ENERGY
NORTHWEST

P.O. Box 968 ■ Richland, Washington 99352-0968

December 30, 2002
GO2-02-198

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATION 2.1.1.2, MCPR SAFETY LIMIT AND SR 3.3.1.3.2, OSCILLATION POWER RANGE MONITOR-LPRM CALIBRATION FREQUENCY**

Reference: Letter GO2-02-138, dated September 3, 2002, RL Webring (Energy Northwest) to NRC, "Request for Amendment to Technical Specification 4.2.1 and 5.6.5.b"

Dear Sir or Madam:

In accordance with 10 CFR 50.90, Energy Northwest is submitting a request for amendment to the Columbia Generating Station Technical Specifications (TS).

Columbia Generating Station is currently operating in cycle 16. The next refueling outage is scheduled to begin in May 2003 and end in June 2003. Energy Northwest requests approval of this request for amendment by May 30, 2003, so that it may be implemented prior to plant restart following completion of refueling outage 16.

This proposed amendment would revise two Technical Specifications. The first proposed change will revise TS 2.1.1.2, "Minimum Critical Power Ratio Safety Limit (MCPRSL)" to support operation during cycle 17. Cycle 17 will be the first cycle of operation with a mixed core of ABB/CE/Westinghouse SVEA-96 fuel and Framatome ANP ATRIUM™-10 reload fuel. The reload design and analyses, including determination of the MCPRSL, for cycle 17 have been performed utilizing the methodologies described in the referenced letter.

The second proposed change would revise Surveillance Requirement (SR) 3.3.1.3.2, the LPRM calibration frequency specified in the TS for the Oscillation Power Range Monitor (OPRM). This change will correct an inconsistency between the LPRM calibration frequency specified in SR 3.3.1.3.2 and SR 3.3.1.1.7, "Reactor Protection System (RPS) Instrumentation."

A001

The proposed amendment has been evaluated in accordance with 10 CFR 50.91(a)(1) using the criteria in 10 CFR 50.92(c) and Energy Northwest has determined that this amendment warrants a no significant hazards consideration. The discussion and justification for this determination are provided in Enclosure 3 to this submittal.

Framatome ANP, Inc. (FRA-ANP) provided supplemental information, Enclosure 4, regarding the MCPRSL analysis for cycle 17. The information in Enclosure 4 is considered to be proprietary. Therefore, pursuant to the requirements of 10 CFR 2.790, affidavits are enclosed to support the withholding of this information from public disclosure. A non-proprietary version of Enclosure 4 is provided as Enclosure 5.

Attachment 1 provides the existing TS pages marked up to show the proposed changes. Attachment 2 provides the revised typed TS pages. Attachment 3 provides the TS Bases pages marked up to show the changes that will be made to the Bases in support of the TS change.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Washington State Official.

If you have any questions or require additional information regarding this matter, please contact Ms. CL Perino, Licensing Manager at (509) 377-2075.

Respectfully,

D.K. Atkinson

DK Atkinson
Vice President, Technical Services
Mail Drop PE08

Enclosures:

1. Notarized Affidavit
2. Notarized Affidavit-Proprietary Information
3. Evaluation: Request for Amendment to Technical Specification 2.1.1.2 and SR 3.3.1.3.2
4. Supplemental Information for Enclosure 3, (Proprietary) and Affidavit
5. Supplemental Information for Enclosure 3, (Non-proprietary version of Enclosure 4)

Attachments:

1. Markup of current TS pages
2. Revised (typed) TS pages
3. Markup of TS Bases pages supporting the TS change

cc: EW Merschoff – NRC RIV (w/o Encl 4)
TC Poindexter – Winston & Strawn (w/o Encl 4)
NRC Resident Inspector – 988C (w/o Encl 4)
RN Sherman – BPA/1399 (w/o Encl 4)
BJ Benney – NRC NRR
JO Luce – EFSEC (w/o Encl 4)

ENCLOSURE 1, NOTARIZED AFFIDAVIT

STATE OF WASHINGTON)
)
COUNTY OF BENTON)

Subject: Amendment to Technical Specification 2.1.1.2 and SR 3.3.1.3.2

I, DK Atkinson, being duly sworn, subscribe to and say that I am the Vice President, Technical Services for ENERGY NORTHWEST, the applicant herein; that I have the full authority to execute this oath; that I have reviewed the foregoing; and that to the best of my knowledge, information, and belief the statements made in it are true.

DATE December 30, 2002

D.K. Atkinson

DK Atkinson
Vice President, Technical Services

On this date personally appeared before me DK Atkinson, to me known to be the individual who executed the foregoing instrument, and acknowledged that he signed the same as his free act and deed for the uses and purposes herein mentioned.

GIVEN under my hand and seal this 30th day of December 2002.

Lori A. Walli

Notary Public in and for the
STATE OF WASHINGTON

LORI A. WALLI
COMMISSION EXPIRES
NOTARY
PUBLIC
3-29-05
STATE OF WASHINGTON

Residing at Benton County

My Commission Expires 3-29-05

ENCLOSURE 2, NOTARIZED AFFIDAVIT FOR PROPRIETARY INFORMATION

STATE OF WASHINGTON)
)
COUNTY OF BENTON)

Subject: Framatome-ANP Letter Report DGC:02:021, Attachment A (FRAEN:02:099 and FRAEN:02:107) [Columbia Generating Station, Docket No. 50-397, Submittal of Request for Amendment to Technical Specifications 2.1.1.2, MCPR Safety Limit and SR 3.3.1.3.2.Oscillation Power Range Monitor-LPRM Calibration Frequency]

I, DK Atkinson, being duly sworn, subscribe to and say that I am the Vice President, Technical Services for ENERGY NORTHWEST, the applicant herein; that I have the full authority to execute this oath; that I have reviewed the foregoing; and that to the best of my knowledge, information, and belief the statements made in it are true.

Enclosure 4 to this letter contains information which is considered by Framatome-ANP to be proprietary. Included with Enclosure 4 is an affidavit executed by Jerald S Holm, Manger, Product Licensing for Framatome ANP, dated December 13, 2002 which provides the basis on which it is claimed that the subject document should be withheld from public disclosure under the provisions of 10 CFR 2.790.

Energy Northwest treats the subject document as proprietary information on the basis of statements by the owner. In submitting this information to the NRC, Energy Northwest requests that the subject document be withheld from public disclosure in accordance with 10 CFR 2.790.

DATE December 30, 2002

D.K. Atkinson
DK Atkinson
Vice President, Technical Services

On this date personally appeared before me DK Atkinson to me known to be the individual who executed the foregoing instrument, and acknowledged that he signed the same as his free act and deed for the uses and purposes herein mentioned.

GIVEN under my hand and seal this 30th day of December 2002.

LORI A. WALLI
COMMISSION EXPIRES
NOTARY
PUBLIC
3-29-05
STATE OF WASHINGTON

Lori A. Walli
Notary Public in and for the
STATE OF WASHINGTON

Residing at Benton County
My Commission Expires 3-29-05

EVALUATION: REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATION 2.1.1.2 AND SR 3.3.1.3.2

1.0 DESCRIPTION

This proposed amendment requests two revisions to the Columbia Generating Station Technical Specifications (TS).

The first proposed change would revise TS 2.1.1.2, Minimum Critical Power Ratio Safety Limit (MCPRSL) to support operation during cycle 17. Cycle 17 will be the first cycle of operation with a mixed core of ABB/CE/Westinghouse SVEA-96 fuel and Framatome ANP ATRIUM™-10 reload fuel. The cycle 17 reload design and analysis, including determination of the MCPRSL, have been performed utilizing the methodologies described in Reference 7.1.

The second proposed change would revise Surveillance Requirement (SR) 3.3.1.3.2, the Local Power Range Monitor (LPRM) calibration frequency specified in the TS for the Oscillation Power Range Monitor (OPRM). This change will correct an inconsistency between the LPRM calibration frequency specified in SR 3.3.1.3.2 and SR 3.3.1.1.7, Reactor Protection System (RPS) Instrumentation.

Approval is requested by May 30, 2003 to allow sufficient time to prepare and submit the Core Operating Limits Report (COLR) to the Plant Operations Committee prior to scheduled restart on June 12, 2003.

2.0 PROPOSED CHANGES

2.1 TS 2.1.1.2, MCPR Safety Limit

Technical Specification 2.1.1.2 currently identifies the MCPRSL for two fuel types, each with a different MCPRSL, for either two loop or single loop recirculation operation. This specification would be revised to eliminate reference to specific fuel types. The proposed MCPRSL would apply to all fuel in the core, i.e., the co-resident SVEA-96 and the ATRIUM™-10 fuel. Limits would continue to be provided for both two loop and single loop recirculation operation.

The current specification is as follows:

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 loop operation.

The proposed specification is as follows:

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

The MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.10 for single recirculation loop operation.

2.2 SR 3.3.1.3.2, LPRM Calibration Frequency

Surveillance Requirement 3.3.1.3.2 currently requires calibration of the LPRM at a frequency of 1000 MWD/T average core exposure. This specification would be revised from a frequency of 1000 MWD/T to a frequency of 1130 MWD/T average core exposure to be consistent with the LPRM calibration frequency required by SR 3.3.1.1.7.

3.0 BACKGROUND

3.1 TS 2.1.1.2, MCPR Safety Limit

Columbia Generating Station is licensed to operate at an up-rated power level of 3486 MWt and is in the 16th cycle of operation. Cycle 16 is a nominal 24-month cycle and is the second cycle in the transition from annual to biennial refueling cycles. Cycle 15 was the first cycle in the transition and the plant operated for 19 months prior to the refueling outage for cycle 16. The cycle 16 fuel load is entirely SVEA-96 fuel.

Cycle 17 will be the first cycle of operation at Columbia Generating Station with a mixed core of Westinghouse SVEA-96 fuel and Framatome ANP (FRA-ANP) ATRIUM™-10 reload fuel. The cycle 17 core will consist of 280 fresh ATRIUM™-10, 12 fresh SVEA-96 and 472 burned SVEA-96 assemblies. ATRIUM-9X fuel is no longer loaded into the core and reference to that fuel type will be removed from the TS.

No change in power level or cycle length will be implemented coincident with the fuel vendor and fuel design change. The MCPRSL analysis was performed to support the current licensed power level and the 24-month operating cycle.

The reload design and licensing analysis, including the MCPRSL analysis to support the cycle 17 operation, are performed by FRA-ANP utilizing NRC approved methodologies. These methodologies are described in the proposed changes to TS 5.6.5.b that were submitted in Reference 7.1. Specific analyses to determine the MCPRSL are performed on a cycle-by-cycle basis because core design changes can affect the safety limit. Since the reactor core for cycle 17 will consist of fuel from two different suppliers, it was necessary to establish a critical power correlation for the SVEA-96 fuel utilizing NRC approved Framatome ANP methodologies and then to apply that correlation to the core design and analysis. The methodologies used for determining the critical power correlation for the SVEA-96 fuel are provided in EMF-2245(P)(A), "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel" (Reference 7.2) and EMF-2209(P)(A), "SPCB Critical

Power Correlation" (Reference 7.3). The methodology used for determining the MCPRSL is found in ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors" (Reference 7.4).

While we do not expect the core configuration described in this request to change, any final core design changes will be evaluated to confirm that the proposed Technical Specification changes remain valid. Core designs for future cycles will be evaluated for the continued applicability of these changes.

3.2 SR 3.3.1.3.2 LPRM Calibration Frequency

The Columbia Generating Station Technical Specification currently identifies two different surveillance frequencies for calibration of the LPRMs. Specification, SR 3.3.1.1.7 establishes a calibration frequency of 1130 MWD/T core average exposure for the LPRMs in support of the APRM requirement for the Rector Protection System (RPS) Instrumentation Limiting Condition for Operation. This specification was approved in Amendment 149 (Reference 7.5) with implementation of the Improved Technical Specifications. Specification SR 3.3.1.3.2 establishes a calibration frequency of 1000 MWD/T and was approved in Amendment 171 (Reference 7.6) with implementation of the OPRM. The proposed change will reconcile that inconsistency by revising SR 3.3.1.3.2 to 1130 MWD/T.

4.0 TECHNICAL ANALYSIS

4.1 MCPR Safety Limit

The cycle 17 core will consist of a combination of fresh and exposed Westinghouse SVEA-96 fuel and FRA-ANP ATRIUM™-10 fuel. A core map providing a description of fuel type, cycle loaded, location and the number of assemblies for each fuel type is shown on Enclosure 4, Figure A-2.

The MCPRSL is developed to ensure compliance with 10 CFR 50 Appendix A, General Design Criteria for Nuclear Power Plants Criterion 10, Reactor Design, by meeting the acceptance criterion in NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Section 4.4, Thermal and Hydraulic Design. In summary, Section 4.4 requires that for critical power correlations, the limiting (minimum) value of critical power ratio is to be established such that at least 99.9% of the fuel rods in the core would not be expected to experience departure from nucleate boiling or boiling transition during normal operation or anticipated operational occurrences.

A critical power correlation for the co-resident SVEA-96 fuel was determined by Energy Northwest consultants using NRC approved methodology (References 7.2 and 7.3). Appropriate additive constants and additive constant uncertainties for the SVEA-96 fuel were then provided to FRA-ANP to perform the analysis required to establish the MCPRSL. The MCPRSL determined by this approach applies to both the SVEA-96 and ATRIUM™-10 fuel.

The Energy Northwest consultants were qualified to perform this task in accordance with the guidelines of Generic Letter (GL) 83-11, Supplement 1, Licensee Qualification for Performing Safety Analyses.

The following discussion provides a summary of the qualification activities, results for the SVEA-96 fuel and the determination of the MCPRSL for the cycle 17 core.

4.1.1 Determination of Critical Power Correlation for the SVEA-96 Fuel

The critical power correlation for the SVEA-96 fuel was determined using the direct correlation application method described in EMF-2245(P)(A) (Reference 7.2). The direct method can be applied to co-resident fuel when sufficient experimental data is available to the licensee or FRA-ANP to establish a direct correlation for the fuel using an approved FRA-ANP correlation. Test data used by Westinghouse to determine their critical power correlation for the SVEA-96 was used in this analysis. This data, available to Energy Northwest, was obtained from the NRC approved topical reports described in References 7.7 and 7.8 and was used to determine the appropriate FRA-ANP correlation for the SVEA-96 fuel. Since this data is proprietary, Energy Northwest, with support from two consultants, performed the evaluations and provided the results to FRA-ANP for use in the reload design and analysis including determination of the MCPRSL.

Use of the direct correlation application described in Reference 7.2 by Energy Northwest's consultants required completion of an appropriate technology transfer process that met the guidelines of GL 83-11, Supplement 1. FRA-ANP developed and conducted a training program and qualified the individuals involved to perform the analysis. This program involved training on the methodology in Reference 7.2, the FRA-ANP process used to perform the analysis and successful completion of comparison calculations. Energy Northwest qualified the consultants to document the work in accordance with the Energy Northwest design control procedures for preparation and control of calculations.

The consultants performed a rigorous statistical evaluation examining the application of several approved critical power correlations to the SVEA-96 experimental data. This evaluation included inspection for unexpected trends and behavior of the correlations over the range of conditions required for application of the correlation. The FRA-ANP SPCB critical power correlation using the ATRIUM-10 branch (Reference 7.3) was determined to best apply to the SVEA-96 fuel. The SVEA-96 additive constant uncertainty analysis was then performed and the results of the evaluation were documented and approved. Energy Northwest provided the results to FRA-ANP for use in determining the MCPRSL.

As discussed above, 12 fresh SVEA-96 fuel assemblies will be included in the cycle 17 reload. The application of the methodology in EMF-2245(P)(A) is intended to apply to exposed co-resident fuel. Since the 12 assemblies are fresh and not exposed, they will be treated as lead fuel assemblies and loaded into non-limiting core locations. However, it is reasonably expected that the SPCB correlation based on SVEA-96 experimental test data would apply equally to the fresh SVEA-96 fuel. Enclosure 4, Figure A-2, is a quarter-core loading map for cycle 17 and shows the core location for these assemblies. The cycle 17 core is generally quarter-core symmetric. Figures A-3 and A-4 show the Radial Power Distribution and MFLCPR Distribution at the cycle limiting exposure (15,600 MWD/MTU) for the fresh SVEA-96 fuel assemblies. MFLCPR is the ratio of the MCPR operating limit divided by the assembly MCPR. Inspection of these figures shows the MFLCPR for the fresh SVEA-96 fuel is less than that for other assemblies and therefore the twelve assemblies are loaded into non-limiting locations in the core.

4.1.2 Determination of the Minimum Critical Power Safety Limit

The MCPRSL was determined using the approved methodology in ANF-524(P)(A) (Reference 7.4). The safety limit is determined via a statistical calculation that combines system measurement and calculational uncertainties to determine a MCPRSL that protects 99.9 percent of the fuel rods from boiling transition during normal operation and anticipated operational occurrences. This methodology utilizes a Monte Carlo procedure to account for the measurement and calculational uncertainties. A design basis power distribution is assumed that conservatively represents the expected power distribution. The analysis used the SPCB critical power correlation additive constants and additive constant uncertainty for the ATRIUM-10 fuel as reported in Reference 7.3. The SPCB critical power correlation with additive constants and uncertainties, determined as described in paragraph 4.1.1, were used for the SVEA-96 fuel.

Information to support the cycle specific MCPRSL is provided in Enclosure 4. The results of the analysis, summarized on Table A-2 for the limiting cycle exposure, 6000 MWD/MTU, show that for the proposed MCPR safety limit of 1.09 for two recirculation loop operation and MCPR safety limit of 1.10 for single loop operation, the number of rods predicted to reach transition boiling is less than 0.1 percent. Therefore, 99.9 percent of the rods will avoid boiling transition during normal operation and anticipated operational occurrences.

4.2 SR 3.3.1.3.2 LPRM Calibration Frequency

The LPRMs are periodically calibrated by comparing their output with the local flux profile measured by utilizing the Traversing Incore Probe System (TIP). This establishes the relative local flux profile signal for the APRM and OPRM systems. The calibration frequency is based on LPRM operating experience. Traditionally, the calibration frequency was established as 1000 effective full power operating hours (EFPH). When SR 3.3.1.1.7 was implemented as part of the Columbia Generating Station Improved Technical Specifications (Reference 7.5), the frequency was changed from 1000 EFPH to 1130 MWD/T average core exposure. This change was considered a unit change and resulted in a more convenient tracking parameter since MWD/T is commonly calculated and recorded by the core monitoring software system. In addition, it represented roughly the same time interval (approximately 6 weeks).

OPRM operation is also based on inputs from the LPRMs. When the OPRM was implemented for Columbia Generating Station, the OPRM standard technical specifications provided in Reference 7.9 were used. The standard specifications provided for incorporation of a plant specific frequency in SR 3.3.1.3.2. This inconsistency was not noted at that time and 1000

MWD/T was selected based on the suggested frequency noted in Reference 7.9. Further review of Reference 7.10 concludes that LPRM calibration frequency established for the APRM was intended to also apply to the OPRM.

As noted in Enclosure 4, the proposed LPRM calibration frequency of 1130 MWD/T is supported by the MCPRSL analysis. Therefore, this change will not impact the proposed MCPRSLs.

5.0 REGULATORY SAFETY ANALYSIS

5.1 10 CFR 50.92 Evaluation

Energy Northwest is proposing that the Columbia Generating Station TS be amended to:

1. Revise the TS 2.1.1.2, MCPR safety limit to reflect the cycle 17 specific analyses such that the MCPR shall be ≥ 1.09 for two recirculation loop operation and shall be ≥ 1.10 for single loop operation.

and,

2. Revise SR 3.3.1.3.2, LPRM calibration frequency for the OPRM to 1130 MWD/T.

The Minimum Critical Power Ratio Safety Limit is established in accordance with General Design Criterion 10 and NUREG-0800, Section 4.4 to ensure that during normal operation and anticipated operational occurrences, at least 99.9 percent of the rods in the core do not experience transition boiling. The analysis to support this change shows that the number of rods protected exceeds 99.9 percent of the rods in the core and therefore the proposed MCPRSLs meet the acceptance criterion. The proposed LPRM surveillance frequency for the OPRM is consistent with a value that has previously been approved for the APRM by the NRC.

No changes in power level or cycle length from those implemented in previous cycles are proposed.

An evaluation of the proposed changes has been performed in accordance with 10 CFR 50.91(a)(1) regarding no significant hazards consideration using the standards in 10 CFR 50.92(c). A discussion of these standards as they relate to this amendment request follows:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

1. The requested change to TS 2.1.1.2, MCPRSL to support the cycle 17 core loading does not involve any plant modifications or operational changes that could affect system reliability, performance, or possibility of operator error. The requested changes do not

affect any postulated accident precursors, do not affect any accident mitigation systems, and do not introduce any new accident initiation mechanisms. The consequences of accidents previously evaluated are not changed because the number of rods that are protected from transition boiling is predicted to be greater than 99.9 percent which meets the acceptance criterion in NUREG-0800, Section 4.4.

2. The requested change to SR 3.3.1.3.2, OPRM/LPRM calibration frequency, does not involve a modification to the plant or introduce the probability of an operator error. The LPRMs are not the precursor to any accident. Making the LPRM surveillance frequency for the OPRM consistent with that approved for the RPS/APRM does not change system reliability. The proposed LPRM surveillance frequency is supported by the uncertainties used to perform the MCPRSL analyses. Therefore, the number of rods that are calculated to experience transition boiling during normal operation or anticipated operational occurrences will not be changed and the consequences of these events will not be increased.

Therefore, these changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

1. The ATRIUM-10 fuel to be used in cycle 17 is compatible with the co-resident SVEA-96 fuel. This compatibility is demonstrated by application of the FRA-ANP critical power methodology to the core design that includes the ATRIUM-10 and SVEA-96 fuel. The proposed changes do not represent any new modes of operation, changes in setpoints or plant modifications other than those required for the reactor core. The change does not introduce new postulated accident precursors or mitigation systems. Reload design and analysis will be performed in accordance with approved NRC methodology.

2. Increasing the time interval for the OPRM/LPRM surveillance reduces the frequency to be consistent with the LPRM surveillance frequency for the RPS/APRM and does not involve a modification to the plant, introduce a new operator error or revise setpoints.

Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No.

1. The proposed MCPRSL does not involve a significant reduction in the margin of safety associated with the criterion set forth in NUREG-0800, Section 4.4. The safety limit

established for the core ensures that the criterion for the number of fuel rods allowed to experience transition boiling will be maintained for normal plant operation and anticipated operational transients.

The core operating limits will continue to be determined using methodologies that have been approved by the NRC.

2. The proposed LPRM surveillance frequency is supported by the uncertainties used to perform the MCPRSL analyses. Therefore, the number of rods that are calculated to experience transition boiling during normal operation or anticipated operational occurrences will not be changed.

Therefore, implementation of the change to the MCPRSL and the LPRM surveillance frequency does not involve a significant reduction in margin of safety.

Based on the above evaluation, Energy Northwest concludes that the proposed amendment presents no significant hazards under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Criteria

The MCPRSL is developed to ensure compliance with the 10 CFR 50 Appendix A, General Design Criteria for Nuclear Power Plants, Criterion 10, Reactor Design. Compliance with Criterion 10 is based upon acceptance criteria in NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Section 4.4. Specific criteria in Section 4.4 requires for CPR correlations, the limiting (minimum) value of CPR is to be established such that at least 99.9% of the fuel rods in the core would not be expected to experience departure from nucleate boiling or boiling transition during normal operation or anticipated operational occurrences.

The NRC approved FRA-ANP methodology used in establishing the MCPRSL ensures that this acceptance criterion is met for the cycle 17 reload design.

In conclusion, (1) there is reasonable assurance that the health and safety of the public will not be endangered by the operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

Energy Northwest has evaluated the proposed amendment against the criteria for identification of license and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

7.1 Letter GO2-02-138, dated September 3, 2002, R. L. Webring (Energy Northwest) to NRC, "Request for Amendment to Technical Specification 4.2.1 and 5.6.5.b"

7.2 EMF-2245 (P)(A) Revision 0, "Application of Siemens Power Corporation's Critical Power Correlation to Co-Resident Fuel," Siemens Power Corporation, August 2000.

7.3 EMF-2209 (P)(A) Revision 1, "SPCB Critical Power Correlation," Siemens Power Corporation, July 2000

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

7.5 Letter, Timothy G Colburn (NRC) to J. V. Parrish (ENW), "Issuance of Amendment for the Washington Public Power Supply System Nuclear Project No. 2 (TAC No. M94226)," dated March 4, 1997

7.6 Letter, Jack Cushing (NRC) to J. V. Parrish, "Columbia Generating Station-Issuance of Amendment Re: Oscillation Power Range Monitoring Technical Specifications (TAC No. MB0483)" dated April 5, 2001

7.7 UR-89-210-P-A, "SVEA-96 Critical Power Experiments on a Full Scale 24-Rod Sub-Bundle," ABB Combustion Engineering Nuclear Fuel, October 1993

7.8 CENPD-392-P-A, "10X10 SVEA Fuel Critical Power Experiments and CPR Correlations: SVEA-96," CE Nuclear Power LLC, September 2000

7.9 CENPD-400-P-A, Rev 01, "Generic Topical Report for the ABB Option III Oscillation Power Range Monitor (OPRM)," ABB Combustion Engineering, May 1995

7.10 NEDO-31960-A, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology," November 1995

ENCLOSURE 5

Supplemental Information for Enclosure 3

Attachment A, Columbia Generating Station Cycle 17 MCPR Safety Limit Analysis

This enclosure is a non-proprietary version of Enclosure 4

Columbia Generating Station Cycle 17
MCPR Safety Limit Analysis

Reactor system measurement uncertainties are statistically convolved with MCPR calculational uncertainties to determine a MCPR safety limit that ensures that less than 0.1% of the fuel rods in the reactor core experience boiling transition during normal operation and or an anticipated operational occurrence. [

] The MCPR safety limit is used in conjunction with transient analysis results to establish the MCPR operating limit. The MCPR safety limit methodology is described in Reference A.1.

The final core design and step through developed to meet the Reference A.2 operating requirements was used in the Columbia Generating Station Cycle 17 (CGSC17) MCPR safety limit analysis. The 24-month Cycle 17 design supports operation to an EOFP cycle exposure of about 17,400 MWd/MTU with a licensed rated power of 3486 MWt.

The CGSC17 MCPR safety limit analysis used the SPCB critical power correlation additive constants and additive constant uncertainty for the ATRIUM™-10* fuel reported in Reference A.3. Energy Northwest developed the SPCB additive constants and additive constant uncertainty for the SVEA-96 fuel (References A.4 and A.5) using the direct approach described in Reference A.6. The effects of channel bow were explicitly accounted for in the analysis consistent with the process described in Reference A.1. Channel bow data for SVEA-96 fuel was provided in Reference A.7. [

] The analysis supports:

- Fuel- and plant-related uncertainties for CGSC17 presented in Table A.1.
- 50% of the LPRMs out of service (LPRM bypass model on or off).
- Up to one TIP machine out of service, or the equivalent number of TIP channels (100% available at startup).
- 1130 MWd/T LPRM calibration interval.
- No reused channels.

[

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The results support a two-loop operation (TLO) MCPR safety limit of 1.09. Table A.2 presents a summary of the analysis results including the MCPR safety limit and the percentage of rods expected to experience boiling transition. Analyses were performed using the power distributions from each

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* ATRIUM is a trademark of Framatome ANP.

exposure in the design step through. The safety limit radial power histogram for the limiting cycle exposure of 6000 MWd/MTU (i.e., the exposure that results in the highest number of rods expected to experience boiling transition) is presented in Figure A.1. Results for single-loop operation (SLO) are also presented in Table A.2 and support a MCPR safety limit of 1.10.

During an October 2, 2002 meeting, the NRC requested that core loading and power distribution maps be included in the CGSC17 Technical Specification submittal. The purpose is to provide information concerning the CPR performance of the 12 fresh SVEA-96 assemblies. The Cycle 17 core loading is presented in Figure A.2. A review of the final Cycle 17 core design shows that the 12 fresh SVEA-96 assemblies always have at least 17.8% margin to the MCPR operating limit during the cycle. The cycle exposure at which the minimum MCPR margin occurs for the fresh SVEA-96 fuel assemblies is 15,600 MWd/MTU. The core radial power and MFLCPR (MCPR operating limit divided by assembly MCPR) distributions at a cycle exposure of 15,600 MWd/MTU are presented in Figures A.3 and A.4 respectively.

References:

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

A.2 Letter, D. F. Richey (EN) to J. L. Raklios (FANP), "Columbia Generating Station Final Operating Requirements for Cycle 17," ENFRA-02-028, October 3, 2002.

A.3 EMF-2209(P)(A) Revision 1, *SPCB Critical Power Correlation*, Siemens Power Corporation, July 2000.

A.4 Letter, D. F. Richey (EN) to L. Raklios (FANP), "Columbia Generating Station Reload Fuel Design Data Package Transmittal No. 012," ENFRA-02-035, October 18, 2002.

A.5 Letter, D. F. Richey (EN) to L. Raklios (FANP), "Columbia Generating Station Reload Fuel Design Data Package Transmittal No. 015," ENFRA-02-041, November 18, 2002.

A.6 EMF-2245(P)(A) Revision 0, *Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel*, Siemens Power Corporation, August 2000.

A.7 Letter D. F. Richey (EN) to L. Raklios (FANP), "Columbia Generating Station Reload Fuel Design Data Package Transmittal No. 013," ENFRA-02-039, November 1, 2002.

A.8 EMF-2158(P)(A), *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2,* Siemens Power Corporation, October 1999.

A.9 EMF-2744 Revision 0, *Columbia Generating Station Cycle 17 Plant Parameters Document*, Framatome ANP, August 2002.

A.10 Letter, H. Donald Curet (SPC) to H. J. Richings (USNRC), "POWERPLEX® Core Monitoring: Failed or Bypassed Instrumentation and Extended Calibration," HDC:96:012, May 1996.

Table A.1 Fuel- and Plant-Related Uncertainties for
Columbia Generating Station Cycle 17
MCPR Safety Limit Analyses

Parameter	Standard Deviation	Reference
Fuel-Related Uncertainties		
[		
		]
System-Related Uncertainties		
Feedwater flow rate	1.76%	A.9
Feedwater temperature	0.76%	A.9
Core pressure	0.50%	A.9
Total core flow rate TLO SLO	 2.50% 6.00%	 A.9 A.9

* Additive constant uncertainty is an absolute value rather than a relative one.

† Values calculated by Framatome ANP based on the methodology presented in Reference A.10.

‡ These values were calculated by Framatome ANP and are applicable only to SVEA-96 fuel.

Table A.2 Results Summary for Columbia Generating Station Cycle 17 MCPR Safety Limit Analysis	
SLMCPR	Percentage of Rods in Boiling Transition
TLO – 1.09	0.0802
SLO – 1.10	0.0741

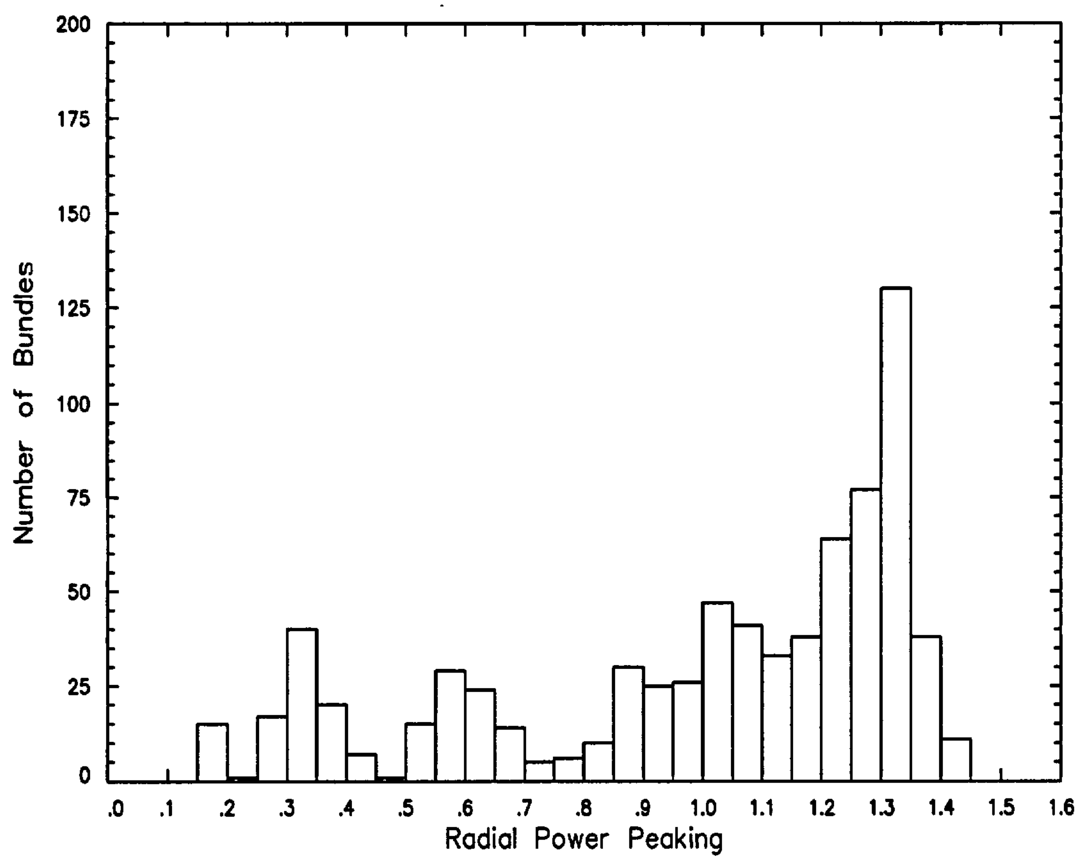

Figure A.1 Columbia Generating Station Cycle 17
Radial Power Distribution for SLMCPR Determination

36	1	36	1	36	1	36	1	36	1	36	1	36	33	33
1	36	1	36	1	36	1	36	1	36	1	36	1	36	33
36	1	36	1	36	36	36	1	36	1	36	1	36	33	33
1	36	1	36	1	36	1	36	1	36	1	36	1	33	33
36	1	36	1	36	1	36	1	36	1	36	1	36	33	33
1	36	36	36	1	36	1	36	1	36	1	36	**38**	33	33
36	1	36	1	36	1	36	1	36	1	36	1	33	33	29
1	36	1	36	1	36	1	36	1	36	1	1	33	29	
36	1	36	1	36	1	36	1	36	1	36	33	33		
1	36	1	36	1	36	1	36	1	**38**	33	33	28		
36	1	36	1	36	1	36	1	36	33	33				
1	36	1	36	1	36	1	1	33	33					
36	1	36	1	36	**38**	33	33	33	28					
33	36	33	33	33	33	33	29							
33	33	33	33	33	33	29								

Fuel Description	Cycle Loaded	Nuclear Fuel Type	Total Number of Assemblies
SVEA-96	13	28	8
SVEA-96	14	29	17
SVEA-96	15	33	147
SVEA-96	16	36	300
SVEA-96	17	38	12
ATRIUM-10	17	1	280

Figure A.2 Quarter-Core Loading Map
for Cycle 17*

* The Cycle 17 core loading is generally quarter-core symmetric.

1.183	1.425	1.181	1.422	1.167	1.319	1.066	1.037	0.894	1.255	1.083	1.209	0.908	0.586	0.314
1.425	1.239	1.430	1.157	1.324	1.095	1.298	1.076	1.281	1.126	1.303	1.030	1.045	0.697	0.316
1.182	1.430	1.178	1.284	1.053	1.063	1.112	1.345	1.165	1.364	1.136	1.223	0.924	0.580	0.305
1.421	1.156	1.284	0.858	0.975	1.055	1.344	1.200	1.430	1.174	1.328	1.063	1.034	0.567	0.292
1.165	1.323	1.053	0.975	0.852	1.301	1.211	1.453	1.223	1.388	1.156	1.203	0.872	0.535	0.270
1.318	1.094	1.064	1.056	1.301	1.171	1.461	1.229	1.424	1.145	1.300	1.016	**0.956**	0.482	0.237
1.065	1.297	1.111	1.344	1.216	1.461	1.230	1.443	1.207	1.344	1.081	1.086	0.652	0.396	0.178
1.036	1.075	1.345	1.200	1.453	1.230	1.443	1.254	1.376	1.079	1.156	0.945	0.516	0.253	
0.894	1.280	1.167	1.430	1.225	1.425	1.210	1.376	1.129	1.184	0.882	0.590	0.354		
1.255	1.127	1.364	1.176	1.390	1.147	1.346	1.079	1.186	**1.047**	0.612	0.369	0.187		
1.085	1.304	1.137	1.330	1.164	1.305	1.085	1.160	0.885	0.635	0.381				
1.211	1.032	1.227	1.069	1.210	1.024	1.095	0.952	0.588	0.366					
0.911	1.049	0.939	1.042	0.882	**0.967**	0.685	0.523	0.356	0.190					
0.589	0.704	0.586	0.575	0.545	0.493	0.405	0.258							
0.317	0.320	0.309	0.294	0.274	0.242	0.183								

Figure A.3 Radial Power Distribution for Cycle 17
at 15,600 MWd/MTU*

0.768	0.845	0.774	0.849	0.815	0.801	0.744	0.691	0.765	0.776	0.749	0.738	0.658	0.448	0.350
0.845	0.874	0.849	0.753	0.791	0.720	0.783	0.754	0.788	0.810	0.809	0.702	0.650	0.526	0.344
0.779	0.849	0.857	0.765	0.748	0.735	0.742	0.822	0.803	0.840	0.809	0.747	0.666	0.444	0.349
0.849	0.763	0.765	0.701	0.595	0.742	0.815	0.848	0.879	0.802	0.826	0.747	0.650	0.431	0.333
0.814	0.790	0.745	0.595	0.715	0.784	0.836	0.892	0.853	0.857	0.790	0.738	0.638	0.404	0.314
0.801	0.721	0.733	0.741	0.785	0.867	0.896	0.847	0.880	0.760	0.814	0.731	**0.722**	0.364	0.280
0.748	0.786	0.743	0.816	0.836	0.896	0.883	0.890	0.830	0.834	0.779	0.668	0.498	0.324	0.213
0.692	0.759	0.823	0.849	0.893	0.850	0.891	0.845	0.853	0.734	0.720	0.573	0.409	0.269	
0.767	0.788	0.801	0.880	0.855	0.881	0.833	0.853	0.816	0.734	0.658	0.461	0.432		
0.777	0.813	0.840	0.798	0.860	0.761	0.837	0.765	0.736	**0.822**	0.484	0.427	0.249		
0.754	0.811	0.811	0.829	0.809	0.818	0.782	0.723	0.660	0.510	0.421				
0.750	0.717	0.750	0.752	0.754	0.710	0.676	0.578	0.465	0.394					
0.659	0.655	0.674	0.657	0.646	**0.731**	0.551	0.413	0.436	0.253					
0.452	0.533	0.445	0.438	0.411	0.372	0.329	0.295							
0.354	0.348	0.354	0.337	0.320	0.285	0.219								

Figure A.4 MFLCPR Distribution for Cycle 17
at 15,600 MWd/MTU*

---

* Radial power and MFLCPR are generally quarter-core symmetric.

ATTACHMENT 1

REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATION 2.1.1.2 AND SR 3.3.1.3.2

Markup of Current Technical Specifications

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.

Insert

The MCPR shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.10 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.

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.1.3.2	Calibrate the local power range monitors.	1000 MWD/T average core exposure 1130
SR 3.3.1.3.3	------------------NOTE------------------ Neutron detectors are excluded. -- Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.3.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.3.5	Verify OPRM is not bypassed when THERMAL POWER is ≥ 30% RTP and core flow ≤ 60% rated core flow.	24 months
SR 3.3.1.3.6	------------------NOTE------------------ Neutron detectors are excluded. -- Verify the RPS RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

ATTACHMENT 2

REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATION 2.1.1.2 AND SR 3.3.1.3.2

Revised (typed) TS 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 shall be ≥ 1.09 for two recirculation loop operation or ≥ 1.10 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.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.3.2	Calibrate the local power range monitors.	1130 MWD/T average core exposure
SR 3.3.1.3.3	------------------NOTE------------------ Neutron detectors are excluded. -- Perform CHANNEL CALIBRATION.	 24 months
SR 3.3.1.3.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.3.5	Verify OPRM is not bypassed when THERMAL POWER is $\geq$ 30% RTP and core flow $\leq$ 60% rated core flow.	24 months
SR 3.3.1.3.6	------------------NOTE------------------ Neutron detectors are excluded. -- Verify the RPS RESPONSE TIME is within limits.	 24 months on a STAGGERED TEST BASIS

ATTACHMENT 3

REQUEST FOR AMENDMENT TO TECHNICAL SPECIFICATION 2.1.1.2 AND SR 3.3.1.3.2

Markup of TS Bases pages supporting the TS Change

LDCN-TSB-02-074, Inserts for TSB pages as identified on the mark-up.

B 2.1.1.2, MCPR Safety Limit

Insert A (p. B 2.1.1-2)

The SPCB critical power correlation is used for the Framatome ANP and Westinghouse SVEA-96 fuel. The use of the correlation for the Framatome ANP fuel is valid for critical power calculations at pressures $\geq$ 571.4 psia and $\leq$ 1432 psia and bundle mass fluxes $\geq 0.087 \times 10^6$ lbm/hr-ft^2 and $\leq 1.5 \times 10^6$ lbm/hr-ft^2 (Reference 2). Application of the SPCB critical power correlation to the Westinghouse SVEA-96 fuel was established using the methodology presented in Reference 3. The correlation for the SVEA-96 fuel is valid for critical power calculations at pressures $\geq$ 576 psia and $\leq$ 1261 psia and bundle mass fluxes $\geq 0.21 \times 10^6$ lbm/hr-ft^2 and $\leq 1.61 \times 10^6$ lbm/hr-ft^2 (Reference 4).

Insert B (p. B 2.1.1-5)

2. EMF-2209(P)(A) Revision 1, " SPCB Critical Power Correlation," Siemens Power Corporation, July 2000

3. EMF-2245(P)(A) Revision 0, " Application of Siemens Power Corporation's Critical Power Correlation to Co-resident Fuel," Siemens Power Corporation, August 2000

4. NE-02-02-15 Revision 0, "Computation of SPCB Critical Power Correlation Additive Constants for SVEA-96," November 2002

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

B 3.2.2 Minimum Critical Power Ratio (MCPR)

Insert C (p. B 3.2.2-2)

Flow dependent MCPR limits are determined by steady-state thermal hydraulic methods using the three-dimensional BWR simulator code (Reference 2) and a multi-channel thermal-hydraulic code (Reference 3).

Insert D (p B 3.2.2-2)

Power dependent MCPR limits ($MCPR_P$) are determined by the three-dimensional BWR simulator code (Reference 2) and a multi-channel thermal-hydraulic code (Reference 3).

Insert E (p. B 3.2.2-2)

MCPR operating limits that include the effects of analyzed equipment out-of-service are also included in the COLR.

Insert F (p. B 3.2.2-4)

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

2. XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, "Exxon Nuclear Methodology for Boiling water reactors - Neutronic Methods for Design and Analysis," Exxon nuclear Company, March 1983

3. XN-NF-80-19(P)(A) Volume 3 Revision 2, "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description," January 1987

B 3.2.3 Linear Heat Generation Rate (LHGR)

Insert G (p. B 3.2.3-3)

1. FSAR Chapter 4

2. FSAR Chapter 15 and 15.F

3. CENPD 287-P-A, "Fuel Assembly Design Methodology for Boiling Water Reactors," ABB Combustion Engineering Nuclear Operations, July 1996

4. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," ABB Combustion Engineering Nuclear Operations," July 1996

5. ANF-89-98(P)(A) Revision 1 and Supplement 1, Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995

6. EMF-85-74(P) Revision 0 Supplement 1 (P)(A) and Supplement 2 (P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model," Siemens Nuclear Power Corporation, February 1998

7. NUREG-0800, Section II A.2(g), Revision 2, July 1981

8. 10 CFR 50.36(c)(2)(ii)

BASES

BACKGROUND (continued)

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of transition boiling and the resultant sharp 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

Insert A

The use of the ANFB correlation is valid for critical power calculations at pressures > 600 psia and < 1500 psia and bundle mass fluxes > 0.1×10^6 lb/hr-ft^2 and < 1.5×10^6 lb/hr-ft^2 (Refs. 2 and 7). The use of the ABBD1.0 correlation is valid for critical power calculations at pressures > 363 psia and < 1261 psia and bundle mass fluxes > 0.21×10^6 lb/hr-ft^2 and < 1.62×10^6 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:

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

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 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.

MWt

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 correlation~~s~~. Reference 7 describes the use of increased ANFB additive constant uncertainty for the SPC ATRIUM-9X fuel. 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 Westinghouse fuel.

References 2, 3 and 5 describe the uncertainties and methodology used in determining the MCPR SL.

Insert

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

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.2 MCPR (continued)

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 flat radial power factors and bounding high local peaking distributions are used to estimate the number of rods in boiling transition. This conservatism and the inherent accuracy of the critical power correlations provide a reasonable degree of assurance that there would be no transition boiling in the core during sustained operation at the MCPR SL. If boiling transition were to occur, there is reason to believe that the integrity of the fuel would not be compromised. Significant test data accumulated by the NRC and private organizations indicate that the use of a boiling transition limitation to protect against cladding failure is a very conservative approach. Much of the data indicate that BWR fuel can survive for an extended period of time in an environment of boiling transition.

Insert

...by using conservative radial and local power distributions...

2.1.1.3 Reactor Vessel Water Level

During MODES 1 and 2, the reactor vessel water level is required to be above the top of the active irradiated fuel to provide core cooling capability. With fuel in the reactor vessel during periods when the reactor is shut down, consideration must be given to water level requirements due to the effect of decay heat. If the water level should drop below the top of the active irradiated fuel during this period, the ability to remove decay heat is reduced. This reduction in cooling capability could lead to elevated cladding temperatures and clad perforation in the event that the water level becomes < 2/3 of the core height. The reactor vessel water level SL has been established at the top of the active irradiated fuel to provide a point that can be monitored and to also provide adequate margin for effective action.

SAFETY LIMITS

The reactor core SLs are established to protect the integrity of the fuel clad barrier to prevent the release of radioactive materials to the environs. SL 2.1.1.1 and SL 2.1.1.2 ensure that the core operates within the fuel design criteria. SL 2.1.1.3 ensures that the reactor vessel

(continued)

BASES

SAFETY LIMITS (continued)	water level is greater than the top of the active irradiated fuel in order to prevent elevated clad temperatures and resultant clad perforations.
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.

Insert B

2. ANF-1125(P)(A), Revision 0, including Supplements 1 and 2, April 1990.
3. CENPD-392-P-A, "10 x 10 SVEA Critical Power Experiments and CPR Correlations: SVEA-96," September 2000
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.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

BASES

BACKGROUND

The APLHGR is a measure of the average LHGR of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the fuel design limits identified in References 1~~, 2, and 3~~ and 2 are not exceeded and that the peak cladding temperature (PCT) during the postulated design basis loss of coolant accident (LOCA) does not exceed the limits specified in 10 CFR 50.46. As a result, core geometry will be maintained by minimizing gross fuel cladding failure due to heatup following a design basis LOCA.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel design limits are presented in References 1 and 2. The analytical methods and assumptions used in evaluating Design Basis Accidents (DBAs) and normal operations that determine APLHGR limits are presented in FSAR, Chapters 4, 6, 15, and 15.F and in References 1~~, 2, 3, 4, and 5.~~ and 2

LOCA analyses are performed to ensure that the specified APLHGR limits are adequate to meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using calculational models that are consistent with the requirements of 10 CFR 50, Appendix K. ~~A complete discussion of the analysis code is provided in References 1 and 5.~~ A complete discussion of the analysis codes is provided in References 1 and 2. The PCT following a postulated LOCA is a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR. For single recirculation loop operation, APLHGR limits are determined when two-loop limits are not bounding.

The APLHGR satisfies Criterion 2 of Reference ~~6~~ 3.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

APLHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution under normal conditions. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

1. NEDC-32115P, "SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 2, June 1993.

2. CE-NPSD-883-P, "Columbia Cycle 16 Reload Licensing Report," March 2001.

3. XN-NF-80-19(A), "Exxon Nuclear Methodology for Boiling Water Reactors," Volumes 2, 2A, 2B, and 2C, September 1982.

2 ~~4~~. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," July 1996.

5. CE-NPSD-801-P, "WNP-2 LOCA Analysis Report," Revision 5, February 2001.

3 ~~6~~. 10 CFR 50.36(c)(2)(ii).

Insert

1. EMF-2361(P)(A) Revision 0, "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP Richland, May 2001

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

BASES

BACKGROUND

MCPR is a ratio of the fuel assembly power that would result in the onset of boiling transition to the actual fuel assembly power. The MCPR Safety Limit (SL) is set such that 99.9% of the fuel rods avoid boiling transition if the limit is not violated (refer to the Bases for SL 2.1.1.2). The operating limit MCPR is established to ensure that no fuel damage results during anticipated operational occurrences insert Reference 1 (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experiences boiling transition (Refs. 1 and 2), the critical power at which boiling transition is calculated to occur has been adopted as a fuel design criterion.

The onset of transition boiling is a phenomenon that is readily detected during the testing of various fuel bundle designs. Based on these experimental data, correlations have been developed to predict critical bundle power (i.e., the bundle power level at the onset of transition boiling) for a given set of plant parameters (e.g., reactor vessel pressure, flow, and subcooling). Because plant operating conditions and bundle power levels are monitored and determined relatively easily, monitoring the MCPR is a convenient way of ensuring that fuel failures due to inadequate cooling do not occur.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the AOOs to establish the operating limit MCPR are presented in the FSAR, Chapters 4, 6, and 15, and References 2 and 3. To ensure that the MCPR SL is not exceeded during any transient event that occurs with moderate frequency, limiting transients have been analyzed to determine the largest reduction in critical power ratio (CPR). The types of transients evaluated are loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest change in CPR (ΔCPR). When the largest ΔCPR is added to the MCPR SL, the required operating limit MCPR is obtained.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The MCPR operating limits derived from the transient analysis are dependent on the operating core flow and power state ($MCPR_f$ and $MCPR_p$, respectively) to ensure adherence to fuel design limits during the worst transient that occurs with moderate frequency as identified in FSAR, Chapters 15 and 15.F.

Insert C

Flow dependent MCPR limits are determined by steady state methods using the three dimensional BWR simulator code (Ref. 2). $MCPR_f$ curves are provided based on the maximum credible flow runout transient for ASD operation (i.e., runout of both loops).

Insert D

Power dependent MCPR limits ($MCPR_p$) are determined by the three dimensional BWR simulator code and the one dimensional transient code (Ref. 2). Due to the sensitivity of the transient response to initial core flow levels at power levels below those at which the turbine stop valve closure and turbine control valve fast closure scram trips are bypassed, high and low flow $MCPR_p$ operating limits are provided for operating between 25% RTP and the previously mentioned bypass power level.

The MCPR satisfies Criterion 2 of Reference 4.

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. Insert E The MCPR operating limits are determined by the larger of the $MCPR_f$ and $MCPR_p$ limits.

APPLICABILITY

The MCPR operating limits are primarily derived from transient analyses that are assumed to occur at high power levels. Below 25% RTP, the reactor is operating at a slow recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 25% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs.

Statistical analyses indicate that the nominal value of the initial MCPR at 25% RTP is expected to be very large. Studies of the variation of limiting transient behavior have

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

The MCPR is required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. It is compared to the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution during normal operation. The 12 hour allowance after THERMAL POWER reaches ≥ 25% RTP is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

Insert F

1. XN-NF-524(A), "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors," Revision 1, November 1983.

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

3. CE-NPSD-883-P, "Columbia Cycle 16 Reload Licensing Report," March 2001.

4. 10 CFR 50.36(c)(2)(ii).

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

BASES

BACKGROUND

The LHGR is a measure of the heat generation rate of a fuel rod in a fuel assembly at any axial location. Limits on the LHGR are specified to ensure that fuel design limits are not exceeded anywhere in the core during normal operation, including anticipated operational occurrences (AOOs). Exceeding the LHGR limit could potentially result in fuel damage and subsequent release of radioactive materials. Fuel design limits are specified to ensure that fuel system damage, fuel rod failure or inability to cool the fuel does not occur during the anticipated operating conditions identified in References 1 and 2.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in evaluating the fuel system design are presented in References 1, 2, 3, 4, 5, and 6. The fuel assembly is designed to ensure (in conjunction with the core nuclear and thermal hydraulic design, plant equipment, instrumentation, and protection system) that fuel damage will not result in the release of radioactive materials in excess of the guidelines of 10 CFR, Parts 20, 50, and 100. The mechanisms that could cause fuel damage during operational transients and that are considered in fuel evaluations are:

a. Rupture of the fuel rod cladding caused by strain from the relative expansion of the UO_2 pellet; and

b. Severe overheating of the fuel rod cladding caused by inadequate cooling.

A value of 1% plastic strain of the fuel cladding has been defined as the limit below which fuel damage caused by overstraining of the fuel cladding is not expected to occur (Reference 7).

Fuel design evaluations have been performed and demonstrate that the 1% fuel cladding plastic strain design limit is not exceeded during continuous operation with LHGRs up to the operating limit specified in the COLR. The analysis also includes allowances for short term transient operation above the operating limit to account for AOOs.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The LHGRs are required to be initially calculated within 12 hours after THERMAL POWER is ≥ 25% RTP and then every 24 hours thereafter. They are compared with the specified limits in the COLR to ensure that the reactor is operating within the assumptions of the safety analysis. The 24 hour Frequency is based on both engineering judgment and recognition of the slowness of changes in power distribution under normal conditions. The 12 hour allowance after THERMAL POWER ≥ 25% RTP is achieved is acceptable given the large inherent margin to operating limits at lower power levels.

REFERENCES

Insert G

e

1. XN-NF-85-67(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload," September 1986.
2. CENPD-287-P-A, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors," July 1996.
3. XN-NF-81-21(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," Revision 1, January 1982.
4. ANF-89-014(P)(A), Revision 1 and Supplements 1 and 2, "Advanced Nuclear Fuels Corporation Generic Mechanical Design for Advanced Nuclear Fuels 9x9-IX and 9x9-9X BWR Reload Fuel," October 1991.
5. CE-NPSD-883-P, "Columbia Cycle 16 Reload Licensing Report," March 2001
6. FSAR, Chapter 4.
7. NUREG-0800, Section II A.2(g), Revision 2, July 1981.
8. 10 CFR 50.36(c)(2)(ii).

BASES

LCO (continued)

c. Increasing the APRM gains to cause the APRM to read greater than 100(%) times MFLPD. This condition is to account for the reduction in margin to the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit.

Framatome ANP

MFLPD is the ratio of the limiting LHGR to the LHGR limit for the specific bundle type. For Siemens fuel, MFDLRX is the equivalent of MFLPD. As power is reduced, if the design power distribution is maintained, MFLPD is reduced in proportion to the reduction in power. However, if power peaking increases above the design value, the MFLPD is not reduced in proportion to the reduction in power. Under these conditions, the APRM gain is adjusted upward or the APRM Flow Biased Simulated Thermal Power — High Function Allowable Value is reduced accordingly. When the reactor is operating with peaking less than the design value, it is not necessary to modify the APRM Flow Biased Simulated Thermal Power — High Function Allowable Value. Adjusting the APRM gain or modifying the Flow Biased Simulated Thermal Power — High Function Allowable Value is equivalent to maintaining MFLPD less than or equal to FRTP, as stated in the LCO.

For compliance with LCO Item b (APRM Flow Biased Simulated Thermal Power — High Function Allowable Value modification) or Item c (APRM gain adjustment), only APRMs required to be OPERABLE per LCO 3.3.1.1, Function 2.b, are required to be modified or adjusted. In addition, each APRM may be allowed to have its gain or Allowable Value adjusted or modified independently of other APRMs that are having their gain or Allowable Value adjusted or modified.

APPLICABILITY

The MFLPD limit, APRM gain adjustment, or APRM Flow Biased Simulated Thermal Power — High Function Allowable Value modification is provided to ensure that the fuel cladding integrity SL and the fuel cladding 1% plastic strain limit are not violated during design basis transients. As discussed in the Bases for LCO 3.2.1, LCO 3.2.2, and LCO 3.2.3, sufficient margin to these limits exists below 25% RTP and, therefore, these requirements are only necessary when the plant is operating at ≥ 25% RTP.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.1.3.1

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. A Frequency of 184 days provides an acceptable level of system average availability over the Frequency and is based on the reliability of the channel (Reference 7).

SR 3.3.1.3.2

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the OPRM System. The 1000 MWD/T Frequency is based on operating experience with LPRM sensitivity changes.

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SR 3.3.1.3.3

The CHANNEL CALIBRATION is a complete check of the instrument loop. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations, consistent with the plant specific setpoint methodology. Calibration of the channel provides a check of the internal reference voltage and the internal processor clock frequency. It also compares the desired trip setpoints with those in processor memory. Since the OPRM is a digital system, the internal reference voltage and processor clock frequency are, in turn, used to automatically calibrate the internal analog to digital converters. The Allowable Values are specified in the (COLR). As noted, neutron detectors are excluded from CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 1000 MWD/T LPRM calibration using the TIPs (SR 3.3.1.3.2). 1130

The Frequency of 24 months is based upon the assumption of the magnitude of equipment drift provided by the equipment supplier (Reference 7).

(continued)
