2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

The MCPR for ATRIUM-9X fuel shall be ≥ ~~1.13~~ 1.11 for two recirculation loop operation or ≥ ~~1.14~~ 1.10 for single recirculation loop operation. The MCPR for the ABB SVEA-96 fuel shall be ≥ ~~1.07~~ 1.10 for two recirculation loop operation or ≥ ~~1.09~~ 1.12 for single recirculation loop operation. ~~The MCPR limits for the ATRIUM-9X fuel are applicable to Cycle 14.~~

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1. ANF-1125(P)(A), and Supplements 1 and 2, "ANFB Critical Power Correlation," April 1990;

2. ANF-NF-524(P)(A), Revision 2 and Supplements 1 and 2, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors," November 1990;

3. ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, "Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels Corporation 9x9-IX and 9x9-9X BWR Reload Fuel," October 1991;

4. XN-NF-81-22(P)(A), "Generic Statistical Uncertainty Analysis Methodology," November 1983;

5. NEDE-23785-1-PA, Revision 1, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume III, SAFER/GESTR Application Methodology," October 1984;

6. NEDO-20566A, "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K," September 1986;

7. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," July 1996; and

8. WPPSS-FTS-131(A), Revision 1, "Applications Topical Report for BWR Design and Analysis," March 1996.

9.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

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Addi

ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANF-1125(P)(A), Supplement 1, Appendix D, Siemens Power Corporation - Nuclear Division, July 1998.

BASES

BACKGROUND (continued)

reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of Emergency Core Cooling System initiation setpoints higher than this safety limit provides margin such that the safety limit will not be reached or exceeded.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that a MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR limit.

2.1.1.1 Fuel Cladding Integrity

Refs. 2 and 7

The use of the ANFB correlation is valid for critical power calculations at pressures > 600 psia and < 1500 psia and bundle mass fluxes > 0.1 x 10^6 lb/hr-ft^2 and < 1.5 x 10^6 lb/hr-ft^2 ~~(Ref. 2)~~. The use of the XL-S96 correlation is valid for critical power calculations at pressures > 392 psia and < 1262 psia and bundle mass fluxes > 0.25 x 10^6 lb/hr-ft^2 and < 1.55 x 10^6 lb/hr-ft^2 (Ref. 3). For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

> Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to ensure a minimum

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

bundle flow for all fuel assemblies that have a relatively high power and potentially can approach a critical heat flux condition. The minimum bundle flow is > 28 x 10^3 lb/hr. The coolant minimum bundle flow and maximum flow area are such that the mass flux is > 0.25 x 10^6 lb/hr-ft^2. Full scale critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25 x 10^6 lb/hr-ft^2 is approximately 3.35 MWt. At 25% RTP, a bundle power of approximately 3.35 Mwt corresponds to a bundle radial peaking factor of > 2.9, which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 25% RTP for reactor pressures < 785 psig is conservative.

2.1.1.2 MCPR

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty inherent in the critical power correlations. Delete ~~Reference 7 describes the interim use of increased ANFB additive constant uncertainty for the SPC ATRIUM-9X fuel during Cycle 14.~~ Reference 4 describes the methodology used in determining the MCPR SL for Siemens Power Corporation fuel. Reference 5 describes the methodology used in determining the MCPR SL for ABB CENO fuel.

The critical power correlations are based on a significant body of practical test data, providing a high degree of assurance that the critical power, as evaluated by the correlation, is within a small percentage of the actual critical power. As long as the core pressure and flow are within the range of validity of the critical power correlations, the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat

(continued)

BASES (continued)

APPLICABILITY	SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.
SAFETY LIMIT VIOLATIONS	Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 6). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. ANF-1125(P)(A), Revision 0, including Supplements 1 and 2, April 1990.
3. UR-89-210-P-A, "SVEA-96 Critical Power Experiments on a Full Scale 24-Rod Sub-Bundle," October 1993.
4. ANF-524(P)(A), Revision 2, including Supplements 1 and 2, November 1990.
5. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," July 1996.
6. 10 CFR 100.
7. ~~Letter HDC:97:033 dated April 18, 1997, HD Curet (Siemens) to US NRC Document Control Deck, Interim Use of Increased ANFB Additive Constant Uncertainty.~~

Delete

Add

ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANF-1125(P)(A), Supplement 1, Appendix D, Siemens Power Corporation - Nuclear Division, July 1998.

Evaluation of Significant Hazards Considerations

Summary of Proposed Change:

A license amendment is being requested to change SLMCPR values (Technical Specifications Section 2.1.1.2):

- SPC ATRIUM-9X SLMCPR shall be 1.10 during two loop operation and 1.11 during single loop operation by recalculation of the two loop and single loop SPC ATRIUM-9X SLMCPRs using the NRC approved additive constant uncertainty for ATRIUM-9X fuel of 0.0201 (References 4 & 5); and

- ABB SVEA-96 SLMCPR shall be 1.10 during two loop operation and 1.12 during single loop operation as calculated using NRC approved methodology documented in Reference 3.

No Significant Hazards Determination:

Washington Public Power Supply System has evaluated the proposed changes to the Technical Specifications using the criteria established in 10CFR50.92(c) and has determined that they do not represent a significant hazards consideration as described below.

The operation of WNP-2 in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The probability of an evaluated accident is derived from the probabilities of the individual precursors to that accident. The consequences of an evaluated accident are determined by the operability of plant systems designed to mitigate those consequences. Limits have been established consistent with NRC approved methods to ensure that fuel performance during normal, transient, and accident conditions is acceptable. The proposed Technical Specifications amendment uses conservatively established SLMCPR values for WNP-2 such that the fuel is protected during normal operation as well as during plant transients or anticipated operational occurrences.

The probability of an evaluated accident is not increased by the use of the ATRIUM-9X MCPR safety limit of 1.10 (two loop operation) or 1.11 (single loop operation). The ATRIUM-9X fuel was evaluated by SPC (Reference 5) using the additive constant uncertainty for ATRIUM-9X fuel of 0.0201 which is contained in the NRC safety evaluation approval of Reference 4. Based upon the NRC approved additive constant of uncertainty of 0.0201, as documented in Reference 5, at least 99.9% of the SPC ATRIUM-9X fuel rods would be expected to avoid boiling transition with a SLMCPR of 1.10 during two loop operation and 1.11 during single loop operation.

The probability of an evaluated accident is not increased by the use of the ABB SVEA-96 SLMCPRs of 1.10 (two loop operation) or 1.12 (single loop operation). NRC approved methodology documented in CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel", July 1996 (Reference 3) was used in deriving these ABB SVEA-96 SLMCPR values. The ABB evaluation as a function of cycle exposure established that late in Cycle 15 conservative two loop and single loop SLMCPRs of 1.10 and 1.12, respectively, can be used to represent the entire cycle.

The SLMCPR changes do not require any physical plant modifications, physically affect any plant component, or entail changes in plant operation. Therefore, no individual precursors of an accident are affected.

Since the operability of plant systems designed to mitigate any consequences of accidents have not changed, the consequences of an accident previously evaluated are not expected to increase.

The operation of WNP-2 in accordance with the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated:

Creation of the possibility of a new or different kind of accident would require the creation of one or more new precursors of that accident. New accident precursors may be created by modifications of the plant configuration, including changes in allowable modes of operation. This Technical Specification submittal does not involve any modifications of the plant configuration or allowable modes of operation. This Technical Specification change establishes SLMCPRs for SPC fuel based upon the NRC approved additive constant of uncertainty of 0.0201, as documented in Reference 5. At least 99.9% of the SPC ATRIUM-9X fuel rods would be expected to avoid boiling transition with a SLMCPR of 1.10 during two loop operation or 1.11 during single loop operation. Additionally, the ABB SVEA-96 SLMCPRs of 1.10 (two loop operation) or 1.12 (single loop operation) were derived using the NRC approved methodology documented in CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel", July 1996 (Reference 3). Therefore, no new precursors of an accident are created and no new or different kinds of accidents are created.

The operation of WNP-2 in accordance with the proposed amendment will not involve a significant reduction in the margin of safety for the following reasons:

Implementation of SLMCPRs derived by proven analytical methods provides a margin of safety by ensuring that less than 0.1% of the rods are expected to be in boiling transition if the MCPR limit is not violated. Because the fuel design safety criteria of more than 99.9% of the fuel rods avoiding transition boiling during normal operation as well as anticipated operational occurrences is met, there is not a significant reduction in the margin of safety.

Environmental Assessment Applicability Review

Washington Public Power Supply System has evaluated the proposed amendment against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10CFR51.21. It has been determined that the proposed changes meet the criteria for categorical exclusion as provided for under 10CFR51.22(c)(9). This conclusion has been determined because the change requested does not pose a significant hazards considerations nor does it involve a significant increase in the amounts, or a significant change in the types of any effluent that may be released off-site. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure.

Revised Technical Specification Pages

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 25% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

The MCPR for ATRIUM-9X fuel shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.11 for single recirculation loop operation. The MCPR for the ABB SVEA-96 fuel shall be ≥ 1.10 for two recirculation loop operation or ≥ 1.12 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

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2. ANF-NF-524(P)(A), Revision 2 and Supplements 1 and 2, "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors," November 1990;

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5. NEDE-23785-1-PA, Revision 1, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident, Volume III, SAFER/GESTR Application Methodology," October 1984;

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7. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," July 1996; and

8. WPPSS-FTS-131(A), Revision 1, "Applications Topical Report for BWR Design and Analysis," March 1996.

9. ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANF-1125(P)(A), Supplement 1, Appendix D, Siemens Power Corporation - Nuclear Division, July 1998.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

BASES

BACKGROUND (continued)

reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The reactor vessel water level SL ensures that adequate core cooling capability is maintained during all MODES of reactor operation. Establishment of Emergency Core Cooling System initiation setpoints higher than this safety limit provides margin such that the safety limit will not be reached or exceeded.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and AOOs. The reactor core SLs are established to preclude violation of the fuel design criterion that a MCPR limit is to be established, such that at least 99.9% of the fuel rods in the core would not be expected to experience the onset of transition boiling.

The Reactor Protection System setpoints (LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"), in combination with other LCOs, are designed to prevent any anticipated combination of transient conditions for Reactor Coolant System water level, pressure, and THERMAL POWER level that would result in reaching the MCPR limit.

2.1.1.1 Fuel Cladding Integrity

The use of the ANFB correlation is valid for critical power calculations at pressures > 600 psia and < 1500 psia and bundle mass fluxes $\geq 0.1 \times 10^6$ lb/hr-ft^2 and $< 1.5 \times 10^6$ lb/hr-ft^2 (Refs. 2 and 7). The use of the XL-S96 correlation is valid for critical power calculations at pressures > 392 psia and $\leq$ 1262 psia and bundle mass fluxes > 0.25×10^6 lb/hr-ft^2 and < 1.55×10^6 lb/hr-ft^2 (Ref. 3). For operation at low pressures or low flows, the fuel cladding integrity SL is established by a limiting condition on core THERMAL POWER, with the following basis:

> Provided that the water level in the vessel downcomer is maintained above the top of the active fuel, natural circulation is sufficient to ensure a minimum

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

bundle flow for all fuel assemblies that have a relatively high power and potentially can approach a critical heat flux condition. The minimum bundle flow is > 28×10^3 lb/hr. The coolant minimum bundle flow and maximum flow area are such that the mass flux is > 0.25×10^6 lb/hr-ft^2. Full scale critical power tests taken at pressures down to 14.7 psia indicate that the fuel assembly critical power at 0.25×10^6 lb/hr-ft^2 is approximately 3.35 MWt. At 25% RTP, a bundle power of approximately 3.35 Mwt corresponds to a bundle radial peaking factor of > 2.9, which is significantly higher than the expected peaking factor. Thus, a THERMAL POWER limit of 25% RTP for reactor pressures < 785 psig is conservative.

2.1.1.2 MCPR

The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an AOO from the limiting condition of operation, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition. The margin between calculated boiling transition (i.e., MCPR = 1.00) and the MCPR SL is based on a detailed statistical procedure that considers the uncertainties in monitoring the core operating state. One specific uncertainty included in the SL is the uncertainty inherent in the critical power correlations. Reference 4 describes the methodology used in determining the MCPR SL for Siemens Power Corporation fuel. Reference 5 describes the methodology used in determining the MCPR SL for ABB CENO fuel.

The critical power correlations are based on a significant body of practical test data, providing a high degree of assurance that the critical power, as evaluated by the correlation, is within a small percentage of the actual critical power. As long as the core pressure and flow are within the range of validity of the critical power correlations, the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat

(continued)

BASES (continued)

APPLICABILITY	SLs 2.1.1.1, 2.1.1.2, and 2.1.1.3 are applicable in all MODES.
SAFETY LIMIT VIOLATIONS	Exceeding an SL may cause fuel damage and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 6). Therefore, it is required to insert all insertable control rods and restore compliance with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and the probability of an accident occurring during this period is minimal.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. ANF-1125(P)(A), Revision 0, including Supplements 1 and 2, April 1990.
3. UR-89-210-P-A, "SVEA-96 Critical Power Experiments on a Full Scale 24-Rod Sub-Bundle," October 1993.
4. ANF-524(P)(A), Revision 2, including Supplements 1 and 2, November 1990.
5. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," July 1996.
6. 10 CFR 100.
7. ANFB Critical Power Correlation Uncertainty for Limited Data Sets, ANF-1125(P)(A), Supplement 1, Appendix D, Siemens Power Corporation - Nuclear Division, July 1998.