoint of 115 gpm [sheet 74 of 98 of Ref. (1)] in combination with a nominal Reactor Coolant Pump seal injection of 5 gpm into each of the three loops. While neither the nominal RCP Seal Injection flow rate of 8 to 13 gpm nor the 5/3 flow split is applicable while at RCS reduced inventory, a positive pressure is usually applied to the RCP seals [see the notes in Ref. (56) and in Ref. (57)]. 5 gpm is merely a reasonable allowance.

Table 2.18 Notes

2.18.1 The lead and lag time constants for the Low Steam Line Pressure setpoint are $\tau_1 \leq$ 45 seconds and $\tau_2 \geq$ 5.5 seconds, respectively (Ref. 5 with 10% conservatism).

Table 2.19 Notes

2.19.1 The ESF Response Time is defined as that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressure reach their required values, etc.). Times shall include Diesel Generator and Sequence Loading delays where applicable (Tech. Spec. 1.13)

2.19.2 In comparing footnotes 1 vs. 4 in Ref. (5), this value is conservative by 29 - 22 = 7 seconds for HHSI (alone, without waiting for RHR/LHSI).

2.19.3 A flow of 60 gpm is maintained through the charging line to simulate flow lost through Reactor Coolant Pump Seal Injection. Seal Injection system resistance is mandated by T.S.3.4.6.2.e. Additional information regarding seal injection is provided in Ref. (19) and Ref. (20).

2.19.4 For the Power Uprate effort, CP&L is electing to return to the more conservative specification of 29 seconds for these parameters, consistent with the specification for the original Cycle 6 Siemens analysis inputs given in Ref. (83).

2.19.5 For the analysis purposes, the boration/dilution/reactivity control aspects of the event progression should assume that the water inventory in the ECCS piping, including the volume of the BIT, is at 0 ppm boron.

Table 2.20 Notes

2.20.1 The ESF Response Time is defined as that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressure reach their required values, etc.). Times shall include Diesel Generator and Sequence Loading delays where applicable (Tech. Spec. Definition 1.13).

2.20.2 The use of 29 seconds for the delay time is conservative relative to the current plant allowable value of 27 seconds. See note 2.19.4 for additional explanation.

Table 2.21 Notes

2.21.1 Correlation between level and volume are found in Ref. (9), Curve Number D-34, Revision 1.

Table 2.22 Notes

2.22.1 Previous estimate of Time to reach full spray flowrate used:
As a conservative simplification, this value is nothing more than time required to fill the spray piping with water. The other time components presented in Attachment 2 to Ref. (5) are taken as zero:

- time to start Emergency Diesel Generator and bring it up to speed (only applicable if offsite power is lost)
- time to sequence Containment Spray pump electrical load onto emergency bus
- time to bring pump up to speed

2.22.2 Values chosen for flow and response time are conservative to previous plant inputs.

Table 2.23 Notes

2.23.1 Use of 0 delay time conservative relative to actual plant operation.

Table 2.26 Notes

2.26.1 The rod control system compares actual T_{avg} with the programmed temperature (T_{ref}), and the resultant error (if any) determines rod speed and direction. For T_{avg} equal to 588.8°F, T_{ref} increases linearly with reactor power, from 557°F at 0% power to 588.8°F at 100% power at a gain of 0.318°F/% power load. For T_{avg} equal to 580.8°F, T_{ref} increases linearly with reactor power, from 557°F at 0% power to 580.8°F at 100% power at a gain of 0.238°F/% power load.

2.26.2 The Automatic Rod Control system will be modeled as in either Automatic or Manual (or analyzed both ways), depending on the particular determination of what is conservative or bounding in the Chapter15 Disposition of Events (DOE). Reference10 provides the DOE for the original transition between Westinghouse and Siemens. Table 15.1.3-1, for instance, shows that the ARC system in Automatic is the modeling assumption for events involving increases in steam flow. A specific activity is scheduled for Siemens to perform a similar DOE for the SGR/PUR analyses, in which Siemens will document the modeling of the ARC system (automatic or manual) for each Chapter 15 event. Should Siemens determine that insufficient margin exists to maintain ARW, SPC should notify CP&L immediately to allow this specification to be reviewed and changed, as necessary.

Chapter 15 Event DNBR Tables & Plots

List of Items in this Attachment

15.1.2 DNBR plots from proprietary calc for this event (Increase in Feedwater Flow)

15.2.3 DNBR plots (ANF-RELAP, not XCOBRA-IIIC) from proprietary calc for this event (Turbine Trip).

15.3.2 DNBR plot (ANF-RELAP, not XCOBRA-IIIC) from proprietary calc (Complete Loss of Reactor Coolant Flow)

15.3.3 DNBR plots (both ANF-RELAP and XCOBRA-IIIC) from proprietary calc (Locked Rotor).

15.4.1 DNBR Table (from XCOBRA-IIIC) as well as ANF-RELAP T/H condition plots from proprietary calc (Uncontrolled RCCA Withdrawal at Low Power)

15.4.2 DNBR Plot (deterministic, not statistical) from the Engineering Report EMF-2377, Rev 1 for this event (Uncontrolled RCCA Withdrawal at Power) as well as individual plots 10.1 through 10.4 from proprietary calc.

15.4.8 Tables (extracted from proprietary calc) of XCOBRA-IIIC calculated DNBR's around the times of the limiting MDNBR for various conditions(Control Rod Ejection)

15.5.1 A pre-Amendment 44 copy of the DNBR plot in FSAR Figure 15.5.1-2 (Inadvertent Operation of ECCS During Power Operation) for SGR-Only submittal. Power Uprate Licensing Report Section 6.2.26, plots 6.2.26-6 and 6.2.26-13 provide similar information for Power Uprate (not retransmitted in this RAI package).

15.6.1 Table 7.8 from proprietary calculation shows the XCOBRA-IIIC DNBR trend in the region of minimum DNBR occurrence. (Inadvertent Opening of Pressurizer Safety or Relief Valve).

Chapter 15 Event DNBR Tables & Plots

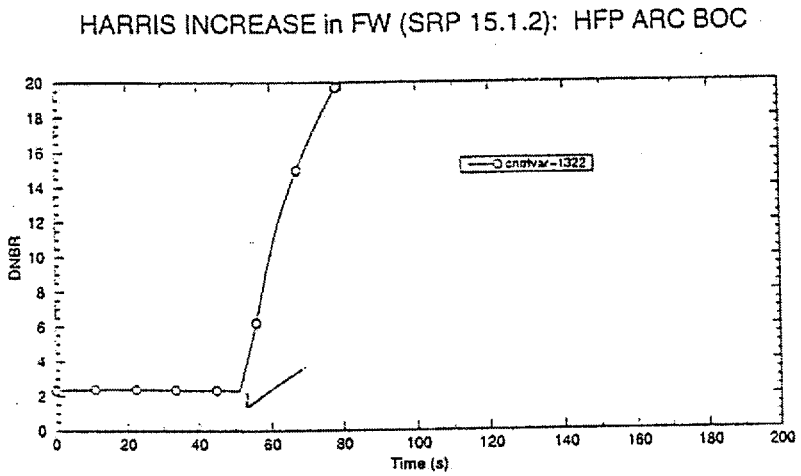


Figure 6.6: DNBR as Calculated by ANF-RELAP (HFP ARC BOC)

The DNBR calculated by ANF-RELAP decreased very little during this calculation. It reached a minimum of 2.2882 at 50.6 seconds.

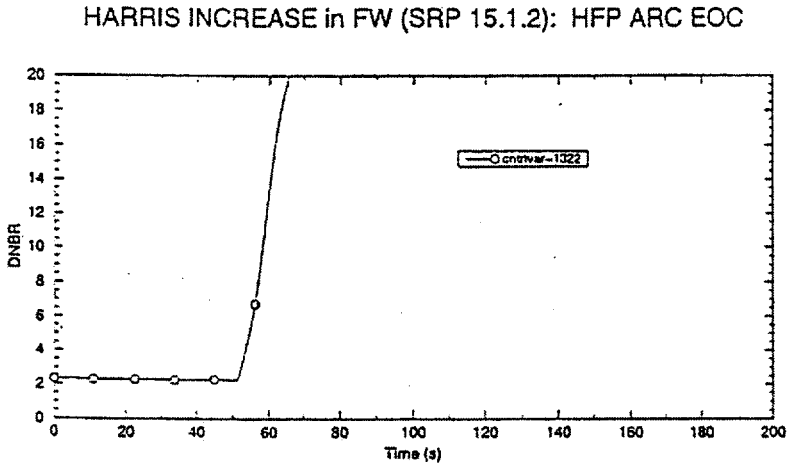


Figure 6.13: DNBR as Calculated by ANF-RELAP (HFP ARC EOC)

The DNBR calculated by ANF-RELAP decreased very little during this calculation. It reached a minimum of 2.2394 at 49.4 seconds.

Chapter 15 Event DNBR Tables & Plots

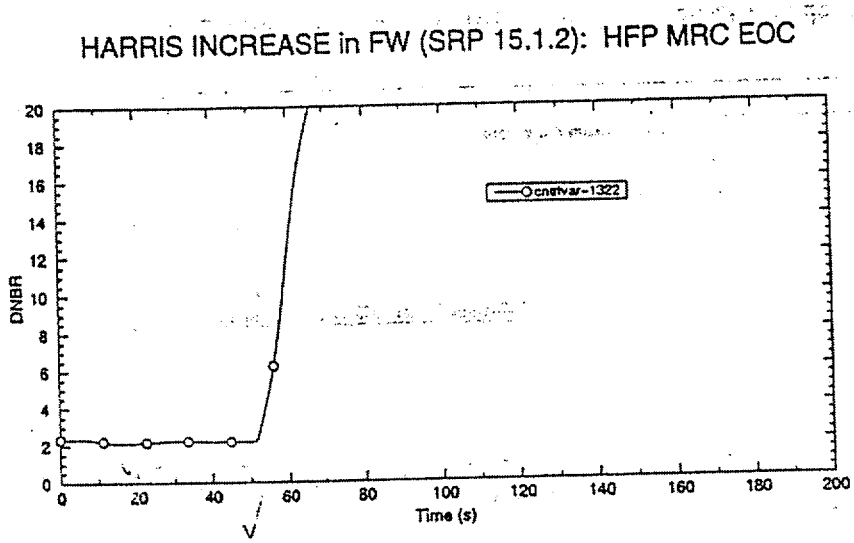


Figure 6.20: DNBR as Calculated by ANF-RELAP (HFP MRC EOC)

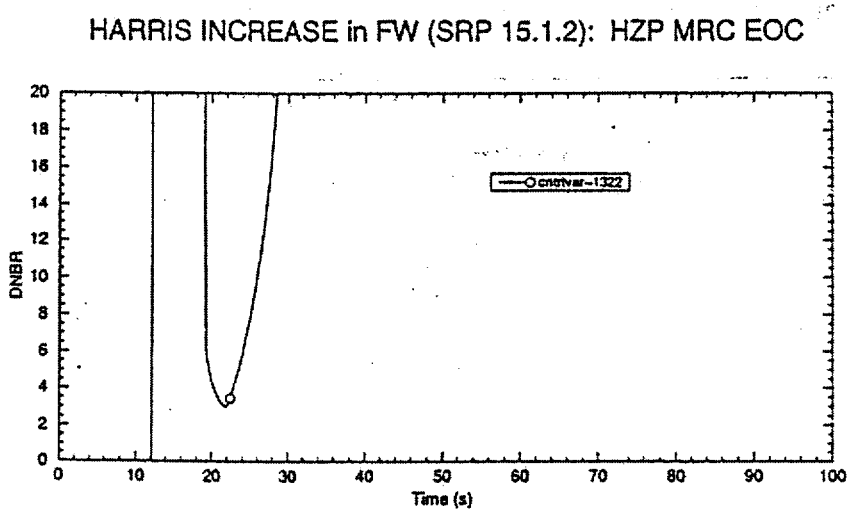


Figure 6.26: DNBR as Calculated by ANF-RELAP (HZP MRC EOC)

Chapter 15 Event DNBR Tables & Plots

15.2.3 DNBR plots (ANF-RELAP, not XCOBRA-IIIC) from proprietary calc for this event (Turbine Trip).

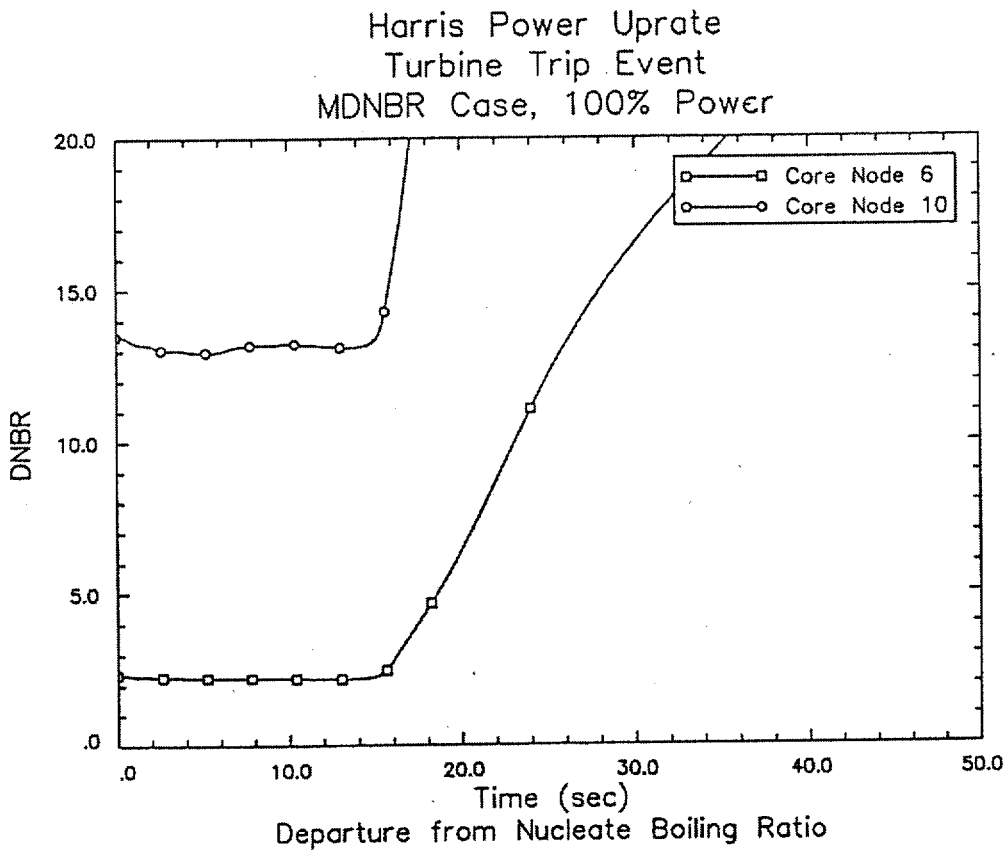


Figure 6.5.1.4 MDNBR Transient Calculation: Tong DNB Ratio (100% Power)

Chapter 15 Event DNBR Tables & Plots

15.2.3 DNBR plots (ANF-RELAP, not XCOBRA-IIIC) from proprietary calc for this event (Turbine Trip).

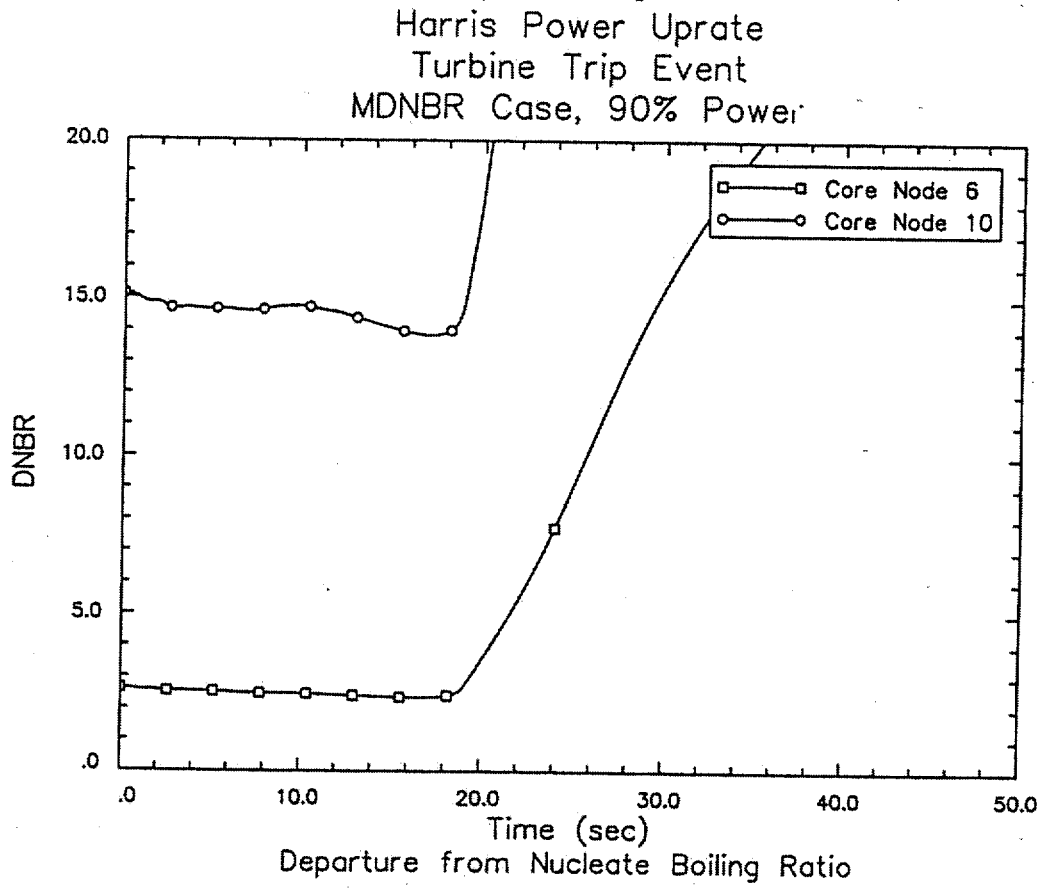


Figure 6.5.2.4 MDNBR Transient Calculation: Tong DNB Ratio (90% Power)

Chapter 15 Event DNBR Tables & Plots

15.2.3 DNBR plots (ANF-RELAP, not XCOBRA-IIIC) from proprietary calc for this event (Turbine Trip).

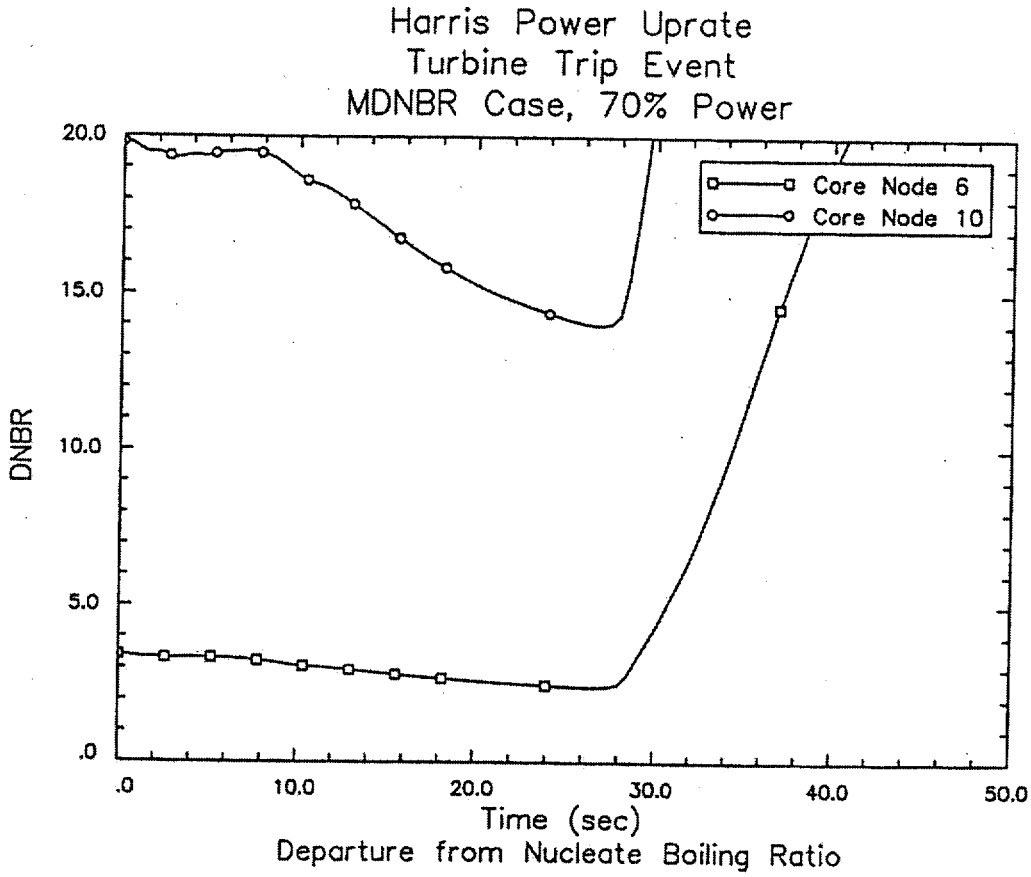
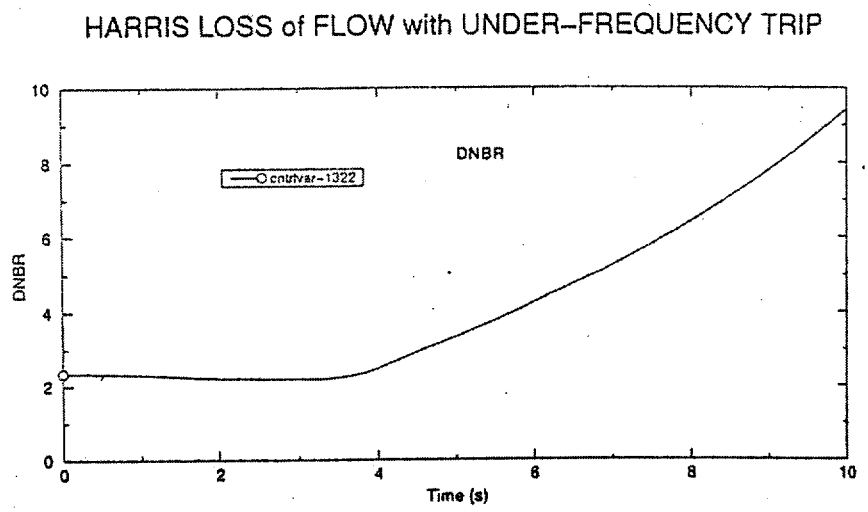


Figure 6.5.3.4 MDNBR Transient Calculation: Tong DNB Ratio (70% Power)

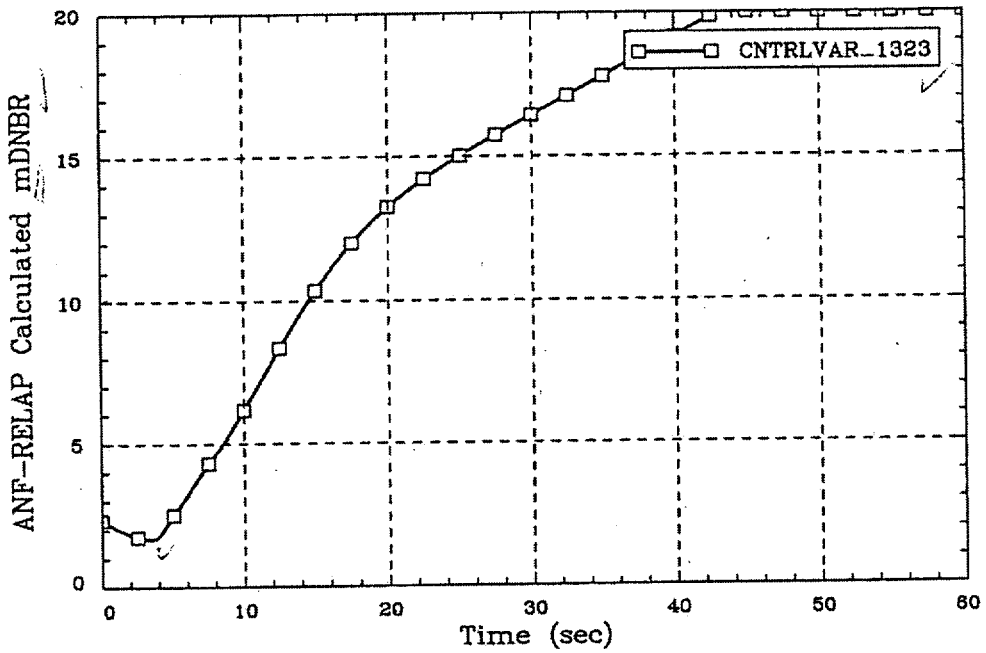
Chapter 15 Event DNBR Tables & Plots

15.3.2 DNBR plot (ANF-RELAP, not XCOBRA-IIIC) from proprietary calc (Complete Loss of Reactor Coolant Flow)



Chapter 15 Event DNBR Tables & Plots

15.3.3 DNBR plots (both ANF-RELAP and XCOBRA-IIIC) from proprietary calc (Locked Rotor)



Seized Rotor Minimum DNBR

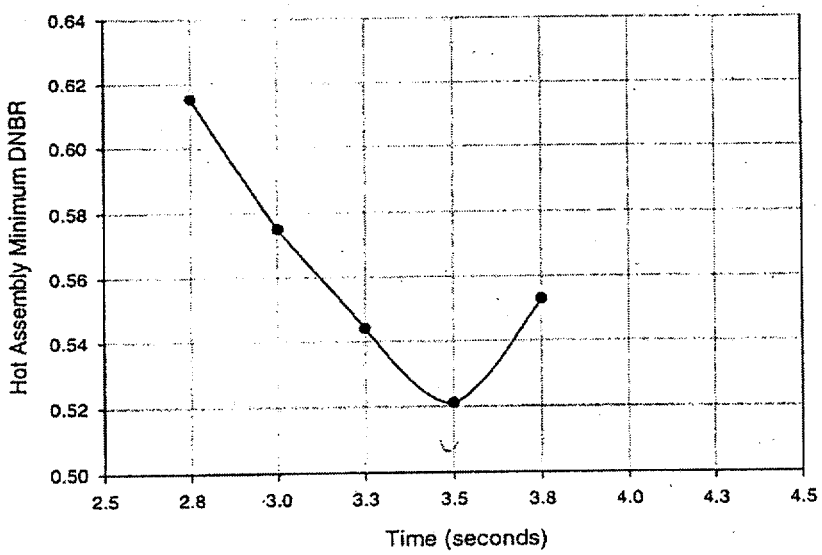


Figure 6.6-1 Seized Rotor MDNBR

Chapter 15 Event DNBR Tables & Plots

15.4.1 DNBR Table (from XCOBRA-IIIC) as well as ANF-RELAP T/H condition plots from proprietary calc (Uncontrolled RCCA Withdrawal at Low Power)

HARRIS PUR BANK WITHDRAWAL FROM SUBCRITICAL

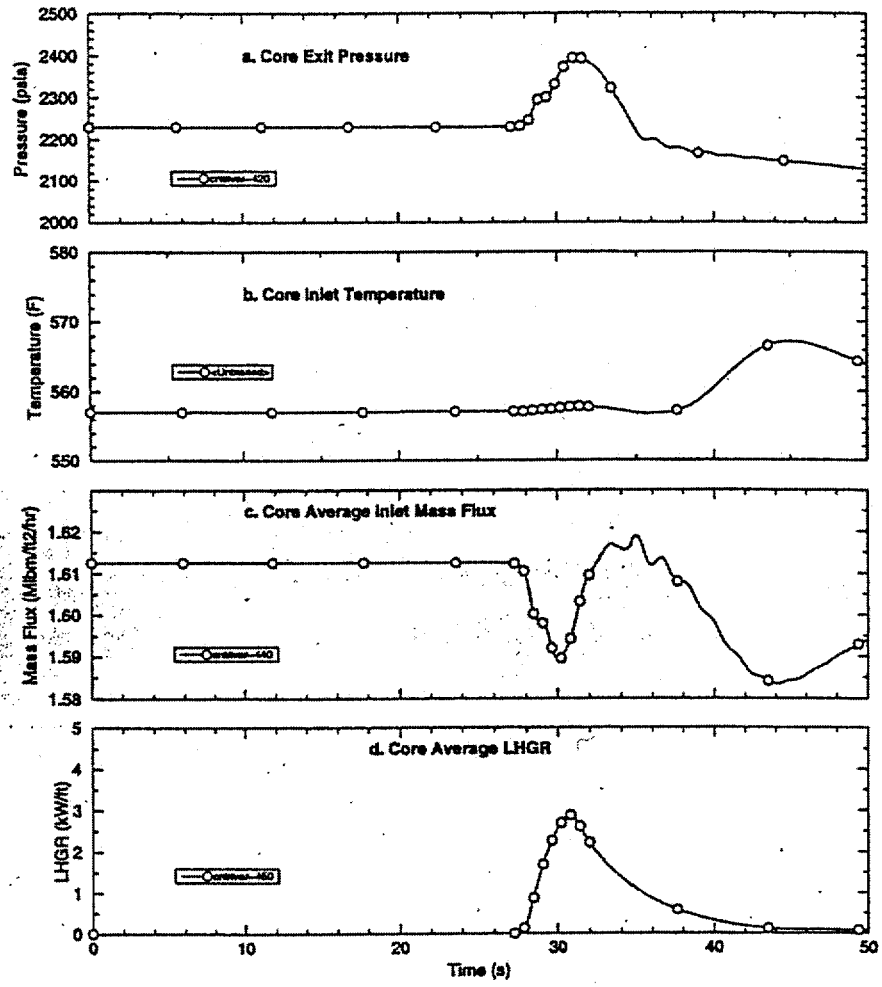


Figure 6.9: ANF-RELAP Parameters for XCOBRA-IIIC

Chapter 15 Event DNBR Tables & Plots

15.4.1 DNBR Table (from XCOBRA-IIIC) as well as ANF-RELAP T/H condition plots from proprietary calc (Uncontrolled RCCA Withdrawal at Low Power)

The DNBR results from each XCOBRA-IIIC calculation are presented in Table 6.9. The overall minimum DNBR was 1.758, which is greater than the 95/95 safety limit for the HTP correlation (1.164, which includes the 2% mixed core penalty).

Table 6.9: XCOBRA-IIIC Calculation Results

Case	Time (sec)	DNBR	Elevation
PDIL	30.70	2.389	130"
	30.74	2.387	
	30.75	2.387	
	30.76	2.388	
	30.77	2.388	
	30.78	2.390	
	30.80	2.403	
PDIL-056	30.70	1.863	14.80"
	30.74	1.860	
	30.75	1.860	
	30.76	1.859	
	30.77	1.859	
	30.78	1.859	
	30.80	1.861	
PDIL-112	30.70	2.505	24.40"
	30.74	2.500	
	30.75	2.500	
	30.76	2.499	
	30.77	2.499	
	30.78	2.499	
	30.80	2.502	
PDIL-168	30.70	3.137	125.2"
	30.74	3.134	
	30.75	3.134	
	30.76	3.134	
	30.77	3.135	
	30.78	3.137	
	30.80	3.150	
ARO	30.70	1.760	134.8"
	30.74	1.758	
	30.75	1.759	
	30.76	1.759	
	30.77	1.759	
	30.78	1.760	
	30.80	1.769	

Chapter 15 Event DNBR Tables & Plots

DNBR Plot (deterministic, not statistical) from the Engineering Report EMF-2377, Rev 1 for this event (Uncontrolled RCCA Withdrawal at Power) as well as individual plots 10.1 through 10.4 from proprietary calc

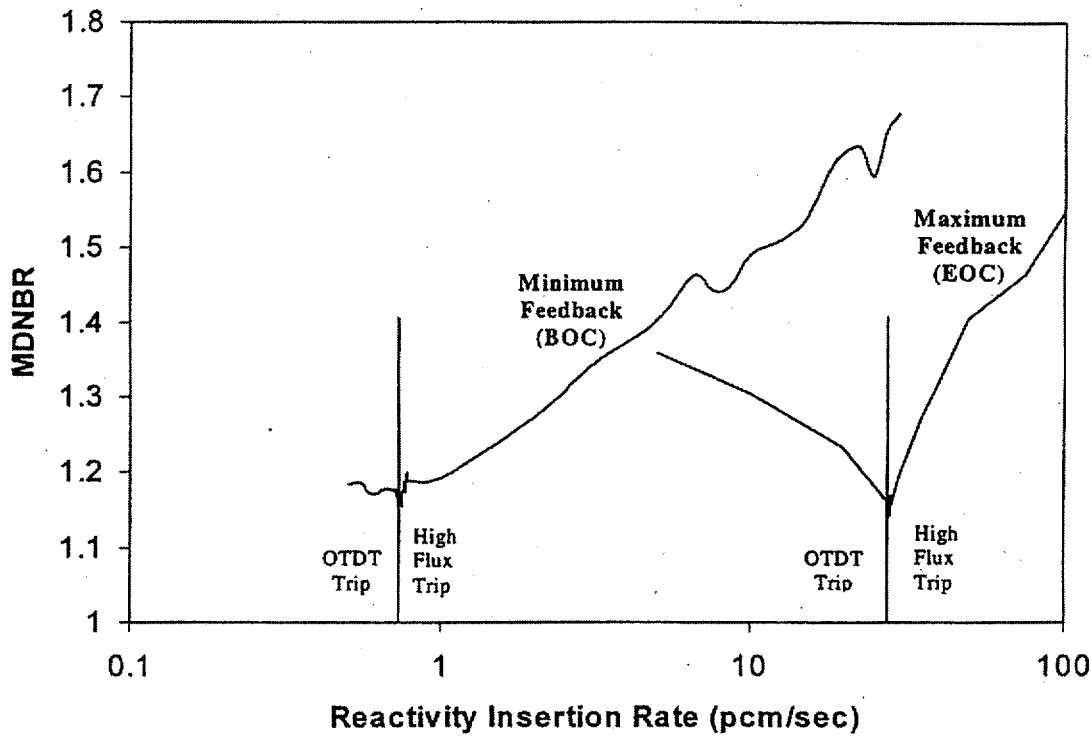


Figure 6.2.19-1 Deterministic MDNBR versus Reactivity Insertion Rate for Full Power Uncontrolled RCCA Bank Withdrawal

Chapter 15 Event DNBR Tables & Plots

DNBR Plot (deterministic, not statistical) from the Engineering Report EMF-2377, Rev 1 for this event (Uncontrolled RCCA Withdrawal at Power) as well as individual plots 10.1 through 10.4 from proprietary calc

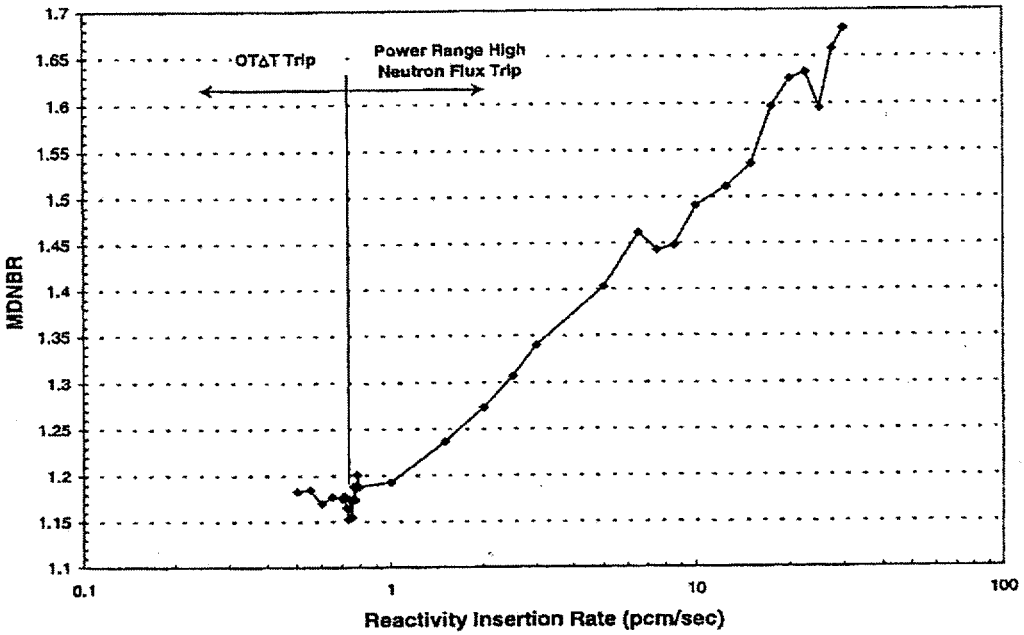


Figure 10.1 MDNBR vs. Reactivity Insertion Rate, 102% BOC Limiting Bank Withdrawal

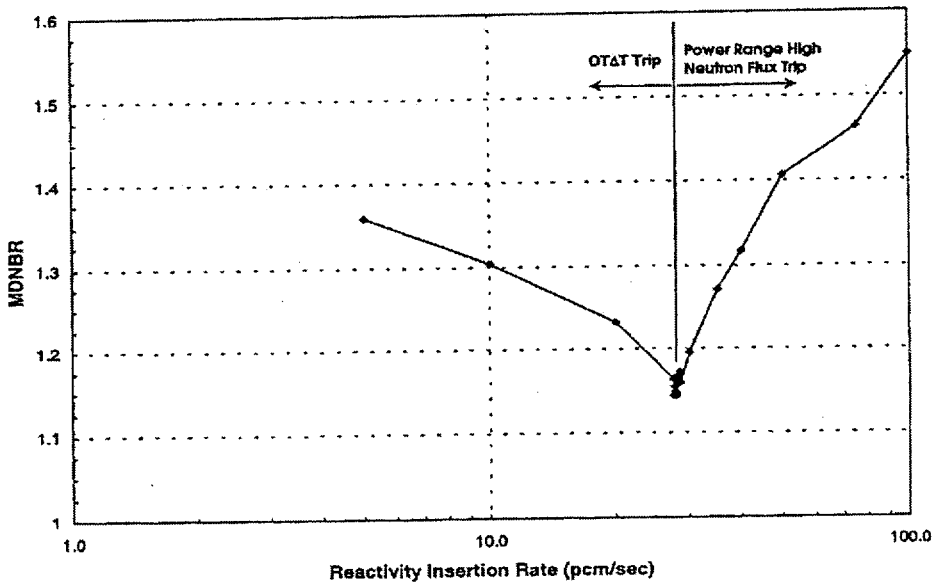


Figure 10.2 MDNBR vs. Reactivity Insertion Rate, 102% EOC Limiting Bank Withdrawal

Chapter 15 Event DNBR Tables & Plots

DNBR Plot (deterministic, not statistical) from the Engineering Report EMF-2377, Rev 1 for this event (Uncontrolled RCCA Withdrawal at Power) as well as individual plots 10.1 through 10.4 from proprietary calc

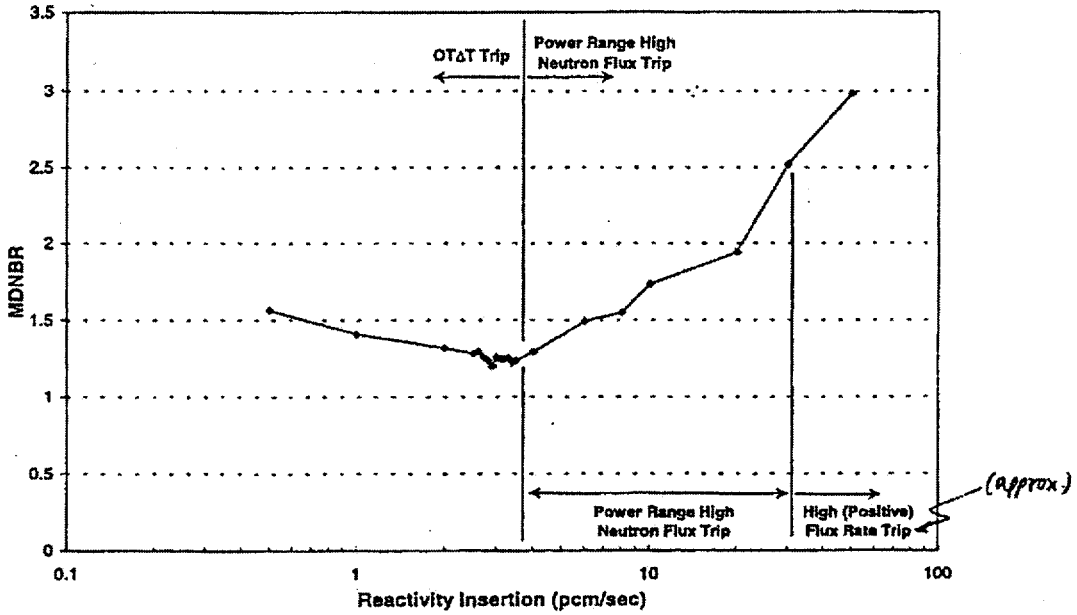


Figure 10.3 MDNBR Response vs. Reactivity Insertion Rate for 60% BOC BWaP

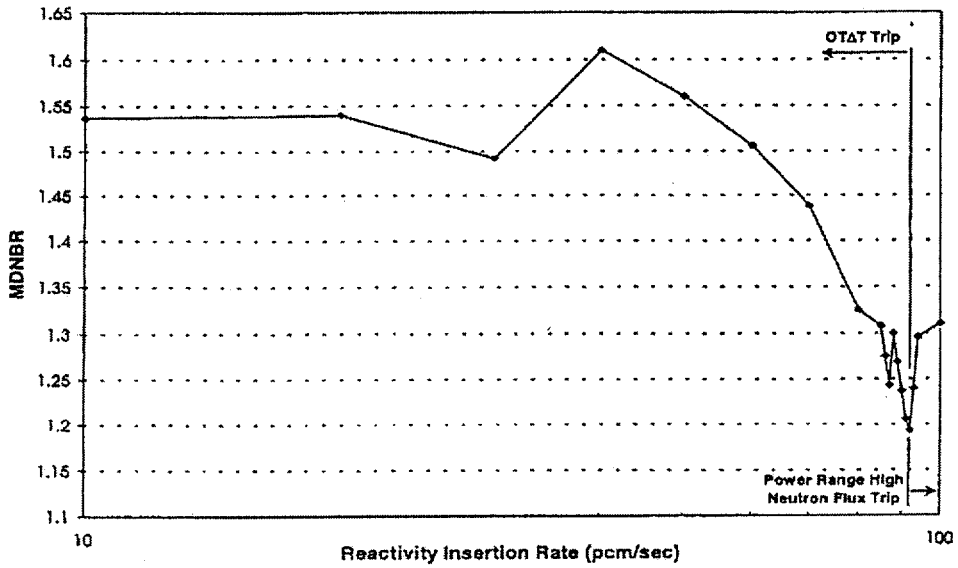


Figure 10.4 MDNBR Response vs. Reactivity Insertion Rate for 60% EOC BWaP

Chapter 15 Event DNBR Tables & Plots

15.4.8 Tables (extracted from proprietary calc) of XCOBRA-IIIC calculated DNBR's around the times of the limiting MDNBR for various conditions(Control Rod Ejection)

Table 9.1 HFP BOC MDNBR Results

Time, sec	MDNBR
11.85	1.222
11.90	1.220
11.95	1.218
12.00	1.216
12.05	1.216
12.10	1.217
12.15	1.219

Table 9.2 HFP EOC MDNBR Results

Time, sec	MDNBR
7.50	1.632
8.00	1.631
8.50	1.630
9.00	1.630
9.50	1.630
10.00	1.630
10.50	1.631
11.00	1.632

Table 9.3 HZP BOC MDNBR Results

Time, sec	MDNBR
12.45	1.874
12.50	1.871
12.55	1.870
12.60	1.871
12.65	1.875

Table 9.4 HZP EOC MDNBR Results

Time, sec	MDNBR
4.25	1.318
4.30	1.315
4.35	1.314
4.40	1.315
4.45	1.318

Chapter 15 Event DNBR Tables & Plots

15.5.1 A pre-Amendment 44 copy of the DNBR plot in FSAR Figure 15.5.1-2 (Inadvertent Operation of ECCS During Power Operation) for SGR-Only

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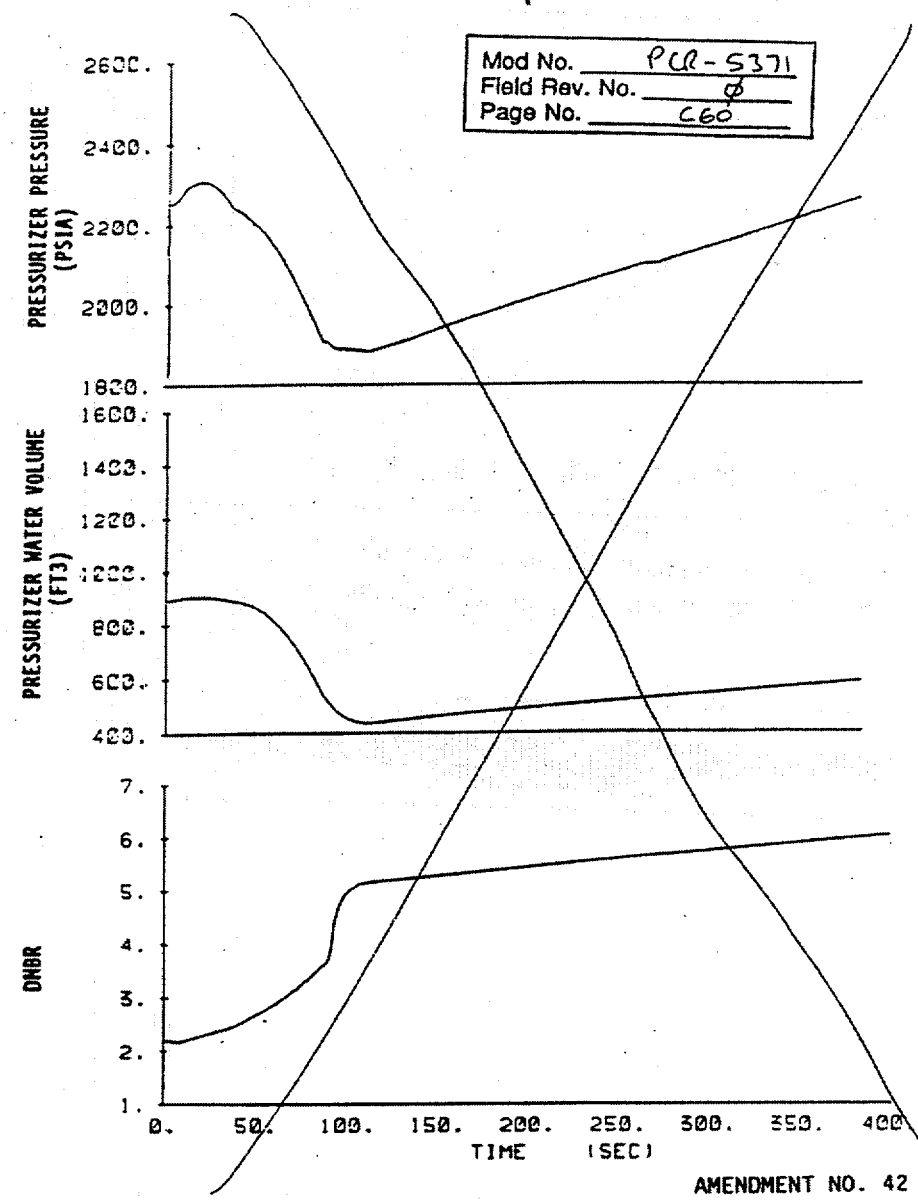
Replace these figures



SHEARON HARRIS NUCLEAR POWER PLANT Carolina Power & Light Company FINAL SAFETY ANALYSIS REPORT	INADVERTENT OPERATION OF THE EMERGENCY CORE COOLING SYSTEM AT POWER - PRESSURIZER WATER VOLUME, PRESSURIZER PRESSURE, AND DNBR VERSUS TIME	FIGURE 15.5.1-2

Chapter 15 Event DNBR Tables & Plots

15.6.1 Table 7.8 from proprietary calculation shows the XCOBRA-IIIC DNBR trend in the region of minimum DNBR occurrence. (Inadvertent Opening of Pressurizer Safety or Relief Valve)

Table 7.8 MDNBR Results

Time (sec)	Case Identifier	MDNBR
24.3	A	1.255
24.4	B	1.253
24.5	C	1.251
24.6	D	1.249
24.7	E	**1.248**
24.8	F	1.251
24.9	G	1.259
25.0	H	1.269
25.1	I	1.276
25.2	J	1.279