ANF-88-108

ADVANCED NUCLEAR FUELS CORPORATION

PALISADES CYCLE 8: DISPOSITION AND ANALYSIS OF STANDARD REVIEW PLAN CHAPTER 15 EVENTS

AUGUST 1988

ADVANCED NUCLEAR FUELS CORPORATION

ANF-88-108

Issue Date: 8/2/88

PALISADES CYCLE 8: DISPOSITION AND ANALYSIS OF STANDARD

REVIEW PLAN CHAPTER 15 EVENTS

Prepared by: TR Lindquist

T. R. Lindquist, Engineer
PWR Safety Analysis
Licensing & Safety Engineering
Fuel Engineering & Technical Services

August 1988

gf

CUSTOMER DISCLAIMER

IMPORTANT NOTICE REGARDING CONTENTS AND USE OF THIS DOCUMENT

PLEASE READ CAREFULLY

Advanced Nuclear Fuels Corporation's warranties and representations concerning the subject matter of this document are those set forth in the Agreement between Advanced Nuclear Fuels Corporation and the Customer pursuant to which this document is issued. Accordingly, except as otherwise expressly provided in such Agreement, neither Advanced Nuclear Fuels Corporation nor any person acting on its behalf makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this document, or that the use of any information, apparatus, method or process disclosed in this document will not infringe privately owned rights; or assumes any liabilities with respect to the use of any information, apparatus, method or process disclosed in this document.

The information contained herein is for the sole use of Customer.

In order to avoid impairment of rights of Advanced Nuclear Fuels Corporation in patents or inventions which may be included in the information contained in this document, the recipient, by its acceptance of this document, agrees not to publish or make public use (in the patent use of the term) of such information until so authorized in writing by Advanced Nuclear Fuels Corporation or until after six (6) months following termination or expiration of the aforesaid Agreement and any extension thereof, unless otherwise expressly provided in the Agreement. No rights or licenses in or to any patents are implied by the furnishing of this document.

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

Section | Page

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION

This report documents the results of a Standard Review Plan (SRP)[1] Chapter 15 disposition of events and analysis performed in support of Palisades Cycle 8 operation. A modified reactor protection system (RPS), including a variable-overpower trip and an improved thermal margin/low pressure (TM/LP) trip with axial monitoring, will be installed prior to Cycle 8 operation and is supported by the the analyses reported in References 2 and 3. Additional changes that will be implemented into Palisades Cycle 8 are:

(1) An increase in Technical Specification radial peaking factor limits to accommodate a low radial leakage loading pattern for the purpose of reducing vessel fluence. The radial peaking factors will be increased by 3.5%.

(2) Insertion of four ANF lead assemblies with high thermal performance spacers.

(3) Reinsertion of sixteen previously burnt assemblies at locations along the core periphery to reduce neutron fluence at critical vessel welds. Each of these assemblies will be reconstituted with 56 stainless steel rods replacing the fuel rods along the four outer rows on one side of the assembly.

The Chapter 15 events were disposed and analyzed in accordance with Advanced Nuclear Fuels Corporation methodology.[9] The LOCA/ECCS analyses in support of Palisades Cycle 8 are documented in Reference 10.

Section 2.0 presents a summary of the results and review of SRP Chapter 15 events. Section 3.0 presents the conditions employed in the event analyses and the results of these event analyses. Events are numbered in accordance with the SRP to facilitate review. A tabular list of the disposition of

Chapter 15 events and analysis of record for Palisades, with a cross reference between SRP event numbers and the Palisades Updated FSAR[8], is included. Section 4.0 presents the results of a thermal-hydraulic compatibility analysis for the four lead assemblies and the sixteen stainless steel shielding assemblies.

2.0 SUMMARY AND CONCLUSIONS

A summary Disposition of Events for the changes proposed for Palisades Cycle 8 is given in Table 2-1. This table lists each SRP Chapter 15 event, indicates whether that event is reanalyzed for this submittal, and provides a reference to the bounding event or analysis of record for events not reanalyzed.

The changes listed in Section 1.0 for Cycle 8 do not alter the plant system response to a transient event relative to the analysis supporting modified RPS operation.[3] The increase in radial peaking limits will, however, impact minimum Departure from Nucleate Boiling Ratio (DNBR). Therefore, the analysis for the events disposed to be reanalyzed for Cycle 8 will consist of an evaluation of the minimum DNBR and DNBR related consequences (e.g., fuel failure) using the appropriate transient conditions in Reference 3. The results of Anticipated Operational Occurrences and Postulated Accidents reanalyzed for this submittal are listed in Table 2-2. Acceptance criteria are met for each event.

The results reported herein confirm that event acceptance criteria are met for Cycle 8 operation. These results support operation with up to 29.3% average steam generator tube plugging at a rated thermal power of 2530 MWt, which is consistent with the Reference 3 analysis.

Table 2-1 Disposition of Events Summary for Palisades

Event Classification	SRP Event Designation	Name	Disposition	Bounding Event or Reference	Updated FSAR Designation
15.1	INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM				
	15.1.1	Decrease in Feedwater Temperature	Bounded	15.1.3	14.9.4
	15.1.2	Increase in Feedwater Flow			
		1) Power	Bounded	15.1.3	14.9.6
		2) Startup	Bounded	15.1.3	14.9.5
	15.1.3	Increase in Steam Flow	Analyze		14.10
	15.1.4	Inadvertent Opening of a Steam Generator Relief of Safety Valve			
		1) Power	Bounded	15.1.3	
		2) Scram Shutdown Margin	Bounded	15.1.3	
	15.1.5	Steam System Piping Failures Inside and Outside of Containment	Bounded	Ref.11,12&13	14.14
15.2	DECREASE IN HEAT REMOVAL BY THE SECONDARY STEAM				
	15.2.1	Loss of External Load	Analyze		14.12
	15.2.2	Turbine Trip	Bounded	15.2.1	
	15.2.3	Loss of Condenser Vacuum	Bounded	15.2.1	
	15.2.4	Closure of the Main Steam Isolation Valves (MSIVs)	Bounded	15.2.1	
	15.2.5	Steam Pressure Regulator Failure	Not applicable; BWR Event		

Table 2-1 Disposition of Events Summary for Palisades (Cont.)

Event Classification	SRP Event Designation	Name	Disposition	Bounding Event or Reference	Updated FSAR Designation
	15.2.6	Loss of Nonemergency A.C. Power to the Station Auxiliaries	Short term bounded Long term bounded	15.3.1 15.2.7	
	15.2.7	Loss of Normal Feedwater Flow	Bounded	Ref. 3	14.13
	15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment	Cooldown Bounded Heatup Bounded	15.1.5 15.2.7	
15.3	DECREASE IN REACTOR COOLANT SYSTEM FLOW				
	15.3.1	Loss of Forced Reactor Coolant Flow	Analyze		14.7
	15.3.2	Flow Controller Malfunction	Not Applicable		14.7
	15.3.3	Reactor Coolant Pump Rotor Seizure	Analyze		14.7
	15.3.4	Reactor Coolant Pump Shaft Break	Bounded	15.3.3	14.7
15.4	REACTIVITY AND POWER DISTRIBUTION ANOMALIES				
	15.4.1	Uncontrolled Control Rod Bank Withdrawal from a Subcritical or Low Power Condition	Analyze		14.2.2.2
	15.4.2	Uncontrolled Control Rod Bank Withdrawal at Power Operation Conditions	Analyze		14.2.2.3
	15.4.3	Control Rod Misoperation			
		1) Dropped Control Bank/Rod	Analyze		14.4
		2) Dropped Part-Length Control Rod	Bounded	15.4.3(1)	14.6

Table 2-1 Disposition of Events Summary for Palisades (Cont.)

Event Classification	SRP Event Designation	Name	Disposition	Bounding Event or Reference	Updated FSAR Designation
		3) Malpositioning of the Part-Length Control Group	Not Applicable		14.6
		4) Statically Misaligned Control Rod/Bank	Analyze		
		5) Single Control Rod Withdrawal	Analyze	Ref. 8	14.2.2.4
		6) Core Barrel Failure	Analyze		14.5
	15.4.4	Startup of an Inactive Loop	Analyze		14.8
	15.4.5	Flow Controller Malfunction	Not applicable; No Flow Controller		
	15.4.6	CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant			
		1) Rated and Power Operation Conditions	Analyze		14.3
		2) Reactor Critical, Hot Standby and Hot Shutdown	Analyze		14.3
		3) Refueling Shutdown Condition, Cold Shutdown Condition and Refueling Operation	Analyze		14.3
	15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	Administrative Procedures Preclude this Event		

Table 2-1 Disposition of Events Summary for Palisades (Cont.)

Event Classification	SRP Event Designation	Name	Disposition	Bounding Event or Reference	Updated FSAR Designation
	15.4.8	Spectrum of Control Rod Ejection Accidents	Analyze		14.16
	15.4.9	Spectrum of Rod Drop Accidents (BWR)	Not applicable; BWR Event		
15.5	INCREASES IN REACTOR COOLANT INVENTORY				
	15.5.1	Inadvertent Operation of the ECCS that Increases Reactor Coolant Inventory	Overpressure Bounded Reactivity Bounded	15.2.1 15.4.6	
	15.5.2	CVCS Malfunction that Increases Reactor Coolant Inventory	Overpressure Bounded Reactivity Bounded	15.2.1 15.4.6	
15.6	DECREASES IN REACTOR COOLANT INVENTORY				
	15.6.1	Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve	Bounded	15.6.5	
	15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside of Containment	Bounded	15.6.5	
	15.6.3	Radiological Consequences of Steam Generator Tube Failure	Bounded	Ref. 8	14.15
	15.6.4	Radiological Consequences of a Main Steamline Failure Outside Containment	Not applicable; BWR Event		

Table 2-1 Disposition of Events Summary for Palisades (Cont.)

Event Classification	SRP Event Designation	Name	Disposition	Bounding Event or Reference	Updated FSAR Designation
	15.6.5	Loss of Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary	Analyze**	Ref. 8,10, 20&21	14.17 14.18 14.22
15.7	RADIOACTIVE RELEASE FROM A SUBSYSTEM OR COMPONENT				
	15.7.1	Waste Gas System Failure	Deleted*		14.21
	15.7.2	Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)	Deleted*		
	15.7.3	Postulated Radioactive Releases due to Liquid-Containing Tank Failures	Bounded	Ref. 8	14.20
	15.7.4	Radiological Consequences of Fuel Handling Accidents	Bounded	Ref. 8	14.19
	15.7.5	Spent Fuel Cask Drop Accidents	Bounded	Ref. 8	14.11

*This section of the Standard Review Plan has been deleted.

**The results of the analysis of the large break LOCA are reported in Reference 10.

Table 2-2 Summary of Results

Event	MDNBR (XNB)
15.1.3 Increase in Steam Flow(1)	1.46
15.2.1 Loss of External Load	1.71
15.3.1 Loss of Forced Reactor Coolant Flow	1.40
15.3.3 Reactor Coolant Pump Rotor Seizure	1.28
15.4.1 Uncontrolled Control Bank Withdrawal at Subcritical or Low Power	1.01(3)(5)
15.4.2 Uncontrolled Control Bank Withdrawal at Power(1)	1.25
15.4.3 Control Rod Misoperation(2)	
o Dropped Rod or Bank	1.25
o Single Rod Withdrawal(1)	1.22
o Core Barrel Failure	1.25
15.4.6 CVCS Malfunction resulting in Decreased Boron Concentration	(Adequacy of Shutdown Margin is Demonstrated.)
15.4.8 Control Rod Ejection	<1.17(4)

(1) 100% power case

(2) Results are based on conservative assumptions pertaining to control rod/bank configurations.

(3) <2.9% of the core is calculated to experience DNB

(4) <12.2% of the core is calculated to experience DNB

(5) Conservatively bounds Reactor Critical, Hot Standby and Hot Shutdown modes.

3.0 ANALYSIS OF PLANT TRANSIENTS

This section provides the results of the event disposition and analyses performed to support the Palisades Cycle 8 operation. Event numbering and nomenclature are consistent with the SRP to facilitate review.

Reference 3 contains information on the plant licensing basis as it affects the event analyses including:

- Classification of plant conditions
- Event acceptance criteria
- Single failure criteria
- Plant operating modes
- Analysis initial conditions
- Core and fuel design parameters
- Listings of systems and components available for accident mitigation, trip setpoints, time delays and component capacities.

These data, together with the design parameters[14] and the event specific input data given in Reference 3 and this report, represent a comprehensive summary of analysis inputs. The plant initial conditions, power distributions and neutronics data for Cycle 8 are given in Sections 15.0.1, 15.0.2 and 15.0.3, respectively.

Section 15.0.4 contains results of an analysis to verify the applicability of the TM/LP trip and the Inlet Temperature Limiting Condition of Operation (T_{inlet} LCO), given in Reference 3, to Cycle 8 operation.

15.0 ACCIDENT ANALYSES

15.0.1 PLANT INITIAL CONDITIONS

The nominal plant rated operating conditions are presented in Table 15.0.1-1. The uncertainties used in the accident analysis applicable to the operating conditions are:

Core Power	± 2%
Primary Coolant Temperature	± 5°F
Primary Coolant Pressure	± 50 psi
Primary Coolant Flow	± 3%

Table 15.0.1-1 Nominal Plant Operating Conditions

Core Thermal Power	2530 MWt
Pump Thermal Power (total)	15 MWt
System Pressure	2060 psia
Vessel Coolant Flow Rate*	120.3 Mlbm/hr
Core Coolant Flow Rate**	116.7 Mlbm/hr
Average Coolant Temperature	570.58°F
Core Inlet Coolant Temperature	543.65°F
Steam Generator Pressure	730 psia
Steam Flow Rate	10.97 Mlbm/hr
Feedwater Temperature	435°F
Number of Active Steam Generator Tubes* (per steam generator)	6023

* Reflects 29.3% average steam generator tube plugging.

** Reflects a 3% bypass flow.

15.0.2 POWER DISTRIBUTION

The radial and axial power peaking factors used in the analysis are presented in Table 15.0.2-1. Figures 15.0.2-1 and 15.0.2-2 show the limiting axial shapes for 100% power and 50% power, respectively. These axial shapes have ASIs of -0.139 for 100% power and -0.342 for 50% power. In this context, ASI is defined as:

$$\frac{P_{Lower} - P_{Upper}}{P_{Lower} + P_{Upper}}$$

P_{Lower} corresponds to the power generated in the lower half of the core and P_{Upper} corresponds to the power generated in the upper half of the core.

The Technical Specification(15) Limiting Condition of Operation radial peaking limits are increased by 3.5% for Palisades Cycle 8. The increase in radial peaking is to accommodate a low radial leakage fuel loading pattern.

The limiting DNBR occurs on an interior pin of an assembly with 208 rods. The Technical Specification(15) Limiting Conditions of Operation assure that the power distribution is maintained within these limits during normal operation. However, some events analyzed result in transient redistribution of the radial power peaking factors. Transient radial power redistribution is treated as described in Section 15.4.3.

The analyses in Reference 3 use an F_r factor that is 3% higher than that specified by the Technical Specifications. This augmentation factor was used to account for the fact that the axial shapes were derived from a one-dimensional core physics model rather than a three-dimensional model. For Cycle 8, minimum DNBR analyses were performed using axial shapes from both one-dimensional and three-dimensional core physics models. Comparison of the minimum DNBRs indicates that the core average axial shapes from the one-dimensional model are conservative relative to the hot assembly axial shapes from the three-dimensional model. Thus, the F_r augmentation factor was unnecessarily conservative and is e iinated from the analyses supporting Cycle 8.

Table 15.0.2-1 Core Power Distribution

	# Fuel Rods/Assembly	
Radial Peaking Factor:	208	216
- Peak interior rod**	1.70	1.73
- Engineering Uncertainty	1.03	1.03
Total Radial, $F_{r,T}$*	1.75	1.78
Axial Peaking Factor:		
- 100% power	1.39	
- 50% power	1.67	
Fraction of Power Deposited in Fuel	0.974	

* For power operation at less than rated, the radial peaking is $F_{r,T}[1+0.3(1-f)]$ for $0.5 \leq f \leq 1$ and 1.15 $F_{r,T}$ for $f<0.5$, where f is the fractional power of 2530 MWt.

** Proposed Technical Specification limit.

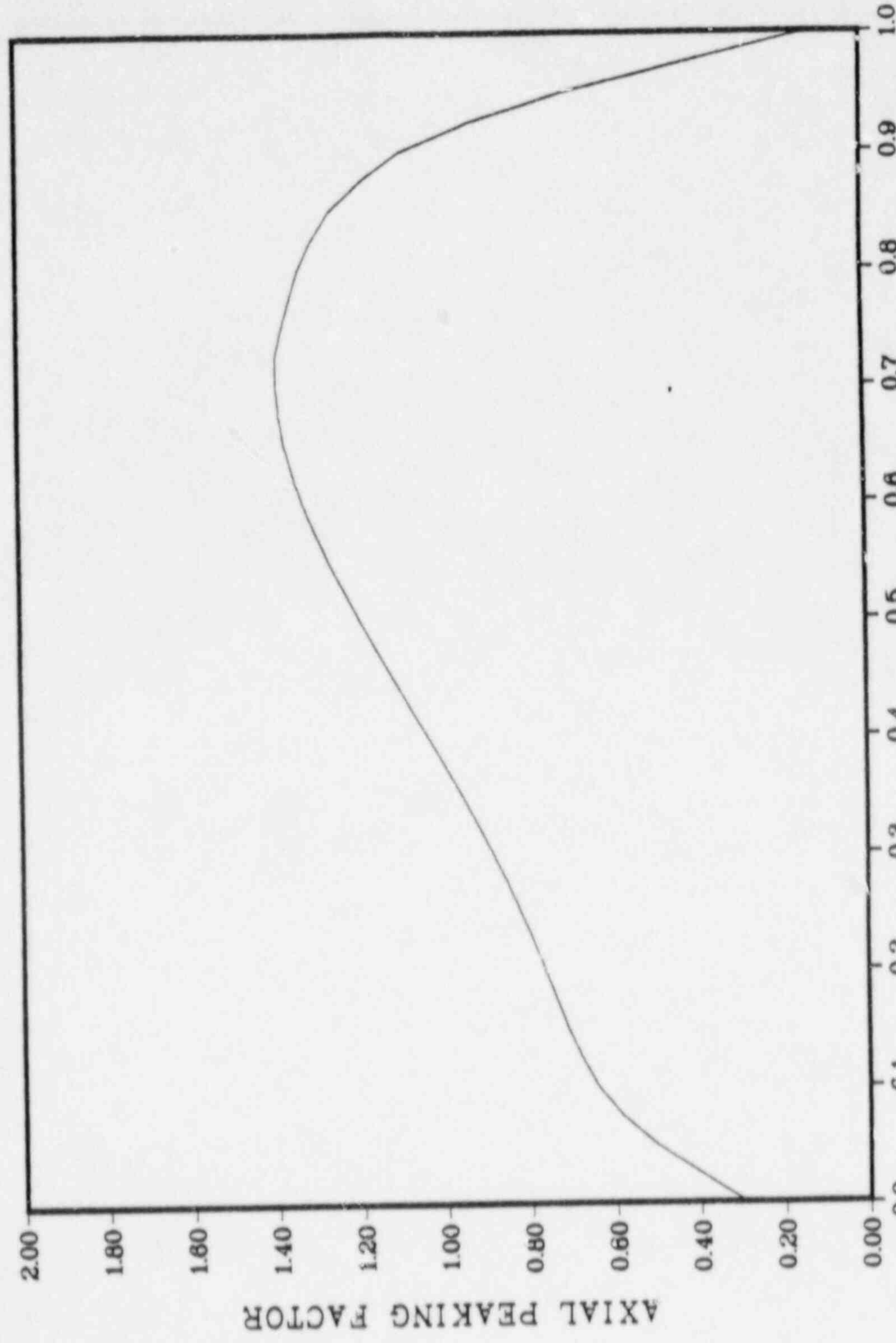

FIGURE 15.0.2-1 AXIAL POWER DISTRIBUTION AT 2530 MWt (ASI = -0.139)

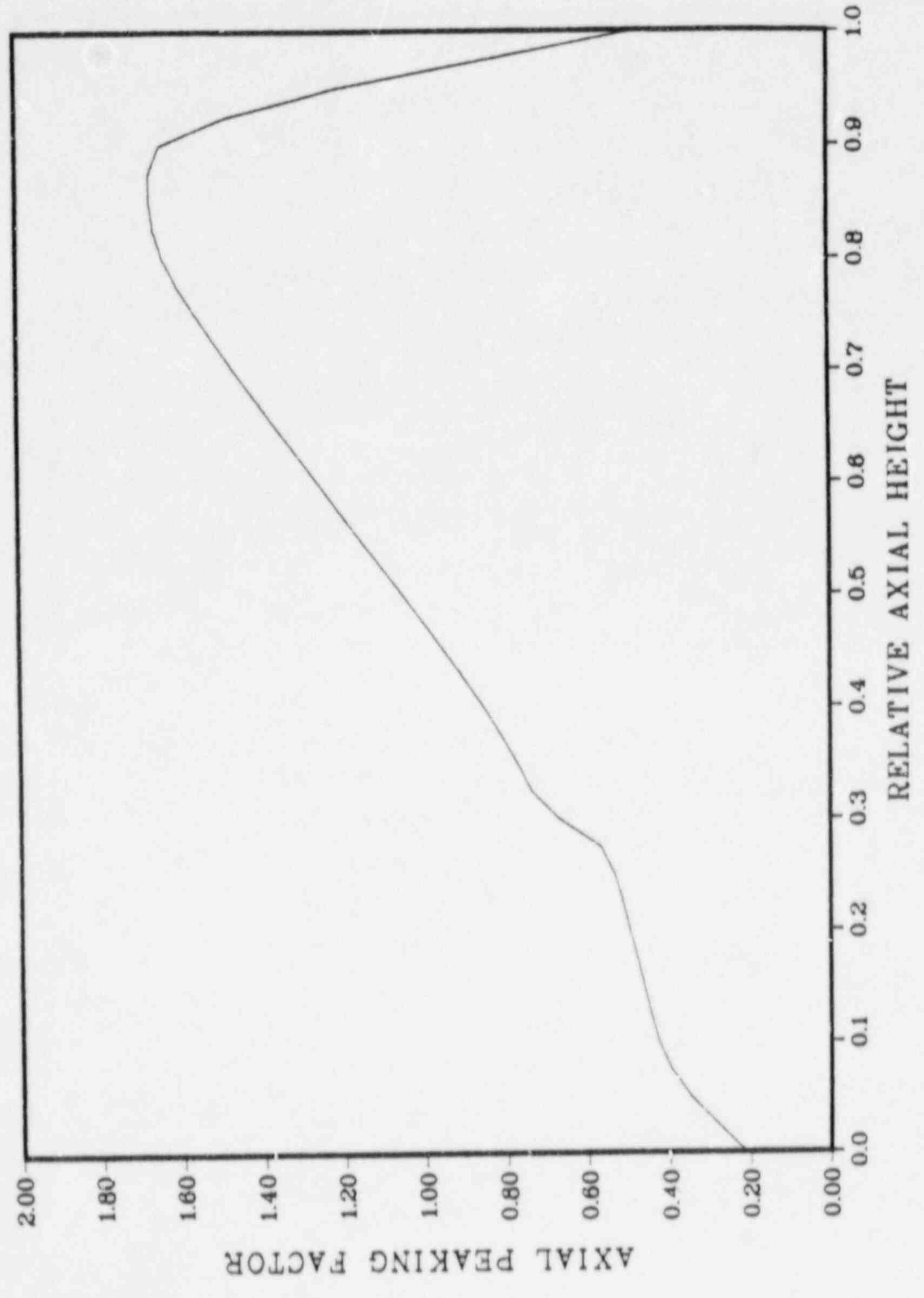

FIGURE 15.0.2-2 AXIAL POWER DISTRIBUTION AT 1265 MWt (ASI = -0.342)

15.0.3 REACTIVITY COEFFICIENTS USED IN THE SAFETY ANALYSIS

Table 15.0.3-1 presents the reactivity coefficients for Cycle 8 and those used in the analysis in Reference 3. As discussed in Reference 3, the set of parameters which most challenges the event acceptance criteria is used in each analysis.

Table 15.0.3-1 Palisades Cycle 8 Reactivity Parameters

Item	BOC Nominal	BOC Bounding	EOC Nominal	EOC Bounding
Moderator Temp Coef, 10^{-4} $\Delta\rho$/°F	0.25	0.5	-2.81	-3.5
Doppler Temp Coef. 10^{-5} $\Delta\rho$/°F	-1.36	-1.09	-1.56	-1.76
Moderator Pres Coef, 10^{-6} $\Delta\rho$/psi	-0.24	-1.0	2.66	7.0
Delayed Neutron Fraction	0.006	0.0075	0.0053	0.0045
Effective Neutron Lifetime, 10^{-6} seconds	21.6	41.9	24.6	19.9
U^{238} Atoms Consumed per Total Atoms Fissioned	.665	.54	.695	.70

15.0.4 TRIP SETPOINTS

Reference 3 presents the trip setpoints, biases, and time delays used in the analysis. The actual trip setpoints used in each transient analysis were biased such that the acceptance criteria for each event is most challenged.

A new T_{inlet} LCO and thermal margin/low pressure (TM/LP) trip were developed for operation with the modified RPS. Their development is presented in Reference 3. The T_{inlet} LCO was used to develop the initial conditions used in the transient analyses and the TM/LP trip was included in the transient analyses(3). The following two sections contain the results of an analysis to verify that the T_{inlet} LCO and TM/LP are applicable to Cycle 8 operation.

15.0.4.1 Inlet Temperature Limiting Condition Of Operation

The T_{inlet} LCO provides protection against penetrating DNB during limiting anticipated operational occurrence (AOO) transients. The T_{inlet} LCO derived in Reference 3 is given below:

$$T_{inlet} \leq 543.35 + .0575*(P-2060) + 5.0 \times 10^{-5}*(P-2060)^2 + 1.173*(W-120) - .0102*(W-120)^2$$

$$1800 \leq P \leq 2200 \text{ psia}$$
$$100 \leq W \leq 130 \text{ Mlb/hr.}$$

As shown in Table 2-2, the most limiting AOO transient that does not produce a reactor trip is the inadvertent drop of a full length control assembly. The T_{inlet} LCO must provide DNB protection for this transient assuming a return to full power with enhanced peaking due to the anomalous control assembly insertion pattern. The T_{inlet} LCO was verified for Cycle 8 using the XCOBRA-IIIC computer code(6,16) with a conservative peaking augmentation factor.

The XCOBRA-IIIC calculations were run to demonstrate that the inlet temperature allowed by the T_{inlet} LCO results in a DNBR greater than 1.17 for the XNB correlation[17,18] over a range of pressurizer pressures and primary coolant system flow rates. These calculations were performed at 102 percent of rated power, i.e. 2530 MWt, and an axial shape with an axial shape index (ASI) of -.139. Based on an analysis of axial shapes within the range of -.14 to +.544, this was the limiting shape for full power transients for Cycle 8. The derived T_{inlet} LCO supports operation at 100 percent of rated power for measured plant ASIs greater than -.08 and less than +.484. This allows for a plant ASI measurement uncertainty of $\pm$.06.

The verification analysis includes the following uncertainties and transient allowances:

- 2% power measurement uncertainty
- $\pm$.06 ASI measurement uncertainty
- $\pm$50 psia pressurizer pressure measurement uncertainty
- $\pm$7°F inlet temperature (5°F tilt allowance + 2°F measurement uncertainty)
- $\pm$6% on the flow rate (3% bypass flow + 3% measurement uncertainty)
- Transient allowances from Reference 3 for a dropped rod event: 65 psia decrease in the pressurizer pressure; a 4.7°F decrease in the inlet temperature; and an increase in the flow rate of 0.42 Mlb/hr.

Applying these biases to the calculations resulted in a minimum DNBR greater than 1.17 for pressure and flow points within the range of the T_{inlet} LCO at full power.

In order that the plant can still operate should the measured ASI become less than -.08 the applicability of the T_{inlet} LCO equation was extended to a measured ASI of -.30 at 70 percent of rated power. This extended T_{inlet} LCO range was verified to be applicable to Cycle 8 in the manner described above.

The limiting part-power axial, shown in Figure 15.0.2-2 was used for these calculations.

15.0.4.2 Thermal Margin/Low Pressure (TM/LP) Trip

The modified RPS includes the hardware for a new TM/LP trip which is to be installed at the Palisades reactor. This new TM/LP is an improvement over the previous trip in that it allows monitoring of the core axial shape index.

The function of the TM/LP trip is to protect against slow heatup and depressurization transient events. In order to perform this function, the TM/LP trip must initiate a scram signal prior to exceeding the specified acceptable fuel design limits (SAFDLs) on departure from nucleate boiling (DNB) or before the average core exit temperature exceeds the saturation temperature. The SAFDL insures that there is no damage to the fuel rods and the limit on core exit saturation is imposed to assure meaningful thermal power measurements.

The TM/LP trip works in conjunction with the other trips and the limiting conditions of operation (LCO) on control rod group position, radial peaking, and reactor coolant flow. The variable high power (VHP) trip is factored into the TM/LP development by limiting the maximum possible power that can be achieved at a particular radial peaking to 10% above the power corresponding to that radial peaking. The LCO on the control rod group position is included in the TM/LP through monitoring of the axial shapes and the LCO on radial peaking is factored in by including its variation with power level in the TM/LP development. Finally, the LCO on reactor coolant flow is built into the TM/LP through the use of conservative flows throughout its development.

The development of the TM/LP trip setpoints are documented in Reference 3. From Reference 3, the TM/LP trip is given as:

$$P_{var} = 1563.7\ (QA)\ (QR_1) + 12.3\ (T_{in}) - 6503.4$$

where:

$$QR_1 = 0.412\ (Q) + 0.588 \qquad Q \le 1.0$$
$$QR_1 = Q \qquad Q \ge 1.0$$

and,

$$QA = +.226\ (ASI) + .964 \qquad +.162 \le ASI \le +.544$$
$$QA = -.521\ (ASI) + 1.085 \qquad -.156 \le ASI \le +.162$$
$$QA = -.691\ (ASI) + 1.058 \qquad -.655 \le ASI \le -.156$$

This TM/LP is applicable over a pressure range from 1700 psia to 2300 psia and to a minimum measured HZP primary coolant flow rate of 124.3 Mlb/hr.

The TM/LP trip function was verified for Cycle 8 by first determining a set of limiting axial shapes. The limiting axial shapes were determined in .06 ASI increments covering the ASI range defined by the T_{inlet} LCO[3]. The limiting axial shapes were used in the XCOBRA-IIIC model to ensure that the minimum DNBR allowed by the TM/LP trip function is greater than the XNB correlation[17,18] 95/95 limit of 1.17. Thus, the TM/LP trip[3] is verified to be applicable over the possible range of axial shapes for Cycle 8.

15.0.5 DISPOSITION AND ANALYSIS OF EVENTS

The following sections discuss the disposition and analysis of each of the SRP Chapter 15 events. Each event is numbered according to the corresponding SRP designation. The plant licensing basis, single failure criteria and acceptance criteria are outlined in Reference 3.

15.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

15.1.1 DECREASE IN FEEDWATER TEMPERATURE

15.1.1.1 Event Description

A decrease in feedwater temperature event may initiate due to the loss of one of several of the feedwater heaters. This loss may be due to the loss of extraction steam flow from the turbine generator or due to an accidental opening of a feedwater heater bypass line.

The event results in a decrease of the secondary side enthalpy leading to an increase in the primary-to-secondary side heat transfer. The steam generator outlet temperature on the primary side decreases causing the core inlet temperature to also decrease. In the presence of a negative moderator coefficient, reduced core inlet temperature results in an increase in the core power and a decrease in thermal margin.

15.1.1.2 Event Disposition and Justification

Reference 2 disposed this event as being bounded by the Increase in Steam Flow event (Event 15.1.3). The changes for Cycle 8 do not change this disposition. Therefore, no further analysis is required for Cycle 8.

15.1.2 INCREASE IN FEEDWATER FLOW

15.1.2.1 Event Description

The Increase in Feedwater Flow event is initiated by a failure in the feedwater system. The failure may be a result of: (1) a complete opening of a feedwater regulating valve; (2) over-speed of the feedwater pumps with the feedwater valve in the manual position; (3) inadvertent startup of the second feedwater pump at low power; (4) startup of the auxiliary feedwater system; or, (5) inadvertent opening of the feedwater control valve bypass line.

The event results in an increase in the primary-to-secondary side heat transfer due to increased feedwater flow. The steam generator outlet temperature on the primary side decreases causing the core inlet temperature to also decrease. In the presence of a negative moderator coefficient, reduced core inlet temperature results in an increase in the core power and a decrease in thermal margin.

15.1.2.2 Event Disposition and Justification

Reference 2 disposed this event as being bounded by the Increase in Steam Flow event (Event 15.1.3). The changes for Cycle 8 do not change this disposition. Therefore, no further analysis is required for Cycle 8.

15.1.3 INCREASE IN STEAM FLOW

15.1.3.1 Event Description

This event is initiated by a failure or misoperation of the main steam system that results in an increase in steam flow from the steam generators. The increased steam flow creates a mismatch between the heat being generated in the core and that being extracted by the steam generators. As a result of this power mismatch, the primary-to-secondary heat transfer increases and the primary system cools down. If the moderator temperature coefficient is negative, the cooldown of the primary system coolant would cause an insertion of positive reactivity and the potential erosion of thermal margin.

15.1.3.2 Event Disposition and Justification

This event was disposed to be analyzed for modified RPS operation for both HZP and HFP conditions[2]. The system response for both cases was evaluated using PTSPWR2[5] and the event minimum DNBR was calculated using XCOBRA-IIIC[6].

For the HZP case, the control rods were initially inserted in the PTSPWR2 simulation[3]. This eliminates the insertion of shutdown reactivity due to activation of the reactor trip system. The system response will remain the same for Cycle 8 as for the modified RPS analysis.

The increased radial peaking for Cycle 8 will change the thermal margin for this event. The thermal margin for the Increase in Steam Flow event from HZP is, therefore, disposed to be reanalyzed for Cycle 8. As was the case for the modified RPS analysis, the thermal margin for the HZP case will be analyzed using the Modified Barnett critical heat flux correlation[4].

For the Increase in Steam Flow event from HFP, the reactor trip system acts to terminate the event. From Reference 3, the variable high power and the TM/LP trips protect the plant from penetrating DNBR limits. For an increase in radial peaking for Cycle 8, the primary system response to an increase in steam flow event will not change for the HFP case. As in the HZP case, the increase in radial peaking will impact minimum DNBR. Therefore, the Increase in Steam Flow event from HFP for Cycle 8 will be analyzed to calculate the minimum DNBR for this event.

15.1.3.3 Analysis and Results

The minimum DNBR for this event initiated from full power occurred for a steam flow increase of about 112%[3]. At this steam flow rate, the TM/LP and the variable high power trips coincide producing nearly simultaneous trip signals. The junction of these two trips represents the worst possible DNB conditions, that is, maximum core power is attained combined with a low pressurizer pressure. The calculated minimum DNBR for Cycle 8 is 1.46. The peak LHGR is calculated to be 14.9 kW/ft.

For the hot shutdown case, the event was initiated by a rapid opening of the atmospheric dump valves and the turbine bypass valves resulting in a steam flow increase of 28% of the nominal full power steam flow. A bounding value for the negative moderator temperature coefficient (EOC conditions) was assumed. Due to the cooldown of the primary coolant, coupled with a negative moderator temperature coefficient, the reactor becomes critical resulting in a significant return-to-power. The Doppler temperature coefficient eventually terminates this event. The minimum critical heat flux ratio (CHFR) computed for this case, using the Modified Barnett correlation, is 2.05. The peak pellet LHGR is calculated to be 8.0 kW/ft.

15.1.3.4 Conclusion

The results of the analysis demonstrate that the event acceptance criteria are met since the minimum DNBR predicted for the full power case is greater than the XNB correlation safety limit of 1.17 and the minimum CHFR predicted for the hot shutdown case is greater than the Modified Barnett CHFR limit of 1.135. The correlation limit assures that with 95% probability and 95% confidence, DNB is not expected to occur; therefore, no fuel is expected to fail. The fuel centerline melt threshold of 21 kW/ft is not approached in this event.

15.1.4 INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE

15.1.4.1 Event Description

This event is initiated by an increase in steam flow caused by the inadvertent opening of a steam generator relief or safety valve. The increase in steam flow rate causes a mismatch between the heat generation rate on the primary side and the heat removal rate on the secondary side.

15.1.4.2 Event Disposition and Justification

The increase in steam flow due to opening a steam generator valve is less than that considered in the Increase in Steam Flow event (Event 15.1.3)[2]. Therefore, an inadvertent opening of a steam generator relief or safety valve is bounded by Event 15.1.3[2]. This conclusion will not change for Cycle 8.

15.1.5 STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE OF CONTAINMENT

15.1.5.1 Event Description

A steam line piping failure event, or steam line break (SLB), is initiated by a rupture of a main steam line pipe causing an uncontrolled steam release from the secondary system. As a result of the uncontrolled release of steam, the heat extraction rate from the primary side is no longer equal to the core heat generation rate. This power mismatch increases the primary-to-secondary side heat transfer and, consequently, reduces the primary side temperatures. When this overcooling on the primary side is coupled with a negative moderator temperature coefficient, the shutdown margin after scram can potentially be eroded. Such an erosion of shutdown margin may result in a return-to-power which, in turn, challenges thermal margin. The consequences of this event are governed by the steam flow rate out of the ruptured steam line, the primary pump operating assumptions (i.e., with or without offsite power), the magnitude of the moderator coefficient and the initial primary side operating state.

15.1.5.2 Event Disposition and Justification

For a steam generator tube plugging level of 29%, the SLB event was disposed as being bounded by previous analyses[2]. The SLB event for Cycle 8 is disposed to be bounded by the current analysis of record. The conservatisms inherent in the SLB analysis with regard to the stuck rod and bounding reactivity feedback are not significantly affected by the changes for Cycle 8.

15.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

15.2.1 LOSS OF EXTERNAL LOAD

15.2.1.1 Event Description

A Loss of External Load event is initiated by either a loss of external electrical load or a turbine trip. Upon either of these two conditions, the turbine stop valve is assumed to rapidly close (0.1 second). Normally, a reactor trip would occur on a turbine trip. However, to calculate a conservative system response, the reactor trip on turbine trip is disabled. The steam dump system (atmospheric dump valves- ADVs) is assumed to be unavailable. These assumptions allow the Loss of External Load event to bound the consequences of: Event 15.2.2 (Turbine Trip- steam dump system available); Event 15.2.3 (Loss of Condenser Vacuum- steam dump system unavailable); and, Event 15.2.4 (Closure of the MSIV- valve closure time is > 0.1 second).

The Loss of External Load event primarily challenges the acceptance criteria on primary system overpressurization and DNBR. The event results in an increase in the primary system temperatures due to an increase in the secondary side temperature. As the primary system temperatures increase, the coolant expands into the pressurizer causing an increase in the pressurizer pressure. The primary system is protected against overpressurization by the pressurizer safety and relief valves. Pressure relief on the secondary side is afforded by the steam line safety/relief valves. Actuation of the primary and secondary system safety valves limits the magnitude of the primary system temperature and pressure increase.

With a positive moderator temperature coefficient, increasing primary system temperature results in an increase in core power. The increasing primary side temperatures and power reduces the margin to thermal limits (i.e., DNBR limits) and challenges the DNBR acceptance criteria.

15.2.1.2 Event Disposition and Justification

The Loss of External Load from HFP was disposed to be analyzed for modified RPS operation[2]. The event initiated from full power bounds all other operating modes. The system response for the DNBR and pressurization cases was evaluated using PTSPWR2[5] and the event minimum DNBR was calculated using XCOBRA-IIIC[6]. In the modified RPS analysis of the Loss of External Load pressurization case, the reactor trip system acts to terminate the event by activating a high pressurizer pressure trip signal[3]. For an increase in radial peaking for Cycle 8, the primary system pressure response to a loss of load will not change for the pressurization case. Therefore, this case will not require reanalysis for Cycle 8 operation.

The increase in radial peaking for Cycle 8 will, however, impact minimum DNBR. Therefore, the Loss of External Load event (minimum DNBR case) from HFP for Cycle 8 is disposed to be reanalyzed. The event minimum DNBR will be calculated using XCOBRA-IIIC[6] with the core conditions taken from the limiting PTSPWR2[5] run for the modified RPS analysis.

15.2.1.3 Analysis and Results

The transient response to a Loss of External Load for the minimum DNBR case is given in Reference 3. Using XCOBRA-IIIC[6], the minimum DNBR for Cycle 8 is computed to be 1.71. The peak pellet LHGR is calculated to be 13.5 kW/ft.

15.2.1.4 Conclusion

The calculated minimum DNBR for the event is above the XNB critical heat flux correlation safety limit, so the DNB SAFDL is not penetrated in this event. Peak pellet LHGR for the event is well below the fuel centerline melt criterion of 21 kW/ft. Applicable acceptance criteria for the event are therefore met.

15.2.2 TURBINE TRIP

15.2.2.1 Event Description

This event is initiated by a turbine trip which results in the rapid closure of the turbine stop valves. A reactor trip would occur on a turbine trip and the steam dump system would operate to mitigate the consequences of this event. The primary system is protected against overpressurization by the pressurizer safety and relief valves. Pressure relief on the secondary side is afforded by the steam line safety/relief valves.

15.2.2.2 Event Disposition and Justification

The assumptions made in the Loss of External Load event (Event 15.2.1) bound the consequences of a Turbine Trip event. Specifically, the Loss of External Load event considers the following: a conservatively fast turbine stop valve closure time; reactor trip does not occur on a turbine trip; and, the atmospheric dump valves are assumed to be unavailable.

The Turbine Trip event was disposed as being bounded by the Loss of External Load event (Event 15.2.1) for modified RPS operation[2]. The changes for Cycle 8 will not invalidate this disposition.

15.2.3 LOSS OF CONDENSER VACUUM

15.2.3.1 Event Description

This event is initiated by a reduction in the circulating water flow or an increase in the circulating water temperature which can impact the condenser back pressure. This condition can result in a turbine trip without the availability of steam bypass to the condenser. The primary system is protected against overpressurization by the pressurizer safety and relief valves. Pressure relief on the secondary side is afforded by the steam line safety/relief valves.

15.2.3.2 Event Disposition and Justification

The assumptions made in the Loss of External Load event bound the consequences of a Loss of Condenser Vacuum transient. The Loss of Condenser Vacuum event was disposed as being bounded by the Loss of External Load event (Event 15.2.1) for rated power and power operating modes[2]. The scenario of this event from other operating modes allows sufficient time for the operator to control the primary and secondary system temperatures[2]. These conclusions will not change for Cycle 8.

15.2.4 CLOSURE OF THE MAIN STEAM ISOLATION VALVES (MSIV) (BWR)

15.2.4 Event Description

Closure of the Main Steam Isolation Valve event is initiated by the loss of control air to the MSIV operator. The valves are swinging check valves designed to fail in the closed position. The inadvertent closure of the MSIVs is primarily a BWR event, however, the closure of these valves in a PWR can drastically reduce the steam load.

15.2.4.2 Event Disposition and Justification

The closure time of the MSIVs is less than 5 seconds, but greater than the value used in Event 15.2.1 (0.1 seconds). A MSIV closure event will progress in a similar fashion as a Loss of External Load (Event 15.2.1), but at a slower rate. The consequences of Event 15.2.1 will bound those for Event 15.2.4 because of the more rapid valve closure time[2].

Since the changes made for Cycle 8 will not impact the system response, Event 15.2.4 will continue to be bounded by Event 15.2.1.

15.2.5 STEAM PRESSURE REGULATOR FAILURE

Palisades does not have steam pressure regulators. Therefore, the Steam Pressure Regulator Failure event is not considered in this analysis.

15.2.6 LOSS OF NONEMERGENCY A.C. POWER TO THE STATION AUXILIARIES

15.2.6.1 Event Description

A Loss of Nonemergency A.C. Power to Station Auxiliaries event may be caused by a complete loss of the offsite grid together with a turbine generator trip or by a failure in the onsite A.C. power distribution system.

The loss of A.C. power may result in the loss of power to the primary coolant pumps and the main feedwater pumps. The combination of the decrease in primary coolant flow rate, the cessation of main feedwater flow and trip of the turbine generator compounds the event consequences. The decrease of both primary coolant flow and main feedwater decreases the primary-to-secondary system heat transfer resulting in the heatup of the primary system coolant. The increase in primary system coolant temperature increases the overpressurization potential and increases the threat of penetrating DNB.

The event is most limiting when initiated from full power conditions. During this mode of operation the amount of stored heat in the fuel rods is the greatest and the margin to DNB is minimized.

15.2.6.2 Event Disposition and Justification

This event can be separated into two distinct phases: the near-term and the long-term. The near-term phase is characterized by the loss of power resulting in the coastdown of the primary coolant pumps, the coastdown of the main feedwater pumps and the trip of the turbine generator. The coastdown of the primary coolant pumps causes an immediate reduction in thermal margin. The trip of the reactor and the subsequent insertion of control rods terminates the challenge to DNB limits.

The near-term phase of th event is similar to that of a Loss of Forced Reactor Coolant Flow transient (Event 15.3.1). The near-term consequences of this event are addressed in the analysis of Event 15.3.1[3].

The long-term consequences of a Loss of A.C. Power event are determined by the heat removal capacity of the auxiliary feedwater system. The long-term portion is similar to the Loss of Normal Feedwater Flow transient (Event 15.2.7). The long-term effects are, therefore, addressed by the analysis of the Loss of Normal Feedwater Flow event[3]. The changes for Cycle 8 will not alter this conclusion.

15.2.7 LOSS OF NORMAL FEEDWATER FLOW

15.2.7.1 Event Description

A Loss of Normal Feedwater Flow transient is initiated by the trip of the main feedwater pumps or a malfunction in the feedwater control valves. The loss of main feedwater flow decreases the amount of subcooling in the secondary-side downcomer which diminishes the primary-to-secondary system heat transfer and leads to an increase in the primary system coolant temperature. As the primary system temperatures increase, the coolant expands into the pressurizer which increases the pressure by compressing the steam volume.

The opening of the secondary-side safety valves controls the heatup of the primary-side. The long-term cooling of the primary system is governed by the heat removal capacity of the auxiliary feedwater flow. The auxiliary feedwater pumps are automatically started upon a steam generator low liquid level signal.

15.2.7.2 Event Disposition and Justification

A Loss of Normal Feedwater Flow event is only credible for rated power and power operating conditions[2]. The worst consequences occur when the feedwater is lost during rated power operation since more stored heat is contained in the fuel than in other modes of operation.

The short-term impacts of the Loss of Normal Feedwater Flow event challenges the DNB and the primary system overpressurization acceptance criteria. The DNB challenge is maximized when it is assumed that offsite power is lost causing the primary coolant pumps to coastdown. The Loss of Forced Reactor Coolant Flow event (Event 15.3.1) addresses the short-term DNB consequences of a Loss of Normal Feedwater Flow transient. After the reactor trip system is activated, the core power is drastically reduced alleviating the challenge to DNB.

The long-term effects of this event primarily challenges the pressurization limits of the primary system due to the filling of the pressurizer and steam generator dryout. If the pressurizer were to fill completely solid with liquid, the primary system pressure control would be lost and primary liquid would be expelled through the pressurizer safety valves.

The dryout of a steam generator causes the loss of a primary-to-secondary system heat sink exacerbating the primary-side heatup. The long-term consequences of a Loss of Normal Feedwater Flow event were analyzed in Reference 2.

The changes for Cycle 8, will not impact the system response to a Loss of Normal Feedwater Flow. The DNB challenge is addressed in the analysis of the Loss of Forced Reactor Coolant Flow event (Event 15.3.1). The primary system pressurization and pressurizer fill cases will not be impacted. Therefore, this event is disposed as being bounded by the modified RPS analysis for the pressurization, steam generator dryout and pressurizer fill cases[3]. The DNB case is bounded by the Loss of Forced Reactor Flow event (Event 15.3.1).

15.2.8 FEEDWATER SYSTEM PIPE BREAKS INSIDE AND OUTSIDE CONTAINMENT

15.2.8.1 Event Description

A Feedwater System Pipe Break event occurs when a main feedwater system pipe is ruptured. The ruptured pipe will cause a blowdown of the affected steam generator if the break occurs upstream of the feedline check valve. If the rupture occurs downstream of the check valve, the event would behave much like the Loss of Normal Feedwater Flow transient. Since the auxiliary feedwater flow is injected into the steam generators via a separate piping network than the main feedwater, the delivery of auxiliary feedwater will not be interrupted by the pipe rupture.

The event results in both a primary system cooldown and a heatup. Initially, the event results in a cooldown of the primary-side coolant due to the energy removal during the blowdown stage of the event. The eventual depletion of secondary-side inventory and lack of main feedwater will cause the primary system to heatup much like a Loss of Normal Feedwater Flow event.

15.2.8.2 Event Disposition and Justification

The event was disposed in Reference 2 as being bounded during rated power operation as follows:

1. The cooldown aspect of the event is bounded by the Steam Line Break event (Event 15.1.5).

2. The heatup effects are bounded by the Loss of External Load event (Event 15.2.1) for the primary system overpressurization and the Loss of Normal Feedwater Flow event (Event 15.2.7) for the long-term cooling requirements.

Feedwater pipe breaks from modes other than rated power result in a primary system cooldown and are bounded by the Steam Line Break accident (Event 15.2.8).

The changes for Cycle 8 will not impact the system response to a Feedwater System Pipe Break event. Therefore, this event is disposed as being bounded as described above.

15.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW

15.3.1 LOSS OF FORCED REACTOR COOLANT FLOW

15.3.1.1 Event Description

The Loss of Forced Reactor Coolant Flow transient is initiated by a disruption of the electrical power supplied to or a mechanical failure in a primary coolant system (PCS) pump. These failures may result in a complete or partial loss of forced coolant flow.

The impact of losing a PCS pump or pumps is a decrease in the active flow rate in the reactor core and, consequently, an increase in core temperatures.

Prior to reactor trip, the combination of decreased flow and increased temperature poses a challenge to DNB limits. The event is terminated by the PCS low flow trip.

15.3.1.[illegible] [illegible] <u>[illegible]isposition and Justification</u>

[illegible] ng scenario for this event is to initiate the loss of four PCS pu[illegible] a[illegible] ated power condition(2). Plant operation with a reduced low [illegible] trip setpoint (60% of rated four PCS flow) for three PCS pump [illegible] or [illegible] reduced power (39% of rated) has been justified(7). This o[illegible] g state is allowed for a limited period of time for repair/pump startup, to provide for an orderly shutdown, or to provided for the conduct of reactor internals noise monitoring test measurements.

For Cycle 8 operation, the increase in radial peaking will impact the minimum DNBR. To assess the minimum DNBR for Cycle 8 operation, the minimum DNBR calculation will be reanalyzed for the loss of four PCS pumps from rated power.

The calculated minimum DNBR for a Loss of Forced Coolant Flow event from a three primary coolant pump initial condition is bounded by the results of the rated power event.(7)

15.3.1.3 <u>Analysis and Results</u>

The transient is initiated by tripping all four primary coolant pumps. As the pumps coast down, the core flow is reduced, causing a reactor scram on low flow. As the flow coasts down, primary temperatures increase. This increase in temperature causes a subsequent power rise due to moderator reactivity feedback. The primary challenge to DNB is from the decreasing flow rate and resulting increase in coolant temperatures. Using XCOBRA-IIIC, the minimum DNBR for Cycle 8 is computed as 1.40. The peak pellet LHGR is calculated to be 13.1 kW/ft.

15.3.1.4 Conclusion

The XNB critical heat flux safety correlation limit of 1.17 is not penetrated, so event results are acceptable with respect to the DNBR SAFDL. Maximum peak pellet LHGR for this event is below the incipient fuel centerline melt criterion of 21 kW/ft. Applicable acceptance criteria for the event are therefore met for Cycle 8.

15.3.2 FLOW CONTROLLER MALFUNCTION

There are no flow controllers on the PCS at Palisades. Therefore, this event is not credible.

15.3.3 REACTOR COOLANT PUMP ROTOR SEIZURE

15.3.3.1 Event Description

This event is initiated by a seizure of a PCS pump rotor. The seizure causes an immediate reduction in PCS flow rate. As in the Loss of Forced Coolant Flow event (Event 15.3.1), the impact of losing a PCS pump is a decrease in the active flow rate in the reactor core and, consequently, an increase in core temperatures. Prior to reactor trip, the combination of decreased flow and increased temperature poses a challenge to DNB limits. The event is terminated by the PCS low flow trip.

15.3.3.2 Event Disposition and Justification

The most limiting scenario for a Reactor Coolant Pump Seizure event occurs for rated power or power operating conditions[2]. Plant operation with a reduced low flow reactor trip setpoint (60% of rated four PCS flow) for three PCS pump operation at reduced power was justified in Reference 7. Results of the three PCS pump case from reduced power were bounded by the event initiated from rated power[7].

For Cycle 8 operation, the increase in radial peaking impacts the minimum DNBR. To assess the minimum DNBR for Cycle 8 operation, the minimum DNBR calculation will be reanalyzed for a pump rotor seizure from rated power conditions. This event initiated from three PCS pump operation at reduced power will remain bounded by the full power event for Cycle 8.

15.3.3.3 Analysis and Results

The first locked rotor case is analyzed using the calculated value of core flow. Assuming the locked pump loss coefficient given by the homologous curves at zero pump speed, the core flow is 78% of the nominal full-power, four-pump operation value. The second case is analyzed at 74.7% flow as specified in the Technical Specifications (Reference 15, page 2-7). The XCOBRA-IIIC calculated minimum DNBRs are 1.35 and 1.28 for Case 1 and Case 2, respectively. The peak pellet LHGR for each case is 13.1 kW/ft.

15.3.3.4 Conclusion

The XNB critical heat flux correlation safety limit of 1.17 is not penetrated and no fuel failures are expected for this infrequent event. Thus, applicable acceptance criteria for this event are met for Cycle 8.

15.3.4 REACTOR COOLANT PUMP SHAFT BREAK

15.3.4.1 Event Description

This event is initiated by a failure of a PCS pump shaft resulting in a free-wheeling impeller. The impact of a coolant pump shaft break is a loss of pumping power from the affected pump and a reduction in the PCS flow rate. The flow reduction due to the seizure of a pump rotor is more severe than that for a shaft break; however, the potential for flow reversal is greater for the shaft break event. The event is terminated by the low reactor coolant flow trip.

15.3.4.2 Event Disposition and Justification

The event is most limiting at rated power conditions because of a minimum margin to DNBR limits. The initial flow reduction for this event is bounded by that for the Reactor Coolant Pump Rotor Seizure event (Event 15.3.3). The potential for greater reverse flow due to a shaft break is accounted for in the seized rotor analysis by decreasing, internally in PTSPWR2[5], the rotor inertia to zero at the time of predicted reversed flow.

The changes made for Cycle 8 will not impact the system response to a PCS pump shaft break. The impact to minimum DNBR is bounded by the analysis of Event 15.3.3. Therefore, this event is disposed as being bounded.

15.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

15.4.1 UNCONTROLLED CONTROL ROD ASSEMBLY (CRA) WITHDRAWAL FROM A SUBCRITICAL OR LOW POWER STARTUP CONDITION

15.4.1.1 Event Description

This event is commenced by an uncontrolled withdrawal of a control rod bank. This withdrawal adds positive reactivity to the core which leads to a power excursion. Event 15.4.1 considers the consequences of the control bank withdrawal at subcritical or low initial power levels.

As the control bank is withdrawn, the positive reactivity insertion causes a significant core power increase as the reactor approaches prompt criticality. As the core power increases, the core average and hot leg temperatures also increase. Due to the increasing power and temperatures, the DNB limits are challenged. An additional assumption included in the event analysis for modified RPS operation is that the plant is operating with three PCS pumps(3). The transient eventually terminates on an overpower reactor trip signal.

15.4.1.2 Event Disposition and Justification

For Cycle 8 operation, the changes to radial peaking will impact the minimum DNBR for this event. The system response to this event will, however, not be affected. To assess the minimum DNBR for Cycle 8 operation, the minimum DNBR calculation will be reanalyzed.

15.4.1.3 Analysis and Results

This event was analyzed assuming three primary coolant pumps to be operating. The event is initiated with control bank withdrawal. The minimum DNBR calculated for the event is 1.01, which is below the 1.17 95/95 DNB safety limit for the XNB critical heat flux correlation. The percent of the core experiencing boiling transition was calculated to be less than 2.9% for Cycle 8, as compared to less than 2.3% for the Reference 3 analysis. Due to conservative assumptions in the fuel failure calculation, the offsite radiological doses for the uncontrolled bank withdrawal from low power are less than 10% of the 10 CFR 100 limits for Cycle 8.

15.4.1.4 Conclusions

In this infrequent event, only a small fraction of the core is calculated to experience boiling transition. Possible radiological releases are less than 10% of the 10 CFR 100 guidelines. Therefore, this event meets the applicable acceptance criteria for Cycle 8 operation.

15.4.2 UNCONTROLLED CONTROL ROD BANK WITHDRAWAL AT POWER

15.4.2.1 Event Description

As with Event 15.4.1, this event is initiated by an uncontrolled withdrawal of a control rod bank. This withdrawal adds positive reactivity to the core which leads to potential power and temperature excursions. Event 15.4.2 considers the consequences of control bank withdrawals at rated and operating initial power levels.

As the control bank is withdrawn, the positive reactivity insertion causes an increase in core power and in primary coolant system temperatures. Due to the increasing power and temperatures, the DNB limits are challenged. In most cases, the transient will terminate on a variable high power, a TM/LP or a high pressurizer pressure trip; however, some cases do not activate a reactor protection system trip.

15.4.2.2 Event Disposition and Justification

The analysis performed for modified RPS operation[3] evaluates the consequences of an uncontrolled rod withdrawal from both rated power and 50% of rated power initial states. A spectrum of reactivity insertion rates were evaluated in order to bound events ranging from boron dilutions to fast control bank withdrawals.

The changes for Cycle 8 operation will impact DNBR for both the full and part-power cases. To assess the minimum DNBR for Cycle 8 operation, the respective limiting minimum DNBR point for 50% and 100% power conditions are reanalyzed for Cycle 8.

15.4.2.3 Analysis and Results

The uncontrolled rod withdrawal transients were analyzed for full power (100% of rated) and mid power (50% of rated). The calculated minimum DNBR occurred for a rod withdrawal from 100% of rated thermal power. The mid power case series was, in general, less limiting than the full power cases.

The limiting rod withdrawal at 50% power and EOC kinetics occurred at an insertion rate of 3×10^{-5} $\Delta\rho$/sec. The minimum DNBR was calculated as 2.36. This transient did not scram, but was ended when the rods were fully withdrawn. The peak pellet LHGR for the 50% power case is calculated to be 10.3 kW/ft.

The limiting uncontrolled control rod bank withdrawal at 100% power and EOC kinetics occurred at an insertion rate of 17.0×10^{-5} $\Delta\rho$/sec. The minimum DNBR was calculated at 1.25. This transient tripped on a thermal margin/low pressure signal. The peak pellet LHGR for the 100% power case is calculated to be 14.8 kW/ft.

15.4.2.4 Conclusion

Reactivity insertion transient calculations demonstrate that the XNB correlation limit of 1.17 will not be penetrated during any credible reactivity insertion transient at full power or mid power. The maximum peak pellet linear heat rate for these events is well below the incipient fuel centerline melt criterion of 21 kw/ft. Applicable acceptance criteria are therefore met for Cycle 8, and the adequate functioning of the thermal margin/low pressure trip demonstrated.

15.4.3 CONTROL ROD MISOPERATION

The control rod misoperation event considers a number of different event initiators. These include:

(1) Dropped control rod or bank;
(2) Dropped part-length control rod;
(3) Malpositioning of a part-length control rod group;
(4) Statically misaligned control rod or bank;
(5) Single control rod withdrawal;
(6) Core barrel failure.

Each of the above events includes a redistribution of power which leads to a local augmentation of the peaking factor in the affected region of the core.

15.4.3.1 Event Description

(1) Dropped Control Rod/Bank

A control rod drop event is initiated by a de-energized control rod drive mechanism (CRDM) or another failure in the control rod system. With the insertion of negative reactivity due to the dropped rod, the core power decreases. Moderator and Doppler temperature feedback, driven by a constant turbine generator load, cause the power to increase to its initial state. A localized increase in the radial peaking results from power redistribution due to the dropped rod. This event is a challenge to DNB limits because of radial peaking augmentation together with near full power operating conditions.

(2) Dropped Part-Length Control Rod

Part-length control rods are not used during power operation and are maintained in a withdrawn state. A failure of the rod brake mechanism could result in a part-length control rod drop.

(3) Malpositioning of a Part-Length Control Rod Group

Use of part-length control rods is not allowed during power operation. The part-length control rods are maintained in a fully withdrawn state; therefore, this event is not credible.

(4) Statically Misaligned Control Rod/Bank

A static misalignment occurs when a malfunction in the CRDM causes a control rod to be out of alignment with its bank or a control group to be in violation of its Power Dependent Insertion Limits (PDILs).

In the case of a static misalignment of a control rod, one control rod is positioned out of the core while the balance of the control bank is inserted. This situation causes a localized increase in radial peaking in the affected region of the core. The increased radial peaking, together with the initial core power level, can significantly reduce the margin to DNB. The reverse condition, i.e. one control rod fully inserted with its bank fully withdrawn, is essentially the same as a dropped control rod event.

(5) Single Control Rod Withdrawal

The withdrawal of a single control rod results in a reactivity insertion and a localized increase in radial peaking. The degradation of core conditions characteristic of a reactivity insertion transient, combined with an increase in local radial peaking, poses a challenge to DNBR limits.

(6) Core Barrel Failure

This event is initiated by the circumferential rupture of the core support barrel. The core stop supports serve to support the barrel and the reactor core by transmitting all loads directly to the vessel. The clearance between the core barrel and the supports is approximately one-half inch at operating temperatures. The worst possible axial location of the barrel rupture is at the midplane of the vessel nozzle penetrations so that a direct flow path is formed between the inlet and exit nozzles in parallel with the path that goes through the core. The core sustains a small reactivity transient induced by the motion of the core relative to the inserted rod bank(s).

Reactor protection for the Core Barrel Failure event during hot shutdown, refueling shutdown, cold shutdown, and refueling operating conditions is provided by Technical Specification Shutdown Margin requirements. For the reactor critical and hot standby operating conditions, reactor protection is provided by the variable overpower trip and a nonsafety grade high rate-of-change of power trip. For the rated power and power operating conditions, reactor protection is afforded for the variable overpower and thermal margin/low pressure trip.

15.4.3.2 Event Disposition and Justification

(1) Dropped Control Rod/Bank

The analysis supporting modified RPS operation evaluates the consequences of this event from rated power conditions[3]. A control bank drop causes a variable high power trip and, therefore, does not pose a challenge to DNB limits. The minimum DNBR for a control rod drop event from full power was analyzed for modified RPS operation.

For Cycle 8 operation, the minimum DNBR for the control rod drop event is disposed to be analyzed at rated power and full flow with increased radial peaking. The system response due to a control bank drop will not vary for Cycle 8 as compared to the analysis supporting modified RPS operation[3].

(2) Dropped Part-Length Control Rod

A dropped part-length control rod will not be as severe as a dropped full-length control rod and is, therefore, bounded by Event 15.4.3(1)[2]. This conclusion will not change for Cycle 8.

(3) Malpositioning of the Part-Length Control Rod Group

Use of part-length control rods is not allowed during power operation. The part-length control rods are maintained in a fully withdrawn state; therefore, this event is not credible.

(4) Statically Misaligned Control Rod/Bank

Reference 2 disposed the misaligned control rod event to be analyzed for modified RPS operation. The modified RPS analysis considered this event at an initial full power operating condition with one control rod fully withdrawn and its control bank inserted beyond the appropriate PDIL[3]. The modified RPS analysis consists of an XCOBRA-IIIC calculation at full power conditions with a limiting assembly radial peaking augmentation factor.

For the statically misaligned control bank at rated power, the statically misaligned control rod reaches the same steady-state conditions[2]. Therefore, the results for the Cycle 8 reanalysis of a misaligned control rod also apply to the misaligned control bank event at rated power.

For power operating conditions, control banks 3 and 4 are inserted in the core for power levels of 35% to 65% of rated. The control bank misalignment event was disposed to be reanalyzed to support modified RPS operation[2,3]. The analysis consists of XCOBRA-IIIC calculations at 50% and 65% of rated power conditions. Each calculation includes a limiting assembly radial peaking augmentation factor.

For Cycle 8 operation, the increase in radial peaking necessitates the reanalysis of minimum DNBR for both the 50% and 65% power cases with four PCP flow.

(5) <u>Single Control Rod Withdrawal</u>

This event was disposed to be analyzed for both rated power and power operating conditions[2]. The analysis performed for modified RPS operation evaluates the consequences of single rod withdrawal from both 50% and 100% rated power initial conditions. A number of reactivity insertion rates were evaluated to bound the minimum insertion rates for this event. The PTSPWR2 portion of the analysis of a single control rod withdrawal is a continuation of the respective reactivity insertion rate curves generated for Event 15.4.2[3].

For Cycle 8 operation, the increased radial peaking will impact DNBR for the 50% and 100% power cases. To assess the minimum DNBR for Cycle 8 operation, the limiting DNBR cases will be reanalyzed under Cycle 8 conditions.

(6) <u>Core Barrel Failure</u>

The probability of a circumferential rupture of the core support barrel has the same low probability of occurrence as a major rupture of the primary system piping. Therefore, this event is classified as a Limiting Fault event with the corresponding acceptance criteria. The acceptance criteria are given in Reference 3.

Reference 2 disposed this event not to be credible during hot shutdown, refueling shutdown, cold shutdown and refueling operation due to the Technical Specification shutdown margin requirements. The event initiated from rated power bounds the power operating, reactor critical and hot standby operating modes. For rated power, the FSAR analysis[8] is bounding due to a conservatively high reactivity insertion.

For the conditions assumed in the analysis supporting modified RPS operation, the maximum reactivity insertion at rated power with the control rods at their PDILs is less than the reactivity insertion for the FSAR analysis. Reference 3, therefore, disposed this event to be bounded by the FSAR analysis[8].

For Cycle 8, however, the increase in radial peaking necessitates the reanalysis of the minimum DNBR for the Core Barrel Failure event at rated power.

15.4.3.4 Analysis and Results

Calculated minimum DNBRs and peak pellet LHGRs are given in Table 15.4.3-1 for the Control Rod Misoperation events.

Radial peaking augmentation factors for dropped control rod/bank events, static misalignment events and single control rod withdrawal events are calculated at full power for different exposure conditions. The radial peaking augmentation factors used in the Reference 3 analysis were verified to remain conservatively applicable to Cycle 8.

Control rod and bank worth for Cycle 8 were verified to be bounded by the values used in the Reference 3 analysis.

Due to the motion of the core relative to the control rod positions, a small reactivity insertion is experienced for the Core Barrel Failure event. The maximum distance the core barrel may fall is 0.547 inches[8] at hot full power. A conservatively high reactivity insertion rate is used in the analysis of minimum DNBR.

The amount of coolant flow that bypasses the reactor core increases as a result of a failure of the core barrel. A parallel flow path between the inlet and exit nozzles can potentially occur. To account for the increase in core bypass flow, the total PCS flow rate is reduced by 10%[8].

The minimum DNBR for the Core Barrel Failure event is 1.25 for Cycle 8, as calculated using the XNB correlation. Therefore, because the minimum DNBR is greater than the 95/95 limit of 1.17, no fuel failures would be expected for this Limiting Fault event. Overpressurization of the primary system is bounded by the results of the Control Rod Ejection event (Event 15.4.8).

15.4.3.5 Conclusion

The moderate frequency events result in minimum DNBRs greater than the XNB critical heat flux correlation safety limit. Thus, the DNBR SAFDL is not penetrated. The maximum peak linear heat rate for these events is below the fuel centerline melt criterion of 21 kw/ft.

For the Core Barrel Failure event, the minimum DNBR is greater than the XNB critical heat flux correlation safety limit. Thus, the DNBR SAFDL is not penetrated and no fuel failures are predicted to occur.

Applicable acceptance criteria for these events are therefore met for Palisades Cycle 8 operation.

Table 15.4.3-1 Summary of MDNBRs for Control Rod Misoperation Events

Event (Power)	Operating Mode*	MDNBR	Maximum LHGR (kW/ft)
Dropped Control Rod (100%)	1	1.25	15.6
Statically Misaligned Control Rod (100%)	1	Bounded (Dropped Rod)	
Statically Misaligned Bank (50%)	2	2.79	10.0
Statically Misaligned Bank (65%)	2	2.08	12.3
Rod Withdrawal (100%)	1	1.22	15.1
Rod Withdrawal (50%)	2	1.59	13.3
Rod Withdrawal (10^{-4}%)	3	Bounded (15.4.1)	
Rod Withdrawal (10^{-4}%)	4	Bounded (15.4.1)	
Rod Withdrawal ($\leq 10^{-4}$%)	5	Subcritical	
Core Barrel Failure (100%)	1	1.25	**

*These modes are defined in Reference 3.

**The Core Barrel Failure transient is classified as a Limiting Fault event.

15.4.4 STARTUP OF AN INACTIVE LOOP

15.4.4.1 Event Description

This event is initiated by the startup of an inactive primary coolant pump. The startup of an inactive pump can lead to an introduction of colder primary coolant into the reactor core. The lower coolant temperature, together with a negative moderator temperature coefficient, can cause an increase in core power and a degradation of DNB margin. Sufficient protection is available to reduce the consequences of this event.

15.4.4.2 Event Disposition and Justification

A Startup of an Inactive Loop is classified as a Moderate Frequency event with the corresponding acceptance criteria. The acceptance criteria for this class of event are given in Reference 3.

Reference 3 disposed this event to be bounded by the FSAR analysis[8] for the analysis supporting modified RPS operation.

For operation with one inoperative pump, the low flow trip setpoint and the variable overpower trip setpoint are simultaneously changed to the allowable values for the selected pump condition. Under this arrangement, the variable overpower trip will terminate any transient resulting from the inadvertent activation of an idle pump before any significant decrease in thermal margin.

For Palisades, this event is most limiting for an initial condition of three operating primary coolant pumps with the corresponding reduced power level and variable high power trip setpoint. Continuous power operation with less than four primary coolant pumps is not allowed by the Technical Specifications. Additionally, startup of an inactive primary coolant pump when operating above hot shutdown is not allowed.

Due to the changes for Cycle 8, the DNBR will be analyzed with an increase in radial peaking for this event.

15.4.4.3 Results of Analysis

As part of the modified RPS, a variable high power trip is to be added. This trip will cause a reactor trip when the reactor power increases to a power level 10% above the current power level. This trip will provide the required protection to mitigate the consequences of an Idle Loop Startup transient. For power operation with three pumps in service, the variable high power trip setpoint has a maximum value of 49% of rated power, which is 10% above the maximum allowed operating power level of 39% of rated.

When a primary pump is removed from service, the thermal power is reduced in accordance with the Technical Specifications. Because of the reduced variable high power trip settings, the maximum nominal reactor power for three pump operation without trip is less than 49% of rated, or 39% maximum operating power level plus a 10% margin to trip. Including a trip uncertainty of ±5.5%[3], the maximum attainable power for three pump operation is 54.5% of rated without causing a reactor trip.

Although a slight temperature drop due to the startup of the inactive pump is experienced, the effect on system pressure and hot channel minimum DNBR is covered by the large power margin to full power conditions. Therefore, the consequences of this event are bounded by the nominal full power minimum DNBR with four primary coolant pump flow.

15.4.5 FLOW CONTROLLER MALFUNCTION

There are no flow controllers on the PCS at Palisades. Therefore, this event is not credible.

15.4.6 CVCS MALFUNCTION THAT RESULTS IN A DECREASE IN THE BORON CONCENTRATION IN THE REACTOR COOLANT

15.4.6.1 Event Description

A boron dilution event can occur when primary grade water is added to the primary coolant system via the Chemical Volume and Control System (CVCS) or the accidental transfer of the contents of the iodine removal system during cold shutdown or refueling shutdown conditions.

The dilution of primary system boron adds positive reactivity to the core. This event can lead to an erosion of shutdown margin for subcritical initial conditions, or a slow power excursion for at-power conditions. A boron dilution at rated or power operating conditions behaves in a manner similar to a slow uncontrolled rod withdrawal transient (Event 15.4.2).

15.4.6.2 Event Disposition and Justification

The boron dilution analysis to support modified RPS operation(3) evaluates the time to criticality caused by the dilution of the primary system boron and the subsequent loss of shutdown margin. The modified RPS analysis addresses the following modes of operation: 1) Refueling; 2) Startup; and, 3) Power operation. The modified RPS boron dilution analysis also includes a calculation to determine the time to criticality due to the failure to borate the core to compensate for reactivity changes after shutdown.

Due to changes in the initial and critical boron concentration for Cycle 8, the boron dilution event is reanalyzed for refueling, startup and failure to reborate after shutdown cases. The consequences for power operation are addressed by the reanalysis of Event 15.4.2 minimum DNBRs for Cycle 8.

15.4.6.3 <u>Results of Analysis</u>

(1) <u>Dilution During Refueling</u>

For dilution to occur during refueling by primary makeup water, it is necessary to have at least one makeup water transfer pump operating, one charging pump operating, and the makeup controller set for dilution. None of these conditions are required for refueling and would be in violation of operating procedures. Nevertheless, such a dilution incident has been analyzed as follows:

1) One shutdown cooling pump is running to remove decay heat.

2) The valve in the bleed-off water header from the primary coolant pumps is closed.

3) The makeup system is set for makeup at shutdown concentration.

4) The boron concentration of the refueling water to maintain a shutdown margin of at least 5.0%[15] with all rods out of the core. Periodic sampling insures that the concentration is maintained above the concentration corresponding to 5.0% shutdown margin.

5) Minimum primary coolant volume for reactor vessel head removal during refueling is considered (3300 ft^3). This is the volume necessary to fill the reactor vessel above the nozzles to insure cooling via the Shutdown Cooling System.

6) The charging dilution flow is assumed to be 44 gpm and the wave front/slug flow approach is utilized.

The operator has adequate indication of any significant boron dilution from the audible count rate instrumentation. High count rate is alarmed in the reactor containment and the main control room. The count rate is a measure of the effective multiplication factor.

With all rods out of the core, the boron concentration must be reduced from the refueling to the critical boron concentration before the reactor will become critical. This would take approximately 110 minutes after arrival of the first wave front. This is ample time for the operator to recognize the audible high count rate signal and isolate the reactor makeup water source by closing valves and/or stopping the primary makeup water transfer pumps.

(2) Dilution During Startup

After refueling and prior to hot standby, the primary coolant system may contain water having the boron concentration corresponding to shutdown margin of 2% $\Delta\rho$. The maximum possible rate of introduction of unborated demineralized water is 133 gpm. The volume of reactor coolant is about 8,628 ft^3, which is the total volume of the primary coolant system with 29.3% steam generator tube plugging, excluding the pressurizer. The primary coolant pumps are assumed to be running (i.e., perfect mixing is assumed).

Under these conditions the minimum time required to reduce the reactor coolant boron concentration to the critical concentration is about 44 minutes. Boron dilution for start-up will be performed under strict procedures and administrative controls.

During dilution at hot standby or reactor critical, the operating staff will be monitoring the nuclear instruments and the boronometer readings. An abnormal change in the reading of these instruments will inform the operator that dilution is occurring. The operator will have further indication of the process from volume control tank level and from operation of the letdown diverter valve. Further, should the makeup controller fail to close the makeup stop valve, the operator has visual indication of makeup water flow and of makeup water transfer pump operation.

In any case, should continued dilution occur, the reactivity insertion rate would be less than that considered for uncontrolled rod/rod bank withdrawals. The reactor protection provided for the rod withdrawal incident will also provide protection for the boron dilution incident.

When the primary system boron concentration is being changed, at least one shutdown cooling pump or one primary coolant pump must be functioning to provide sufficient heat removal capacity. Under the condition of one operating shutdown cooling pump, imperfect mixing is conceivable. With imperfect mixing, a shutdown cooling pump flow greater than or equal to 2810 gpm is required to ensure that the acceptance criteria for this event is not violated for 2% $\Delta\rho$. Alternatively, a minimum shutdown cooling flow of 1500 gpm will not violate the event acceptance criteria for a shutdown margin of at least 3.5% $\Delta\rho$. These values were calculated by evaluating the minimum shutdown cooling pump flow rate necessary to bring the plant to a critical state in at least 15 minutes[1], assuming a maximum charging flow rate of 133 gpm and a reactor coolant volume of about 8628 ft^3.

(3) <u>Dilution During Power Operation</u>

Inadvertent injection of primary makeup water into the primary coolant system while the reactor is at power would result in a reactivity addition initially causing a slow rise in power, temperature and possibly pressure. Assuming that unborated water is injected at the maximum possible rate of 133 gpm, the rate of reactivity addition would be about 6×10^{-6} $\Delta\rho$/s. This is much slower than the maximum rate possible with a rod withdrawal.

Continued boron dilution after reactor trip, if the operator takes no corrective action, is addressed in Reference 3. The assumptions used in the Reference 3 analysis bound Cycle 8 operation.

(4) Failure to Add Boron To Compensate for Reactivity Changes After Shutdown

The analysis of the boron dilution event for this case is presented in Reference 3. The assumptions employed in the Reference 3 analysis remain valid for Cycle 8 operation.

15.4.6.4 Conclusion

The results of the analysis for this event are summarized in Table 15.4.6-1. The results show that there is adequate time for the operator to manually terminate the source of dilution flow. The operator can then initiate reboration to recover the shutdown margin. Boron dilution during power operation is bounded by the analyses presented in Sections 15.4.1 and 15.4.2. However, the results presented here demonstrate that there is adequate time for the operator to manually terminate the source of dilution flow following reactor trip.

Table 15.4.6-1 Summary of Results for the Boron Dilution Event

Reactor Conditions	Dilution By	Time to Criticality
Refueling	Primary Water	110 minutes (Charging at 44 gpm)
Refueling and Startup with Primary Coolant System Filled	Primary Water	44 minutes (Charging at 133 gpm, main reactor coolant pumps running)
Refueling and Startup with Primary Coolant System Filled	Primary Water	>15 minutes*
Hot Standby or Critical	Primary Water	Considered in the uncontrolled rod/rod bank withdrawal analysis
Following a trip from the Power Operation Condition		Bounded by Ref. 3
Failure to add boron to compensate for Reactivity changes after Shutdown		Bounded by Ref. 3

* Charging flow is 133 gpm and RHR flow ≥2810 gpm with ≥2% $\Delta\rho$ shutdown margin <u>or</u> RHR flow ≥1500 gpm with ≥3.5% $\Delta\rho$ shutdown margin.

15.4.7 INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN AN IMPROPER POSITION

15.4.7.1 Event Description

An inadvertent loading of a fuel assembly in an improper position can result in an alteration of the power distribution in the core which can adversely affect thermal margin.

15.4.7.2 Event Disposition and Justification

The event is disposed as bounded for modified RPS operation due to the administrative controls and procedures that ensure a properly loaded core[2]. The changes for Cycle 8 will not invalidate this disposition; consequently, this event will not require analysis.

15.4.8 SPECTRUM OF CONTROL ROD EJECTION ACCIDENTS

15.4.8.1 Event Description

This event is initiated by a failure in the CRDM pressure housing causing a rapid ejection of the affected control rod. The ejection of the control rod inserts positive reactivity causing an increase in core power. Because of the increase in core power, this event challenges both DNBR and overpressurization acceptance criteria.

15.4.8.2 Event Disposition and Justification

The minimum DNBR and pressurization consequences of a control rod ejection event were analyzed for the analysis supporting modified RPS operation[3]. The HFP case was determined to be most challenging to the acceptance criteria.

For Cycle 8, the system response to an ejected control rod will not change from that for the modified RPS analysis. Therefore, the pressurization results for the modified RPS analysis are applicable to Cycle 8.

The fuel failure evaluation must be reanalyzed for Cycle 8 using cycle specific post-ejection radial peaking factors.

15.4.8.3 Analysis and Results

The minimum DNB case is initiated by the rapid insertion of positive reactivity due to the ejection of a control rod. A minimum DNBR less than 1.17 is calculated to occur for this event.

With the core boundary conditions predicted at the time of minimum DNBR, along with an asymmetric core power distribution, the amount of fuel failure is calculated. In Reference 3, it was determined that 12.2% of the fuel rods in the core will fail due to the penetration of DNB. Due to conservative assumptions employed in the Reference 3 analysis, the amount of fuel that is predicted to fail for Cycle 8 is less than 12.2%. The offsite radiological doses for this event were calculated in Reference 3 to be below the 10 CFR 100 dose limits for 12.2% fuel failure.

15.4.8.4 Conclusion

The radiological doses are conservatively calculated to be less than the 10 CFR 100 dose limits. Applicable acceptance criteria are considered, therefore, to be met for Cycle 8.

15.4.9 SPECTRUM OF ROD DROP ACCIDENTS (BWR)

This event is not applicable to Palisades since it is not a BWR.

15.5 INCREASES IN REACTOR COOLANT SYSTEM INVENTORY

15.5.1 INADVERTENT OPERATION OF THE ECCS THAT INCREASES REACTOR COOLANT INVENTORY

15.5.1.1 Event Description

This event is caused by an inadvertent actuation of the ECCS that results in an increase in the primary system inventory. The primary challenge is to the primary system overpressurization criteria. For the case where the primary system boron concentration is reduced as a result of ECCS actuation, Event 15.4.6 is bounding.

15.5.1.2 Event Disposition and Justification

This event was disposed to be bounded by Events 15.4.6 and 15.2.1 for the analysis supporting modified RPS operation[2]. The event initiators and significant parameters remain unchanged for Cycle 8 operation as compared to the modified RPS analysis[2,3]. Therefore, the event is not analyzed for Cycle 8.

15.5.2 CVCS MALFUNCTION THAT INCREASES REACTOR COOLANT INVENTORY

15.5.2.1 Event Description

A malfunction in the CVCS could result in the inadvertent operation of the charging system pumps. If the letdown system is not operating, the result leads to an increase in the primary system coolant inventory and, potentially, an overpressurization of the primary system and/or a dilution of the primary system boron concentration.

15.5.2.2 Event Disposition and Justification

Sufficient relief capacity exists to limit the overpressurization potential to less than the 110% design value of 2750 psia. The potential for dilution of the primary system boron is addressed in Event 15.4.6.

Reference 2 disposed this event as being bounded by Events 15.4.6 and 15.2.1 for modified RPS operation. The event initiators and significant parameters remain unchanged for Cycle 8 operation. Therefore, the event is not analyzed for Cycle 8.

15.6 DECREASES IN REACTOR COOLANT INVENTORY

15.6.1 INADVERTENT OPENING OF A PWR PRESSURIZER PRESSURE RELIEF VALVE

15.6.1.1 Event Description

An inadvertent opening of a pressurizer pressure relief valve or safety valve causes a decrease in the primary system pressure resulting in a loss of both thermal margin and primary coolant inventory.

The pressurizer relief valves at Palisades are blocked closed during power operation by downstream isolation valves. Therefore, an inadvertent opening of a relief valve will not result in a loss of primary coolant inventory. For a stuck open safety valve after a transient, the loss of coolant accident (LOCA) mitigating procedures will begin.

15.6.1.2 Event Disposition and Justification

Reference 2 disposed this event as not being credible for Modes 1-5. For a stuck open safety valve after a transient, the event is bounded by the small break LOCA (Event 15.6.5). Changes for Cycle 8 operation will not change this disposition.

15.6.2 RADIOLOGICAL CONSEQUENCES OF THE FAILURE OF SMALL LINES CARRYING PRIMARY COOLANT OUTSIDE OF CONTAINMENT

15.6.2.1 Event Description

This event occurs when a small line carrying primary coolant outside of containment ruptures leading to a depletion of primary system coolant and a release of contaminated liquid. The charging and HPSI systems provide sufficient coolant to replenish that which is lost. Consequently, no fuel failures would be predicted assuming a reactor trip on low pressurizer pressure, TM/LP or Safety Injection Signal (SIS). The radiological consequences are limited by the maximum primary coolant activity level allowed by the Technical Specifications.

15.6.2.2 Event Disposition and Justification

Reference 2 disposed this event as being bounded by the small break LOCA (Event 15.6.5). Changes for Cycle 8 operation will not change this disposition.

15.6.3 RADIOLOGICAL CONSEQUENCES OF STEAM GENERATOR TUBE FAILURE

15.6.3.1 Event Description

This incident occurs when a steam generator tube fails causing a leakage of coolant from the primary system to the secondary system. The leakage results in a depletion of primary coolant, a reduction of primary system pressure and a release of fission products to the main steam system. The consequences of this event are maximized for a rated power initial condition due to the amount of stored energy and decay heat that must be removed prior to bringing the two systems to an equilibrium pressure state.

15.6.3.2 Event Disposition and Justification

The FSAR analysis was performed at a reactor power level of 2650 MWt and a primary system pressure of 2100 psia[8]. For a complete severance of one steam generator tube with a subsequent leakage rate greater than the capacity of the charging pumps, the reactor would trip on a low pressurizer (TM/LP) pressure signal of 1750 psia. The TM/LP trip acts to protect against significant fuel damage in this event. The dose calculations in the FSAR analysis were performed with a source term based on 1% fuel rod failure[8].

For Cycle 8, the core power is 2530 MWt with a 3.5% increase in radial peaking limits relative to previous cycles. The Cycle 8 core power is about 4.5% less than the FSAR analysis while the radial peaking factor is 3.5% higher. For the same assembly exposure and 1% fuel rod failure, the primary coolant activity for the FSAR analysis is about 1% higher than would be the case for Cycle 8. Therefore, the amount of radioactive fission products that leak from the primary to the secondary system is greater for the FSAR assumptions.

After the reactor has tripped, the decay heat and stored energy in the core is removed via the atmospheric dump valves and steam bypass. For the modified RPS analysis and Cycle 8 operation, the reactor power is 2530 MWt and the pressurizer pressure is 2060 psia, as compared to 2650 MWt and 2100 psia for the FSAR analysis. The time required to remove the primary system energy for a power level of 2530 MWt is less than that for 2650 MWt. Therefore, for Cycle 8 operation, the secondary system steam valves are open for a shorter period of time resulting in a smaller radioactive release to the atmosphere.

Reference 2 disposed this event as being bounded by the FSAR analysis[8]. This disposition will not change for Cycle 8.

15.6.4 RADIOLOGICAL CONSEQUENCES OF A MAIN STEAM LINE FAILURE OUTSIDE CONTAINMENT (BWR)

This event pertains to BWRs and is, therefore, not applicable to Palisades.

15.6.5 LOSS OF COOLANT ACCIDENTS RESULTING FROM A SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

15.6.5.1 Event Description

This event is initiated by a breach in the primary system pressure boundary. The event initiators vary from relatively small breaks for small break LOCAs (SBLOCA) to complete ruptures of the PCS piping for large break LOCAs (LBLOCA). The primary concerns of LBLOCA and SBLOCA analyses are the peak clad temperature (PCT) and, the amount of localized and core-wide metal-water reaction.

15.6.5.2 Event Disposition and Justification

ANF has performed a LBLOCA analysis for Palisades which supports operation with the radial peaking limits given in Reference 15. The results of this analysis are provided in Reference 8. According to Reference 8, the LBLOCA results are more limiting than the SBLOCA results.

For Cycle 8, the LBLOCA is disposed to be analyzed to show that the increased radial peaking does not result in a violation of 10 CFR 50.46(b) acceptance criteria. For Cycle 8, the radial peaking factors will increase by 3.5%. The changes to the Cycle 8 core will not cause the SBLOCA to become more limiting than the LBLOCA. Therefore, a LBLOCA analysis for Cycle 8 operation with increased radial peaking limits will bound the consequences of a SBLOCA.

15.6.5.3 Analysis and Results

The analysis and results of the LBLOCA performed for Palisades Cycle 8 are documented in Reference 10.

15.7 RADIOACTIVE RELEASES FROM A SUBSYSTEM OR COMPONENT

15.7.1 WASTE GAS SYSTEM FAILURE

15.7.2 RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE (RELEASE TO ATMOSPHERE)

15.7.3 POSTULATED RADIOACTIVE RELEASES DUE TO LIQUID-CONTAINING TANK FAILURES

The results of the three events above are not dependent on either fuel type, steam generator tube plugging, reactor coolant flow rate, reactor coolant inlet temperature, or reactor protection system modifications. The reference analysis is therefore not affected by the current licensing action and remains the bounding analysis for this event. The reference analysis is provided in the Updated Palisades FSAR, Reference 8.

15.7.4 RADIOLOGICAL CONSEQUENCES OF FUEL HANDLING ACCIDENT

15.7.4.1 Event Description

A fuel handling accident occurs when a fuel assembly is damaged during refueling operations such that fuel rods are ruptured resulting in a release of radioactivity. The inventory of radioactive fission products is determined by the exposure and power level of the assemblies or fuel rods.

15.7.4.2 Event Disposition and Justification

The FSAR analysis assumes that the affected assembly is resident in the core for three full power years with a power of 2650 MWt and a peak rod radial peaking factor of 1.65[8]. The effective power level of the peak assembly is about 21.4 MWt. The fission product inventory for the assembly is conservatively calculated based on the fission products contained in the peak powered fuel rod.

For Cycle 8 operation, the core power is 2530 MWt and the peak rod radial peaking factor is increased 3.5%. For this peaking, the effective peak assembly power is about 21.4 MWt. The effective assembly powers for both the reference analysis[8] and Cycle 8 are essentially the same. For the given assembly exposure, the amount of fission products will be the same for the Cycle 8 conditions as compared to the FSAR conditions. Therefore, the consequences of a fuel handling accident for Cycle 8 are addressed by the FSAR analysis[8].

15.7.5 SPENT FUEL CASK DROP ACCIDENTS

15.7.5.1 Event Description

A spent fuel cask drop accident can result in the damage of an irradiated fuel assembly and the subsequent release of radioactivity.

15.7.5.2 Event Disposition and Justification

Reference 8 contains an analysis of the radiological consequences of this event. The FSAR analysis conservatively assumes that the assembly with the maximum exposure is damaged. A radial peaking factor of 2.0 is applied to this assembly.

The disposition of this event for the analysis supporting the modified RPS operation states that the FSAR analysis is bounding. The peaking factor used in the FSAR analysis bounds that for Cycle 8. Therefore, the FSAR analysis bounds the consequences for Cycle 8 operation.

4.0 THERMAL-HYDRAULIC COMPATIBILITY

This section describes the thermal-hydraulic analyses performed in support of the following for Palisades Cycle 8:

(1). Insertion of four ANF lead assemblies with high thermal performance (HTP) spacers. The HTP spacer lead assemblies are each composed of 216 fuel rods.

(2). For Cycle 8, 16 assemblies will be inserted along the core periphery to reduce neutron fluence on critical vessel welds. The outer four rows of rods (56 rod locations) along one side of each of these shielding assemblies will be replaced with stainless steel rods.

The purpose of the analyses is to demonstrate hydraulic compatibility of the these assemblies with the existing Palisades core. Discussed in this Section are analyses of the affect of the ANF lead assemblies and stainless steel assemblies on the minimum departure from nucleate boiling ratio (DNBR) for the Palisades core. The lead assemblies and reconstituted stainless steel assemblies will have no adverse impact on LOCA/ECCS performance.

4.1 Thermal Hydraulic Design Criteria

The primary thermal hydraulic design criteria for ANF reload fuel assure that fuel rod integrity is maintained during normal operation and Anticipated Operational Occurrences (AOOs). Specific criteria are:

(1). Avoidance of DNB for the limiting rod in the core with 95% probability at a 95% confidence level.

(2). Fuel centerline temperatures remain below the melting point of the fuel pellets.

Observance of these criteria is considered conservative relative to the requirement that AOOs not result in fuel rod failures or loss of functional capability.

4.2 Summary of Results

Results of minimum DNBR calculations performed to support ANF HTP spacer lead assemblies in the Palisades reactor for Cycle 8 show that the XNB 95/95 limit of 1.17 is not violated for a limiting AOO event. Likewise, for a limiting assembly adjacent to a stainless steel shielding assembly, the minimum DNBR is well above the XNB 95/95 limit of 1.17. The minimum DNBR performance of the core during AOOs thus accords with the thermal hydraulic design criterion on DNBR.

The thermal hydraulic simulations employed to evaluate minimum DNBR were performed in accordance with ANF's NRC-approved thermal hydraulics methodology for mixed cores(16). The 2% mixed core penalty of minimum DNBR has not been assessed in these calculations because the lead assemblies do not represent a significant fraction of the core.

For standard ANF fuel assemblies, fuel centerline temperatures have been shown in the Chapter 15 event analysis of AOOs to be less than the limit for incipient melt of 21 kW/ft. The centerline temperatures for the lead assemblies and shielding assemblies will also be less than this limit.

These results adequately demonstrate the thermal-hydraulic compatibility of the HTP spacer lead assemblies and stainless steel shielding assemblies with the co-resident ANF standard fuel at Palisades. Thermal-hydraulic design criteria are met for these fuel types.

4.3 Analysis and Results

The thermal-hydraulic analysis for the lead assemblies with HTP spacers and the stainless steel assemblies will be discussed in the following two sections.

4.3.1 Lead Assemblies with HTP Spacers

The spacer loss coefficients for the ANF standard fuel are derived from pressure drop tests performed in ANF's portable loop hydraulic test facility(19). The HTP spacer loss coefficient is also based on pressure drop test data from ANF's portable loop hydraulic test facility. The ANF standard assembly has ten bi-metallic spacers. The ANF HTP spacer assembly modelled has ten HTP spacers. The loss coefficients for the other assembly components (i.e., upper and lower tie plates) are identical for both the lead and standard fuel designs.

The overall assembly loss coefficient for an ANF lead assembly exceeds that of the ANF standard fuel by about 10%. A full core of ANF fuel with HTP spacers would slightly decrease the total vessel flow relative to the current Palisades core, due to the greater hydraulic resistance of the HTP spacers.

The core flow distribution (CFD) analysis is performed to assess crossflow between assemblies in the core for use in subsequent minimum DNBR subchannel analyses. The core flow distribution analysis is particularly important for mixed fuel loadings where hydraulically different fuel types are co-resident in the core. The result of the CFD analysis is a set of axially varying boundary conditions on heat, mass, and momentum fluxes through the vertical boundaries of the assemblies of interest. These boundary conditions are employed in the subsequent 1/8th assembly simulations in which minimum DNBR is computed.

In the analysis each fuel assembly in an octant of the Palisades core is modeled as a hydraulic channel. The calculations are performed with the XCOBRA-IIIC computer code[6]. Crossflow between adjacent assemblies in the open lattice core is directly modeled. The single-phase loss coefficients are used in the analyses to hydraulically characterize the assemblies in a mixed core.

The core flow and subchannel calculations are performed at conditions representative of the dropped rod AOO for Palisades Cycle 8. The lowest DNBR for a dropped rod event is calculated at full power with a nominal pressure of 2200 psia and a flow of 130 Mlbm/hr, as allowed by the T_{inlet} LCO. For the standard fuel assembly design the minimum DNBR under these conditions is calculated to be 1.22.

The radial peaking factor for the lead assembly was set equal to the proposed increased Technical Specification limit of 1.73 for a 216 rod assembly. The limiting standard fuel design is a 208 rod assembly. A 5% inlet flow maldistribution is assumed for the limiting assembly and surrounding assemblies. The axial power distribution employed in the calculations is the limiting full power axial with an ASI of -0.139.

To establish the limiting assembly boundary conditions for the subsequent minimum DNBR analyses, two separate calculations were made. These calculations provide heat, mass and momentum flux boundary conditions as a function of axial position for the following cases:

(1) Limiting ANF HTP spacer lead assembly loaded in an interior location.

(2) Limiting ANF HTP spacer lead assembly loaded on the core periphery.

Boundary conditions from these cases were passed to the 1/8 assembly analysis for the minimum DNBR calculations.

In the 1/8 assembly simulation, the XCOBRA-IIIC computer code is employed to evaluate the pertinent thermal hydraulic variables in the inter-rod flow channels of the fuel assembly of interest. Heat, mass, and momentum fluxes between the inter-rod flow channels are explicitly calculated. Local values of mass velocity and enthalpy are determined, and used to calculate the DNBR via the XNB critical heat flux correlation[17,18]. Axially varying boundary conditions on the vertical boundaries of the assembly are obtained from the appropriate CFD calculation, discussed above.

The calculations include factors to account for manufacturing tolerances and densification effects. Specifically, a 3% engineering factor is applied to the limiting rod power to account for fabrication tolerances on pellet diameter, density, enrichment and cladding diameter. These manufacturing tolerances potentially affect heat flux at the limiting DNBR location in the assembly.

The XNB DNB correlation is demonstrated to be applicable to the ANF standard fuel assemblies in Reference 18. The ANF HTP spacer is specifically designed to yield improved DNB performance relative to the ANF standard spacer. Flow mixing data for the similar 17x17 HTP spacer design demonstrate significantly improved mixing relative to the ANF standard spacer, supporting the expectation of improved DNB performance. The XNB correlation may be conservatively applied to the ANF HTP spacer lead assemblies in this analysis.

For Case 1, a minimum DNBR of 1.18 is conservatively calculated for the ANF HTP lead assembly. For Case 2, a minimum DNBR of 1.28 is calculated. Because of the higher spacer loss coefficient for the HTP lead assembly, flow is diverted from these assemblies to surrounding assemblies with standard spacers. Consequently, local mass velocity decreases and local enthalpy increases yielding a lower DNBR (about 3%) relative to a standard ANF design. DNBR benefit due to increased mixing in the HTP spacer assemblies has been conservatively neglected for this analysis.

With the lead assembly loaded on the core periphery, Case 2, less flow is diverted to adjacent assemblies due to the proximity of the core baffle plate. Because less flow is diverted from an assembly loaded on the core periphery, as compared to the flow diversion of an interior assembly, the minimum DNBR conditions are less severe. Therefore, the minimum DNBR for Case 2 is about 8% higher than that for Case 1 with the lead assemblies loaded in interior locations.

The results of this analysis show that the calculated minimum DNBRs for HTP spacer lead assemblies in the Palisades reactor meet the 95/95 DNBR limit for the limiting AOO transient event for Cycle 8. Therefore, safety margin is not compromised for the Palisades Cycle 8 core with four HTP spacer lead assemblies.

4.3.2 Stainless Steel Shielding Assemblies

The shielding assemblies will be loaded along the core periphery to reduce the neutron fluence on critical vessel welds. Because the shielding assemblies are previously burnt assemblies reconstituted with stainless steel rods, the assembly power level will be substantially lower than the surrounding conventional fuel assemblies. Higher powered assemblies adjacent to the shielding assemblies may potentially experience an increase in crossflow due to the thermal differences between the two fuel types. This increase in crossflow could adversely impact minimum DNBR in the affected assemblies.

To assess the impact to minimum DNBR for Cycle 8, a thermal-hydraulic analysis was performed. The details of the analysis are similar to those discussed above for the HTP spacer lead assemblies.

The core flow and subchannel calculations were performed using XCOBRA-IIIC. The core flow model consists of an octant of the Palisades Cycle 8 core with each assembly modelled as a hydraulic channel. The hydraulic characteristics of the shielding assemblies are similar to those for the standard fuel design. The assembly design parameters for the stainless steel assemblies are given in Table 4-1.

The core conditions used in this analysis are the same as those used in the HTP spacer calculations. The radial peaking factor of an assembly adjacent to a stainless steel shielding assembly was increased to the Technical Specification limit for that fuel type. Axially varying crossflow boundary conditions for the limiting assembly are generated by the core flow calculation.

Using the crossflow boundary conditions from the core flow calculation in the $1/8^{th}$ assembly subchannel model, the thermal-hydraulic conditions in the limiting subchannel are evaluated. These conditions in conjunction with the XNB critical heat flux correlation yields a minimum DNBR.

The minimum DNBR for an assembly located adjacent to a shielding assembly is 1.33 which is well above the XNB 95/95 correlation limit of 1.17. The minimum DNBR for a standard fuel assembly under these conditions is 1.22. This result indicates that the presence of stainless steel shielding assemblies will not impact thermal margin for Cycle 8.

Because of the relatively low assembly power level, the stainless steel shielding assemblies will not penetrate minimum DNBR limits.

Table 4-1 Fuel Design Parameters for the Stainless Steel Shielding Assemblies

Fuel Parameters	
Fuel Rod OD	0.417 inches
Stainless Steel Rod OD	0.437 inches
Guide Tube OD	1.115 inches
Rod Array	15x15
Rod Pitch	0.55 inches
Number of Fuel Rod Positions/ Assembly	152
Number of Stainless Steel Rod Positions/Assembly	56
Number of Guide Bars	8
Number of Guide Tubes	8
Number of Instrument Tubes	1

5.0 REFERENCES

1. "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, LWR Edition, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, July 1981.

2. "Palisades Modified Reactor Protection System Report - Disposition of Standard Review Plan Chapter 15 Events," ANF-87-150(NP), Vol. 1, Advanced Nuclear Fuels Company, June 1988.

3. ANF-87-150(NP), Volume 2, "Palisades Modified Reactor Protection System Report: Analysis of Chapter 15 Events", Advanced Nuclear Fuels Company, June 1988.

4. E. Daniel Hughes, "A Correlation of Rod Bundle Critical Heat Flux for Water in the Pressure Range 150 to 725 psia", IN-1412 (TID-4500), Idaho Nuclear Corporation, July 1970.

5. "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTS-PWR), XN-NF-74-5(A), Rev. 2, Exxon Nuclear Company, October 1986, and Supplements 3-6.

6. "XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady-State and Transient Core Operation", XN-NF-75-21(A), Revision 2, Exxon Nuclear Company, January 1986.

7. XN-NF-86-91(P), "Low Flow Trip Setpoint and Thermal Margin Analysis for Three Primary Coolant Pump Operation of the Palisades Reactor", Exxon Nuclear Company, November 1986.

8. Palisades Final Safety Analysis Report, Updated Version (as of July 1985), Consumers Power Company.

9. "Advanced Nuclear Fuels Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," ANF-84-73(P), Rev. 3, Advanced Nuclear Fuels Company, May 1988.

10. "Palisades LOCA-ECCS Analysis for 2530 MWt Operation with Increased Radial Peaking and 29.3% Steam Generator Tube Plugging", ANF-88-107, Advanced Nuclear Fuels Company, August 1988.

11. "Plant Transient Analysis of the Palisades Reactor for Operation at 2530 MWt", XN-NF-77-18, Exxon Nuclear Company, July 1977.

12. "Systematic Evaluation Program Design Basis Transient Reanalysis for the Palisades Reactor", XN-NF-81-25, Revision 1, Exxon Nuclear Company, May 1981.

13. "Palisades Cycle 5 Reload Fuel Safety Analysis Report," XN-NF-81-34(P), Supp. 1, Exxon Nuclear Company, December 1981.

14. "Palisades Primary Design Parameters for PTS Analysis, XN-NF-85-101(P), Exxon Nuclear Company, October 1985.

15. Palisades Plant Technical Specifications, Consumers Power Company, Appendix A to License No. DPR-20, Revision dated October 28, 1987.

16. "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," XN-NF-82-21(A), Rev. 1, Exxon Nuclear Company, September 1983.

17. "Exxon Nuclear DNB Correlation for PWR Fuel Design," XN-NF-621(A), Revision 1, Exxon Nuclear Company, April 1982.

18. "Justification of XNB Correlation for Palisades", XN-NF-709, Exxon Nuclear Company, May 1983.

19. "Single Phase Hydraulic Performance of Exxon Nuclear and Combustion Engineering Palisades Fuel Assemblies", XN-76-1, Revision 0, Exxon Nuclear Company, January 1976.

20. Letter, D.J. VandeWalle (CPCo) to Director, Nuclear Reactor Regulation, dated June 20, 1985, Docket 50-255, License DPR-20, Palisades Plant, "NUREG-0737 Item II.K.3.30 Small Break LOCA Models and Item II.K.3.31 Plant Specific Analysis".

21. Letter, John A. Zwolinski (NRC) to Mr. VandeWalle (CPCo), dated July 3, 1985, Docket 50-255, License DPR-20, Palisades Plant, "TMI Action Plan Item II.K.3.30, Small Break LOCA Analysis and II.K.3.31, Plant Specific Analysis".

ANF-88-108

Issue Date: 8/2/88

PALISADES CYCLE 8: DISPOSITION AND ANALYSIS OF STANDARD REVIEW PLAN CHAPTER 15 EVENTS

Distribution

RA Copeland
RC Gottula
JS Holm
JW Hulsman
JD Kahn
TR Lindquist
LA Nielsen
LD O'Dell
FB Skogen
HE Williamson

CPCo/HG Shaw (20)

Document Control (3)

ERRATA SHEET FOR ANF-88-108 DATED AUGUST 1988

1. Section 1.0, Page 1, a fourth change is being implemented for Cycle 8 in that 16 assemblies having 6 w/o Gd_2O_3 and 12 assemblies having 4 w/o Gd_2O_3 are being introduced instead of the normal 20 assemblies with 4 w/o Gd_2O_3.

2. Table 2-1, 15.1.4, "Relief of Safety Valve" should be "Relief or Safety Valve"

3. Table 2-1, 15.2.7, under disposition, should say "Short term bounded" by "15.3.1" and "long term bounded" by "Ref 3".

4. Table 2-1, line 15.4.3(5), "Ref 8" should be deleted under Bounding Event or reference column.

5. Page 23, definitions should be added for the variables P_{Var}, QA, QR_1, and Q.

6. Section 15.1.3.3, second line, "flow increase of about 112%" should be "flow increase to about 112%".

7. Page 28, second paragraph, first sentence, should begin "For the Hot Zero Power Case" instead of "For the hot shutdown case".

8. On page 37, first paragraph, "reference 2" should be "reference 3".

9. Page 54, The last sentence starting with "additionally, startup of..." should be deleted.

10. On page 58, the first paragraph should be revised to indicate that "during fuel moves the operator has adequate indication of any significant boron dilution from the audible count rate instrumentation. High count rate is indicated by an increased Tick frequency in the reactor containment and the main control room."

11. On page 58, last paragraph, first sentence, delete the last four words "and the boronometer readings."

12. Page 59, third paragraph should be modified to indicate that event acceptance criteria are also met with the assumption of a 3.5% shutdown margin, shutdown cooling flow of at least 650 gpm and no more than one charging pump operating.

13. Page 65, section 15.6.1.1, in the first sentence of the second paragraph the word "downstream" should be deleted.

14. Page 75, second full paragraph on the second and third lines the words "dropped rod" should be "rod withdrawal".

15. Table 4-1, Guide tube O.D. should be 0.417 inches.