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November 10, 2000

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Document Control Desk

Subject: Grand Gulf Nuclear Station
Docket No. 50-416
License No. NPF-29
Cycle 12 Reload
Proposed Amendment to the Operating License, LDC-2000-076

GNRO-2000/00084

Gentlemen:

Attached for your review and approval are proposed changes to the Grand Gulf Nuclear Station (GGNS) Technical Specifications (TS). This proposed amendment requests a change to the minimum critical power ratio safety limit (SLMCPR) and changes to the references for the analytical methods used to determine the core operating limits.

This change is required to support the GGNS upcoming Cycle 12 operation. Cycle 12 will be the first cycle of operation with a mixed core of General Electric (GE) GE11 and Siemens Power Corporation (SPC) ATRIUM-10 reload fuel. The proposed amendment reflects a decrease of the two recirculation loop SLMCPR limit from 1.09 to 1.08 with the single recirculation loop SLMCPR limit remaining at 1.10.

The discussion and justification to support the requested amendments are provided in the attachment of this submittal. This amendment request has been reviewed and accepted by the Plant Safety Review Committee and the Safety Review Committee.

The proposed change has been evaluated in accordance with 10CFR50.91(a)(1) using criteria in 10CFR50.92(c) and it has been determined that this change involves no significant hazards considerations. The attachment to this submittal includes the bases for these determinations.

APO1

Additional information regarding the two-loop and single-loop cycle specific SLMCPRs for Cycle 12 was provided by Siemens Power Corporation (SPC). This information isincluded in Attachment 4 of this submittal. The information in Attachment 4 is considered to be proprietary. In accordance with 10 CFR 2.790, an application for withholding this information in whole from public disclosure and the accompanying affidavit by the information owner, SPC, is included as Attachment 4.

The proposed change introduces no new commitments.

Entergy Operations requests NRC approval and issuance of the proposed Technical Specifications changes prior to the Grand Gulf Refueling Outage 11 now scheduled to begin in April 2001. Entergy Operations requests that the amendment go into effect after Operating Cycle 11, but prior to reactor steam dome pressure reaching 785 psig or core flow reaching 10% rated core flow in Cycle 12. Although this request is neither exigent nor emergency, your prompt review is requested.

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 10, 2000.

Very truly yours,

William A Eaton

/as

Attachments:
1. Proposed Technical Specification Change
2. Markup of Current Technical Specification
3. Markup of Technical Specification Bases (Information Only)
4. SPC GGNS Cycle 12 MCPR Safety Limit Analysis

cc:

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ATTACHMENT 1

TO

LETTER NO. GNRO-2000-00084

PROPOSED TECHNICAL SPECIFICATION

AND

RESPECTIVE SAFETY ANALYSES

IN THE MATTER OF AMENDING

LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC.

GRAND GULF NUCLEAR STATION

DOCKET NO. 50-416

DESCRIPTION OF PROPOSED CHANGES

This proposed amendment contains changes to those Technical Specifications (TS) Reactor Core Safety Limits and other sections of the TS required to support Grand Gulf Nuclear Station (GGNS), Cycle 12 operation.

The following Technical Specifications are affected by the proposed changes:

Technical Specifications

2.1.1 Reactor Core Safety Limits

The proposed change revises the Safety Limit MCPR for Two Loop Operation from 1.09 to 1.08. The Single Loop Operation MCPR shall remain at 1.10.

5.6.5 Core Operating Limits Report

The proposed change deletes references to the analytical methods no longer used to determine the core operating limits and adds references to the analytical methods to be used beginning in Cycle 12.

The following Technical Specification Bases are affected by the proposed change. These are provided for information only.

Technical Specification Bases

B2.1.1.1 Fuel Cladding Integrity

The proposed change revises Ref. 6 in the first paragraph to Ref. 3, 5, and 6.

B2.0 References

Updates Reference 2 from XN-NF524(A) to ANF-524(P)(A) and adds References 3 and 5.

B3.2.2 Minimum Critical Power Ratio (MCPR)

Applicable Safety Analysis

Deletes Reference 2 in Paragraph 1, Sentence 1. Adds Reference 2 in Paragraph 2, Sentence 2. Changes Reference 7 to Reference 2 in Paragraph 3, Sentence 1.

References

Changes Reference 2 to XN-NF-80-19(P)(A), Volume 3. Changes Reference 6 to XN-NF-80-19(P)(A), Volume 2. Deletes Reference 7.

BACKGROUND

Grand Gulf Cycle 12 is the first cycle of operation with Siemens Power Corporation (SPC) ATRIUM-10 reload fuel. The current Cycle 12 core design consists of 204 fresh ATRIUM-10 bundles and 596 reload GE11 bundles. While we do not expect this to change, any final core design changes will be evaluated to confirm that the proposed Technical Specification changes remain valid.

The introduction of Atrium-10 fuel supplied by SPC in the upcoming refuelingoutage, requires the use of the new fuel vendor's analytical methods for determining core operating limits and the MCPR Safety Limit. These methods have been approved by the USNRC and are proposed to be listed in TS section 5.6.5.

The proposed Miniumum Critical Power Ratio Safety Limit values for the upcoming operating cycle were developed with Siemens Power Corporation's SLMCPR methodology. The methodology used is found in ANF-524(P)(A), "Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors". The critical power was calculated per EMF-2209(P)(A), "SPCB Critical Power Correlation" and EMF-2245(P)(A), "Application of SPC Critical Power Correlations to Co-Resident Fuel".

BASIS FOR PROPOSED CHANGE

The MCPR Safety Limit is developed to assure compliance with General Design Criterion 10 of 10CFR50 Appendix A. The Bases to Technical Specification 2.1.1 states that "The MCPR SL ensures sufficient conservatism in the operating MCPR limit that, in the event of an Anticipated Operational Occurrence (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".

Entergy performed analyses to determine additive constants and additive constant uncertainties for the Global Nuclear Fuel (GNF) GE11 fuel type for use with the SPC critical power correlation. EOI applied the direct correlation application process described in EMF-2245 (P)(A) with GE11 experimental CPR data. Qualification of Entergy personnel performing these calculations was outlined in Entergy letter to the NRC CNRO-2000-00024, "Entergy Operations, Inc. Implementation of GL 83-11, Supplement 1, for Co-Resident Fuel CPR Calculations", dated August 4, 2000. The results of this analysis concluded that the ANFB-Edge correlation in EMF-1997(P)(A) provided the best fit to the GE11 CPR test data and this correlation was subsequently applied in determining the MCPR safety limit. The SPCB correlation and uncertainties from EMF-2209(P)(A) were applied to the ATRIUM-10 fuel.

In addition, the hydraulic characteristics of the GE11 fuel design have been evaluated in Siemens hydraulic test facility. This test was used to characterize the component pressure drop coefficients of the inlet region, the exit region, and the loss coefficients of the grid spacers as well as the hydraulic resistance of the lower tie plate spring seals. The assembly flow uncertainty associated with a mixed core was applied in this calculation.

With the above inputs, the GGNS MCPR safety limits were developed for the GGNS Cycle 12 core design and expected operation using SPC's NRC approved MCPR safety limit methodology in ANF-524(P)(A). The resulting changes to the GGNS Technical Specifications are included as part of this attachment.

Additional information to support the cycle specific SLMCPR is included in Attachment 4. This attachment summarizes the MCPR Safety Limit Analysis, methodology, and results Note that EOI has chosen to increase the calculated MCPR safety limits by an additional margin to bound any potential cycle-to-cycle variations. The GGNS Cycle 12 core will consist of only GE11 and ATRIUM-10 fuel types. The COLR references in TS section 5.6.5 are being updated to reflect the methods and codes that apply to Cycle 12 GE11 fuel and ATRIUM-10 fuel.

For two-loop operation, a Safety Limit MCPR of 1.08 was demonstrated to be adequate to ensure that 99.9 percent of the rods in the core avoid a boiling transition during the most limiting AOO. For single-loop operation, this assurance is provided by the existing value of 1.10.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Energy Operations, Inc. is proposing that the Grand Gulf Nuclear Station Operating License be amended to modify the Minimum Critical Power Ratio (MCPR) safety limits reported in Technical Specification 2.1.1.2 and the references listed in Technical Specification 5.6.5. The proposed changes are necessary in order to reflect the NRC approved methods used in determining the GGNS Cycle 12 core operating limits and reflect the safety limit changes for the mixed core.

An evaluation of the proposed change has been performed in accordance with 10CFR50.91(a)(1) regarding no significant hazards considerations using the standards in 10CFR50.92(c). A discussion of these standards as they relate to this amendment request follows:

1. **Will operation of the facility in accordance with this proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The Minimum Critical Power Ratio (MCPR) safety limit is defined in the Bases to Technical Specification 2.1.1 as that limit which "ensures that during normal operation and during Anticipated Operational Occurrences (AOOs), at least 99.9% of the fuel rods in the core do not experience transition boiling." The MCPR safety limit satisfies the requirements of General Design Criterion 10 of Appendix A to 10 CFR 50 regarding acceptable fuel design limits. The MCPR safety limit is re-evaluated for each reload using NRC-approved methodologies. The analyses for GGNS Cycle 12 have concluded that a two-loop MCPR safety limit of 1.08, based on the application of Siemens Power Corporation's NRC-approved MCPR safety limit methodology, will ensure that this acceptance criterion is met . For single-loop operation, a MCPR safety limit of 1.10 (unchanged), also ensures that this acceptance criterion is met.

In addition to the MCPR safety limit, core operating limits are established to support the Technical Specification 3.2 requirements which ensure that the fuel design limits are not exceeded during any conditions of normal operation or in the event of any anticipated operational occurrences (AOO). The methods used to determine the core operating limits for each operating cycle are based on methods previously found acceptable by the NRC and listed in TS section 5.6.5. A change to TS section 5.6.5 is requested to include the SPC methods in the list of NRC approved methods applicable to GGNS. These NRC approved methods will continue to ensure that acceptable operating limits are established to protect the fuel cladding integrity during normal operation and in the event of an AOO.

The requested Technical Specification changes do not involve any plant modifications or operational changes that could affect system reliability or performance or that could affect the probability of operator error. The requested changes do not affect any postulated accident precursors, do not affect any accident mitigating systems, and do not introduce any new accident initiation mechanisms.

Therefore, these changes to the Minimum Critical Power Ratio (MCPR) safety limit and to the list of methods used to determine the core operating limits do not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. **Will operation of the facility in accordance with this proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The ATRIUM-10 fuel to be used in Cycle 12 is of a design compatible with the co-resident GE-11. Therefore, the introduction of ATRIUM-10 fuel into the Cycle 12 core will not create the possibility of a new or different kind of accident. The proposed changes do not involve any new modes of operation, any changes to setpoints, or any plant modifications. The proposed revised MCPR safety limits have accounted for the mixed fuel core and have been shown to be acceptable for Cycle 12 operation. Compliance with the criterion for incipient boiling transition continues to be ensured. The core operating limits will continue to be developed using NRC approved methods which also account for the mixed fuel core design. The proposed MCPR safety limits or methods for establishing the core operating limits do not result in the creation of any new precursors to an accident.

Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Will operation of the facility in accordance with this proposed change involve a significant reduction in a margin of safety?

The MCPR safety limits have been evaluated in accordance with Siemens Power Corporation's NRC-approved cycle-specific safety limit methodology to ensure that during normal operation and during Anticipated Operational Occurrences (AOO's) at least 99.9% of the fuel rods in the core are not expected to experience transition boiling. On this basis, the implementation of this Siemens Power Corporation methodology does not involve a significant reduction in a margin of safety.

Therefore, this change does not involve a significant reduction in the margin of safety.

Therefore, based on the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change does not involve a significant hazards consideration.

ENVIRONMENTAL IMPACT EVALUATION

Pursuant to 10CFR51.22(b), an evaluation of the proposed amendment has been performed to determine whether or not it meets the criteria for categorical exclusion set forth in 10CFR 51.22 (c) (9) of the regulations. The basis for this determination is as follows:

1. The proposed license amendment does not involve a significant hazards consideration as described previously in the evaluation.

2. As discussed in the significant hazards evaluation, this change does not result in a significant change or significant increase in the radiological doses for any Design Basis Accident. The proposed license amendment does not result in a significant change in the types or a significant increase in the amounts of any effluents that may be released off-site.

3. The proposed license amendment does not result in a significant increase to the individual or cumulative occupational radiation exposure because this change does not change the number of fuel rods experiencing boiling transition during Anticipated Operational Occurrences (AOOs) beyond the current limit of "at least 99.9% of the fuel rods in the core do not experience transition boiling".

ATTACHMENT 2

TO GNRO-2000-00084

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:

MCPR shall be ≥ ~~1.09~~ 1.08 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.

(continued)

5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR) (continued)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

INSERT

1. XN-NF-79-71(P), "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," Exxon Nuclear Company, Inc., Richland, WA.

2. XN-NF-80-19(P)(A), Volume 1, "Exxon Nuclear Methodology for Boiling Water Reactors-Neutronic Methods for Design and Analysis," Exxon Nuclear Company, Inc., Richland, WA.

3. XN-NF-80-19(P)(A), Volume 1, "Advanced Nuclear Fuels Methodology for Boiling Water Reactors: Benchmark Results for the CASMO-3G/MICROBURN-B Calculation Methodology," Advanced Nuclear Fuels Corporation, Richland, WA.

4. XN-NF-80-19(P)(A), Volume 3, "Exxon Nuclear Methodology for Boiling Water Reactors THERMEX: Thermal Limits Methodology Summary Description," Exxon Nuclear Company, Inc., Richland, WA.

5. ANF-913(P)(A), Volume 1, "COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analysis," Advanced Nuclear Fuels Corporation, Richland, WA.

6. ANF-1125(P)(A), "ANFB Critical Power Correlation," Advanced Nuclear Fuels Corporation, Richland, WA.

7. XN-NF-84-105(P)(A), Volume 1, "XCOBRA-T: A Computer Code for BWR Transient Thermal Hydraulic Core Analysis," Exxon Nuclear Company, Inc., Richland, WA.

8. XN-NF-573(P), "RAMPEX Pellet-Clad Interaction Evaluation Code for Power Ramps," Exxon Nuclear Company, Inc., Richland, WA.

9. XN-NF-81-58(P)(A), "RODEX2: Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, Inc., Richland, WA.

(continued)

5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR) (continued)

10. XN-NF-85-74(P)(A), "RODEX2A (BWR): Fuel Rod Thermal-Mechanical Response Evaluation Model," Exxon Nuclear Company, Inc., Richland, WA.

11. XN-CC-33(P)(A), "HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option," Exxon Nuclear Company, Inc., Richland, WA.

12. XN-NF-825(P)(A), "BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$ for Plant Operation Within the Extended Operating Domain," Exxon Nuclear Company, Inc., Richland, WA.

13. XN-NF-81-51(P)(A), "LOCA-Seismic Structural Response of an Exxon Nuclear Company BWR Jet Pump Fuel Assembly," Exxon Nuclear Company, Inc., Richland, WA.

14. XN-NF-84-97(P)(A), "LOCA-Seismic Structural Response of an ENC 9x9 BWR Jet Pump Fuel Assembly," Advanced Nuclear Fuels Corporation, Richland, WA.

15. XN-NF-86-37(P), "Generic LOCA Break Spectrum Analysis for BWR/6 Plants," Exxon Nuclear Company, Inc., Richland, WA.

16. XN-NF-82-07(P)(A), "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," Exxon Nuclear Company, Inc., Richland, WA.

17. XN-NF-80-19(A), Volumes 2, 2A, 2B, & 2C, "Exxon Nuclear Methodology for Boiling Water Reactors EXEM BWR ECCS Evaluation Model," Exxon Nuclear Company, Inc., Richland, WA.

18. XN-NF-79-59(P)(A), "Methodology for Calculation for Pressure Drop in BWR Fuel Assemblies," Exxon Nuclear Company, Inc., Richland, WA.

19. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR-II) with exception to the misplaced fuel bundle analyses as discussed in GNRO-96/00087 and the generic MCPR Safety Limit analysis as discussed in GNRO-96/00100, letters from C. R. Hutchinson to USNRC.

20. J11-02863SLMCPR, Revision 1, "GGNS Cycle 9 Safety Limit MCPR Analysis."

21. NEDO-32339-A, "Reactor Stability Long Term Solution: Enhanced Option I-A," and Supplements 1-4.

(continued)

INSERT

1. XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model", Exxon Nuclear Company, Richland, WA.

2. XN-NF-85-67(P)(A), "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel", Exxon Nuclear Company, Richland, WA.

3. EMF-85-74(P) Supplement 1 (P)(A) and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model", Siemens Power Corporation, Richland, WA.

4. ANF-89-98(P)(A), "Generic Mechanical Design Criteria for BWR Fuel Designs", Advanced Nuclear Fuels Corporation, Richland, WA.

5. EMF-93-177(P)(A), "Mechanical Design for BWR Fuel Channels", Siemens Power Corporation, Richland, WA.

6. XN-NF-80-19(P)(A) Volume 1, "Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis", Exxon Nuclear Company, Richland, WA.

7. XN-NF-80-19(P)(A) Volume 4, "Exxon Nuclear Methodology for Boiling Water Reactors: Application for the ENC Methodology to BWR Reloads", Exxon Nuclear Company, Richland, WA.

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

9. XN-NF-80-19(P)(A) Volume 3, "Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description", Exxon Nuclear Company, Richland, WA.

10. XN-NF-84-105(P)(A), "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis", Exxon Nuclear Company, Richland, WA.

11. ANF-524(P)(A), "ANF Critical Power Methodology for Boiling Water Reactors", Advanced Nuclear Fuels Corporation, Richland, WA.

12. ANF-913(P)(A) Volume 1, "CONTRANSA2: A Computer Program for Boiling Water Reactor Transient Analysis", Advanced Nuclear Fuels Corporation, Richland, WA.

13. XN-NF-825(P)(A), "BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$ for Plant Operations within the Extended Operating Domain", Exxon Nuclear Company, Richland, Wa.

14. ANF-1358(P)(A), "The Loss of Feedwater Heating Transient in Boiling Water Reactors", Advanced Nuclear Fuels Corporation, Richland, WA.

15. EMF-1997(P)(A), "ANFB-10 Critical Power Correlation", Siemens Power Corporation, Richland, WA.

16. EMF-1997(P) Supplement 1 (P)(A), "ANFB-10 Critical Power Correlation: High Local Peaking Results", Siemens Power Corporation, Richland, WA.

17. EMF-2209(P)(A), "SPCB Critical Power Correlation", Siemens Power Corporation, Richland, WA.

18. EMF-2245(P)(A), "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel", Siemens Power Corporation, Richland, WA.

19. XN-NF-80-19(P)(A) Volumes 2, 2A, 2B, And 2C, "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model", Exxon Nuclear Company, Richland, WA.

20. ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model", Advanced Nuclear Fuels, Richland, WA.

21. ANF-91-048(P)(A) Supplements 1 and 2, "BWR Jet Pump Model Revision for RELAX", Siemens Power Corporation, Richland, WA.

22. XN-CC-33(A), "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option Users Manual", Exxon Nuclear Company, Richland, WA.

23. EMF-CC-074(P)(A), Volume 4, "BWR Stability Analysis Assessment of STAIF with Input from MICROBURN-B2", Siemens Power Corporation, Richland, WA.

24. EMF-2292(P)(A), "ATRIUM-10 Appendix K Spray Heat Transfer Coefficients", Siemens Power Corporation, Richland, WA.

25. NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR-II) with exception to the misplaced fuel bundle analyses as discussed in GNRO-96/00087 and the generic MCPR Safety Limit analysis as discussed in GNRO-96/00100, letters from C. R. Hutchinson to USNRC.

26. Attachment 4 to GNRO-2000-00084, “Siemens Power Corporation Grand Gulf Cycle 12 Safety Limit MCPR Analysis”.

ATTACHMENT 3

TO GNRO-2000-00084

MARKUP OF TECHNICAL SPECIFICATION BASES

FOR INFORMATION ONLY

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.

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 an MCPR SL 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 SL.

<u>2.1.1.1 Fuel Cladding Integrity</u>

The use of the fuel vendor's critical power correlations ~~is~~ are valid for critical power calculations at pressures ≥ 785 psig and core flows ≥ 10% of rated (Ref. ~~6~~ 3, 5 and 6). 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:

> Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flow will always be > 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lb/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus the bundle flow with a 4.5 psi driving head will be > 28 x 10^3 lb/hr. Full scale

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER > 50% RTP. Thus a THERMAL POWER limit of 25% RTP for reactor pressure < 785 psig is conservative. Because of the design thermal hydraulic compatibility of the reload fuel designs with the cycle 1 fuel, this justification and the associated low pressure and low flow limits remain applicable for future cycles of cores containing these fuel designs.

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 2
in the critical power correlation. Reference ~~6~~ describes the methodology used in determining the MCPR SL.

The calculated MCPR safety limit is reported to the customary three significant digits (i.e., X.XX); the MCPR operating limit is developed based on the calculated MCPR safety limit to ensure that at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

The fuel vendor's 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 being estimated. As long as the core pressure and flow are within the range of validity of the correlations, the assumed reactor conditions used in defining the SL introduce conservatism into the limit because bounding high radial power factors and bounding flat local peaking distributions are used to estimate the number of rods in boiling transition. These conservatisms and the

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.

INSERT 1

2. ~~XN-NF524(A), Revision 2, April 1989.~~

3. ~~Not used.~~

4. 10 CFR 100.

INSERT 2

5. ~~Not used.~~

6. NEDE-24011-P-A, GESTAR-II.

INSERT 1

2. ANF-524 (P)(A), Revision 2, Supplements 1 and 2, November 1990.

3. EMF-2209 (P)(A), Revision 1, July 2000.

INSERT 2

5. Letter: CEXO-2000-00293, J. B. Lee (EOI) to K. V. Walters (SPC), "Grand Gulf Nuclear Station Unit 1 and Riverbend Station Unit 1, Reload Transition Data – GE11 Additive Constants", July 25, 2000.

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 (AOOs). Although fuel damage does not necessarily occur if a fuel rod actually experiences boiling transition (Ref. 1), 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 UFSAR, Chapters 4, 6, and 15, and References 2, 3, 4, and 5. 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 (Refs. 3, 4, and 5). Flow dependent MCPR limits are determined by steady state thermal hydraulic methods using the three dimensional BWR simulator code (Ref. 6) and the multi channel thermal hydraulic code (Ref. 2). $MCPR_f$ curves are provided based on the maximum credible flow runout transient for Loop Manual operation. The result of a single failure or single operator error during Loop Manual operation is the runout of only one loop because both recirculation loops are under independent control.

Power dependent MCPR limits ($MCPR_p$) are determined by the three dimensional BWR simulator code and the one dimensional transient code (Ref. ~~7~~ 2). The $MCPR_p$ limits are established for a set of exposure intervals. The limiting transients are analyzed at the limiting exposure for each interval. 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 the NRC Policy Statement.

LCO

The MCPR operating limits specified in the COLR are the result of the Design Basis Accident (DBA) and transient analysis. 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.

(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

1. NUREG-0562, "Fuel Failures As A Consequence of Nucleate Boiling or Dry Out," June 1979.

INSERT 1

2. ~~NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR II)~~.

3. UFSAR, Chapter 15, Appendix 15B.

4. UFSAR, Chapter 15, Appendix 15C.

5. UFSAR, Chapter 15, Appendix 15D.

INSERT 2

6. ~~NEDE-30130-P-A, Steady State Nuclear Methods~~.

~~7. NEDO-24154, Qualification of the One Dimensional Core Transient Model for Boiling Water Reactors~~.

INSERT 1

2. 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.

INSERT 2

6. XN-NF-80-19(P)(A) Volume 1, "Exxon Nuclear Methodology for Boiling Water Reactors – Neutronic Methods for Design and Analysis" March 1983 (As Supplemented).

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is H. Donald Curet. I am Manager, Product Licensing, for Siemens Power Corporation ("SPC"), and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by SPC to determine whether certain SPC information is proprietary. I am familiar with the policies established by SPC to ensure the proper application of these criteria.

3. I am familiar with the SPC information included in the Attachment transmitted by letter KVW:00:226 which is referred to herein as "Document." Information contained in this Document has been classified by SPC as proprietary in accordance with the policies established by SPC for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by SPC and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in the Document be withheld from public disclosure.

6. The following criteria are customarily applied by SPC to determine whether information should be classified as proprietary:

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9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

____

SUBSCRIBED before me this 2nd day of November, 2000.

Amy R. Nixon

Amy R. Nixon
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 12/06/03

AMY R. NIXON
NOTARY PUBLIC
STATE OF WASHINGTON
COMMISSION EXPIRES DECEMBER 6, 2003