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May 23, 2001

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

Attention: Document Control Desk

Subject: Cycle 11 Reload
Proposed Amendment to the Operating License, LAR 2000-28

River Bend Station, Unit 1
Docket No. 50-458
License No. NPF-47

File Nos. G9.5, G9.42

RBG-45738
RBF1-01-0111
RBEXEC-01-025

Gentlemen:

Attached for your review and approval are proposed changes to the River Bend Station, Unit 1 (RBS) 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 RBS upcoming Cycle 11 operation. Cycle 11 will be the first cycle of operation with a mixed core of General Electric (GE) GE11 and Framatome ANP Richland, Inc. (FRA-ANP) ATRIUM-10 reload fuel. The proposed amendment reflects a decrease of the two recirculation loop SLMCPR limit from 1.12 to 1.08 and a decrease of the single recirculation loop SLMCPR limit from 1.13 to 1.10.

The discussion and justification to support the requested amendments are provided in the attachment to this submittal. This request is similar to the Grand Gulf Nuclear Station proposed TS change dated November 10, 2000[1]. This amendment request has

[1] W. A. Eaton to NRC, GNRO-2000/00084: Cycle 12 Reload Proposed Amendment to the Operating License, LDC-2000-076

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been reviewed and accepted by the Facility 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.

Additional information regarding the two-loop and single-loop cycle specific SLMCPRs for Cycle 11 was provided by Framatome ANP Richland, Inc. (FRA-ANP). This information is included 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, FRA-ANP, 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 River Bend Refueling Outage 10 now scheduled to begin in September 2001. Entergy Operations requests that the amendment go into effect after Operating Cycle 10, but prior to reactor steam dome pressure reaching 785 psig or core flow reaching 10% rated core flow in Cycle 11. 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 May 23, 2001.

Very truly yours,

Randall K. Edge

RKE/MAK/as

Attachments:
1. Proposed Technical Specification Change
2. Markup of Current Technical Specification
3. Markup of Technical Specification Bases (Information Only)
4. FRA-ANP RBS Cycle 11 MCPR Safety Limit Analysis
5. Additional Information

cc:

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ATTACHMENT 1

TO

LETTER NO. RBG-45738

PROPOSED TECHNICAL SPECIFICATION

AND

RESPECTIVE SAFETY ANALYSES

IN THE MATTER OF AMENDING

LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

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 River Bend Station, Unit 1 (RBS), Cycle 11 operation. Cycle 11 will be RBS's first cycle of operation with Framatome ANP, Richland (FRA-ANP) ATRIUM-10 fuel. Cycle specific reviews will be performed for future cycles to determine the continued applicability of these changes.

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 1.12 to 1.08 and the Single Loop Operation MCPR from 1.13 to 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 11. In addition, an editorial clarification has been made to 5.6.5 a. 5) to include the RPS Instrumentation function applicable to this COLR limit.

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
Changes to this section are made to reflect the use of a different fuel vendor.

B2.1.1.2 MCPR
Changes to this section are made to reflect the use of a different fuel vendor.

B2.0 References
Updates References to reflect the changes made in B2.1.1.1 and B2.1.1.2 above.

B3.2.2 Minimum Critical Power Ratio (MCPR)

Applicable Safety Analysis
Changes to this section are made to reflect the use of a different fuel vendor.

References
Updates References to reflect the changes made in B3.2.2 above.

BACKGROUND

River Bend Cycle 11 is the first cycle of operation with Framatome ANP Richland, Inc. (FRA-ANP) ATRIUM-10 reload fuel. The current Cycle 11 core design consists of 200 fresh ATRIUM-10 bundles and 424 reload GE11 bundles. The ATRIUM-10 fuel assemblies include depleted uranium in their design. Further information concerning the Cycle 11 core design is located in Attachment 5 of this letter. 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. Core designs for future cycles will be evaluated for the continued applicability of these changes.

The introduction of Atrium-10 fuel supplied by FRA-ANP in the upcoming refueling outage, 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 Minimum Critical Power Ratio Safety Limit values for the upcoming operating cycle were developed with Framatome ANP Richland, Inc.'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 fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling".

Entergy performed analyses to determine additive constants and additive constant uncertainties for the Global Nuclear Fuel (GNF) GE11 fuel type for use with the FRA-ANP critical power correlation. Entergy 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 RBS MCPR safety limits were developed by FRA-ANP for the RBS Cycle 11 core design and expected operation using FRA-ANP's NRC approved MCPR safety limit methodology in ANF-524(P)(A). The resulting changes to the RBS Technical Specifications are included in Attachment 2 of this letter.

The MCPR safety limit is determined by a statistical convolution of all the uncertainties associated with the calculation of thermal margin. Both fuel-related (which may vary from cycle to cycle) and non-fuel related (which are characteristics of the reactor system) uncertainties are used in the calculation. A Monte Carlo method is used to simulate a variety of reactor states around a base state, where the reactor states are determined by randomly varying the reactor conditions according to the magnitude of the associated uncertainty. Each of the fuel rods in the core is evaluated to determine if it is in boiling transition. The rods in boiling transition for each bundle are summed over the entire core to determine the number of rods expected to be in boiling transition for the reactor state for a given Monte Carlo trial. The procedure is repeated until a sufficient number of trials have been performed to adequately determine the expected number of rods in boiling transition. Using a non-parametric procedure, the expected number of rods in boiling transition is determined from the number of Monte Carlo trials and the distribution of the number of rods in boiling transition is determined from the Monte Carlo calculation. The use of a non-parametric procedure avoids the need to assume any particular shape for the distribution of the number of rods in boiling transition. The River Bend Cycle 11 MCPR safety limit analysis used 1000 Monte Carlo trials.

Application of the NRC-approved methodology (Reference: 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.) for River Bend Station Unit 1 Cycle 11 shows that a 1.08 MCPR safety limit is supported for two-loop operation. The single-loop operation MCPR safety limit analysis supports a 1.10 MCPR safety limit. That is, 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 Anticipated Operating Occurrence (AOO). For single-loop operation, this assurance is provided by a Safety Limit MCPR of 1.10.

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 Entergy has chosen to increase the calculated MCPR safety limits by an additional margin to bound any potential cycle-to-cycle variations. The RBS Cycle 11

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 11 GE11 fuel and ATRIUM-10 fuel. Attachment 5 provides justification for the inclusion of these proposed references in TS section 5.6.5.

The previous fuel vendor's methodology had conservatively assumed that the Misplaced Bundle Accident was an Anticipated Operational Occurrence (AOO). The current fuel vendor's methodology assumes that this is an infrequent event. The RBS FSAR Section 15.4.7 considers this event to be an infrequent event. Therefore, this is not a change to the current licensing basis. The previous fuel vendor's methodology was overly conservative when compared to the RBS FSAR. This allows the event to be analyzed consistently as in the event classification as described in the RBS FSAR and is also consistent with the Grand Gulf Nuclear Station (GGNS) treatment.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Entergy Operations, Inc. is proposing that the River Bend Station, Unit 1 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. In addition, an editorial clarification was made to Technical Specification 5.6.5. The proposed changes are necessary in order to reflect the NRC approved methods used in determining the RBS Cycle 11 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 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 RBS Cycle 11 have concluded that a two-loop MCPR safety limit of 1.08, based on the application of Framatome ANP Richland, Inc.'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, 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 FRA-ANP methods in the list of NRC approved methods applicable to RBS. 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 11 is of a design compatible with the co-resident GE-11. Therefore, the introduction of ATRIUM-10 fuel into the Cycle 11 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 11 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 Framatome ANP Richland, Inc.'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 Framatome ANP Richland, Inc. 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 RBG-45738

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 ≤ 23.8% RTP.

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

MCPR shall be ≥ ~~1.12~~ 1.08 for two recirculation loop operation or ≥ ~~1.13~~ 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:

2.2.1 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.2 Within 2 hours:

2.2.2.1 Restore compliance with all SLs; and

2.2.2.2 Insert all insertable control rods.

2.2.3 Within 24 hours, notify the plant manager and the corporate executive responsible for overall plant nuclear safety.

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and process control program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

1) LCO 3.2.1, Average Planar Linear Heat Generation Rate (APLHGR).
2) LCO 3.2.2, Minimum Critical Power Ratio (MCPR) (including power and flow dependent limits).
3) LCO 3.2.3, Linear Heat Generation Rate (LHGR) (including power and flow dependent limits).
4) LCO 3.2.4, Fraction of Core Boiling Boundary (FCBB)
5) LCO 3.3.1.1, RPS Instrumentation (RPS), Function 2.b
6) LCO 3.3.1.3, Periodic Based Detection System (PBDS)

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.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

INSERT

1) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version);

2) NEDC-32489P (April 1996), "T-Factor Setdown Elimination Analysis for River Bend Station" (for power and flow dependent limits methodology only as evaluated and approved by Safety Evaluation and License Amendment 100 dated October 10, 1997).

3) NEDO-32339P-A, "Reactor Stability Long-Term Solution: Enhanced Option I-A," including Supplements 1 through 4 (latest revisions dated through April 1998).

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

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

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. 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.

6. 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.

7. 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.

8. 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.

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

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

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

12. XN-NF-825(P)(A), Supplement 2, "BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$ for Plant Operations within the Extended Operating Domain", Exxon Nuclear Company, Richland, Wa.

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

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

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

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

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

18. 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.

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

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

21. 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.

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

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

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

ATTACHMENT 3

TO RBG-45738

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.

2.1.1.1 Fuel Cladding Integrity

INSERT 1

~~General Electric (GE) critical power correlations are applicable for all~~ critical power calculations at pressures ≥ 785 psig and core flows ≥ 10% of rated flow. For operation at low pressures or low flows, another basis is used, as follows:

Ref. 2, 7 and 8

> Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be > 4.5 psi. Analyses (Ref. 2) 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 ATLAS test data taken at pressures from 14.7 psia to 800 psia

(continued)

BASES

APPLICABLE SAFETY ANALYSES

2.1.1.1 Fuel Cladding Integrity (continued)

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% of original RTP. Thus, a THERMAL POWER limit of 23.8% RTP for reactor pressure < 785 psig is conservative.

INSERT 2

2.1.1.2 MCPR

INSERT 3

The fuel cladding integrity SL is set such that no significant fuel damage is calculated to occur if the limit is not violated. Since the parameters that result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions that result in the onset of transition boiling have been used to mark the beginning of the region in which fuel damage could occur. Although it is recognized that the onset of transition boiling would not result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity SL is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition, considering the power distribution within the core and all uncertainties.

The MCPR SL is determined using a statistical model that combines all the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric critical power correlations. Details of the fuel cladding integrity SL calculation are given in Reference 2. Reference 2 also includes a tabulation of the uncertainties used in the determination of the MCPR SL and of the nominal values of the parameters used in the MCPR SL statistical analysis.

(continued)

BASES

SAFETY LIMIT VIOLATIONS

2.2.2 (continued)

with the SL within 2 hours. The 2 hour Completion Time ensures that the operators take prompt remedial action and also ensures that the probability of an accident occurring during this period is minimal.

2.2.3

If any SL is violated, the General Manager and the Vice President shall be notified within 24 hours. The 24 hour period provides time for plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to the senior management.

2.2.4

If any SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 5). A copy of the report shall also be provided to the General Manager and the Vice President.

2.2.5

If any SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.

REFERENCES

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

2. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel, GESTAR-II," (latest approved revision).

3. 10 CFR 50.72.

4. 10 CFR 100.

INSERT 4 → 5. 10 CFR 50.73.

INSERT 1

The use of the fuel vendor's critical power correlations are valid for

INSERT 2

Because of the design thermal hydraulic compatibility of the reload fuel designs with the cycle 10 fuel, this justification and the associated low pressure and low flow limits remain applicable for future cycles of cores containing these fuel designs.

INSERT 3

The MCPR SL ensures sufficient conservatism in the operating limit MCPR 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. 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 conservatisims and the inherent accuracy of the fuel vendor's correlation provide a reasonable degree of assurance that 99.9% of the rods in the core would not be susceptible to transition boiling 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 4

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

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

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

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. Flow dependent MCPR limits ($MCPR_f$) are determined by steady state thermal hydraulic methods using the three dimensional BWR simulator code (Ref. 5). $MCPR_f$ curves are provided based on the maximum credible flow runout transient for Non Loop Manual operation. Non Loop Manual operation bounds Loop Manual because Non Loop Manual operation can result in a more severe flow runout transient. 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. Non Loop Manual operational modes allow simultaneous runout of both loops because a single controller regulates core flow.

INSERT 1

Power dependent MCPR limits ($MCPR_p$) are determined by the three dimensional BWR simulator code and the one dimensional transient code (Ref. ~~6~~ 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 23.8% RTP and the previously mentioned bypass power level.

INSERT 2

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 23.8% RTP, the reactor is operating at a slow recirculation pump speed and the moderator void ratio is small. Surveillance of thermal limits below 23.8% RTP is unnecessary due to the large inherent margin that ensures that the MCPR SL is not exceeded even if a limiting transient occurs.

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 ≥ 23.8% 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 ≥ 23.8% RTP is acceptable given the large inherent margin to operating limits at low power levels.

REFERENCES

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

INSERT 3

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

3. USAR, Chapter 4, Appendix 4B.

4. USAR, Chapter 15, Appendix 15B.

INSERT 4

5. ~~NEDO-30130-A, "Steady State Nuclear Methods," May 1985~~.

6. ~~NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," October 1978~~.

INSERT 1

and the multi channel thermal hydraulic code (Ref. 2).

INSERT 2

The MCPR limits are established for a set of exposure intervals. The limiting transients are analyzed at the limiting exposure for each interval.

INSERT 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.

INSERT 4

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).

ATTACHMENT 5

TO RBG-45738

ADDITIONAL INFORMATION.

The Cycle 11 core is composed of the following fuel types:

Fuel Type	Batch	Total Bundles	Irradiation History
Atrium-10	F	200	Fresh
GE-11	O	208	Once Burnt
GE-11	T	216	Twice Burnt

The full core loading map is provided in Figure 5.1. While we do not expect this information to change, any final core design changes will be evaluated to confirm that the proposed Technical Specification changes remain valid.

The River Bend Cycle 11 core loading continues to use a scatter loading strategy in which the central portion of the core is loaded with fresh and once burned fuel in a checkerboard configuration. The periphery region is loaded with low reactivity twice burned fuel. This fuel management strategy maintains power and reactivity limits while maximizing cycle length. This strategy is similar to pervious cycles except the cycle 11 fuel batch size (32 % of core) is significantly lower than the cycle 9 and 10 cores. Cycles 9 and 10 contained additional fresh fuel in order to compensate for the early discharge of the cycle 8 reload batch. The Cycle 8 reload batch was discharged during Refueling Outage eight (RF08) due to the presence of accelerated clad corrosion due to elevated crud thicknesses. The cycle 9 and 10 cores contained 49% and 38% fresh fuel, respectively. This is significantly above the more typical reload core size that was employed in cycle 8 (34%).

The MCPR safety limit, determined by General Electric for cycles 8-10 varied from 1.102 to 1.128. This illustrates the sensitivity of the safety limit to relatively large variations in fuel management. The cycle 11 safety limit, determined by FRA-ANP methodology is 1.06. Consistent with the Grand Gulf assessment (Ref: GNRO-2001/00025, Grand Gulf Response to RAI), the change in the MCPR safety limit for River Bend Cycle 11 is largely attributed to differences in methodology.

Figure 5.1: RBS Cycle 11 Core Loading Map

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Fuel Type	Batch	Total Bundles	Irradiation History
Atrium-10	F	200	Fresh
GE-11	O	208	Once Burnt
GE-11	T	216	Twice Burnt

The Framatome ANP (FRA-ANP) topical reports are listed in the remainder of this attachment. It should be noted that NEDE-24011-P-A has been retained due to the presence of GE11 fuel in the core. This reference was used to develop the LHGR (LCO 3.2.3) and APLHGR (LCO 3.2.1) operating limits for the existing GE fuel.

FRAMATOME ANP

March 22, 2001
KVW:01:090

File # 488-83
788-83

Mr. J. B. Lee
Fuel Fabrication Coordinator
Entergy Operations, Inc.
Echelon One
1340 Echelon Parkway
Jackson, MS 39213

Dear Mr. J. B. Lee:

Technical Specification and COLR References for
Grand Gulf Nuclear Station and River Bend Station
GEXI: 2001-00030
RBC-49449

Attached are lists of NRC-approved topical reports to be used as COLR references for Grand Gulf Nuclear Station and River Bend Station. For each topical report listed, a justification is provided for use of the topical report as a reference.

If you have any questions or require additional information, please call me.

Very truly yours,

K. V. Walters
Project Manager

Enclosure

cc: Echelon: J. B. Lee (Original plus one copy), w/ encl.
D. K. Stringer, w/ encl.
R. M. Wilkins, w/ encl.

cc: GGNS: K. L. Walker, w/ encl.
M. D. Withrow, w/ encl.
GGNS Central File, w/ encl. [5]

cc: RBS: (c/o Donna Fancher, GBS-30)
H. A. Goodman w/ encl.
P. A. Sicard, w/ encl.
P. Vo w/ encl.
RBS-PPF G25.4.3, w/ Encl. [5]

Framatome ANP Richland, Inc.

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Richland, WA 99352

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BWR Approved Topical Reports for GGNS and RBS Technical Specifications and COLR References

Report	Applicable LCO	Justification
XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, *RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model*, Exxon Nuclear Company, March 1984.	3.2.1 3.2.2 3.2.3	Provides an analytical capability to predict BWR fuel thermal and mechanical conditions for normal core operation and to establish initial conditions for power ramping, non-LOCA and LOCA analyses.
XN-NF-85-67(P)(A) Revision 1, *Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel*, Exxon Nuclear Company, September 1986.	3.2.3	Describes the process used to develop linear heat generation rates for fuel designs.
EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), *RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model*, Siemens Power Corporation, February 1998.	3.2.3	Extends the exposure limit of the RODEX2A code which is a version of RODEX2 that includes a fission gas release model specific to BWR fuel designs.
ANF-89-98(P)(A) Revision 1 and Supplement 1, *Generic Mechanical Design Criteria for BWR Fuel Designs*, Advanced Nuclear Fuels Corporation, May 1995.	3.2.3	Establishes a set of design criteria which assures that BWR fuel will perform satisfactorily throughout its lifetime.
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.	3.2.1 3.2.2 3.2.3	Development of BWR core analysis methodology which comprises codes for fuel neutronic parameters and assembly burnup calculations, reactor core simulation diffusion theory calculations, core and channel hydrodynamic stability predictions, and producing input for nuclear plant transients. Subsequently approved codes or methodologies have superceded portions of this report. Applicable portions include CRDA, and methodology to determine neutronic reactivity parameters, void reactivity, Doppler reactivity, scram reactivity, delayed neutron fraction, and prompt neutron lifetime.
XN-NF-80-19(P)(A) Volume 4 Revision 1, *Exxon Nuclear Methodology for Boiling Water Reactors: Application of the ENC Methodology to BWR Reloads*, Exxon Nuclear Company, June 1986.	3.2.1 3.2.2 3.2.3	Summarizes the types of BWR licensing analyses performed, identifies the methodologies used.

Report	Applicable LCO	Justification
EMF-2158(P)(A) Revision 0, *Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2*, Siemens Power Corporation, October 1999.	3.2.2 3.2.3	Describes the reactor core simulator code MICROBURN-B2 and the lattice physics code CASMO-4.
XN-NF-80-19(P)(A) Volume 3 Revision 2, *Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description*, Exxon Nuclear Company, January 1987.	3.2.2	Provides overall methodology for determining a MCPR operating limit.
XN-NF-84-105(P)(A) Volume 1 and Volume 1 Supplements 1 and 2, *XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis*, Exxon Nuclear Company, February 1987.	3.2.2	Provides a capability to perform analyses of transient heat transfer behavior in BWR assemblies.
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.	3.2.2	Provides a methodology for the determination of thermal margins, specifically the MCPR safety limit.
ANF-913(P)(A) Volume 1 Revision 1 and Volume 1 Supplements 2, 3 and 4, *COTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses*, Advanced Nuclear Fuels Corporation, August 1990.	3.2.2	Provides a computer program for analyzing BWR system transients
XN-NF-825(P)(A) Supplement 2, *BWR/6 Generic Rod Withdrawal Error Analysis, MCPRp for Plant Operations within the Extended Operating Domain*, Exxon Nuclear Company, October 1986.	3.2.2	Extends previously approved topical report for the CRWE transients for BWR/6 plants operating in the extended operating domain.
ANF-1358(P)(A) Revision 1, *The Loss of Feedwater Heating Transient in Boiling Water Reactors*, Advanced Nuclear Fuels Corporation, September 1992.	3.2.2	Presents a generic methodology for evaluating the loss of feedwater heating event.
EMF-1997(P)(A) Revision 0, *ANFB-10 Critical Power Correlation*, Siemens Power Corporation, July 1998.	3.2.2	Presents an approved critical power correlation for ATRIUM™-10* fuel. The ANFB-10 critical power correlation will be used for the GE11 fuel.

* ATRIUM is a trademark of Framatome ANP.

Report	Applicable LCO	Justification
EMF-1997(P), Supplement 1(P)(A), Revision 0, *ANFB-10 Critical Power Correlation: High Local Peaking Results*, Siemens Power Corporation, July 1998.	3.2.2	Presents experimental results which justify the local peaking limit approved for fuel designs.
EMF-2209(P)(A) Revision 1, *SPCB Critical Power Correlation*, Siemens Power Corporation, July 2000.	3.2.2	Presents an improved critical power correlation for use with the ATRIUM-10 fuel designs.
EMF-2245(P)(A) Revision 0, *Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel*, Siemens Power Corporation, August 2000.	3.2.2	Provides direct and indirect approaches to develop parameters necessary to appropriately model co-resident fuel with an approved critical power correlation.
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, September 1982.	3.2.1	Provides an evaluation model methodology for licensing analyses of postulated LOCAs in jet pump BWRs. The methodology was developed to comply with 10 CFR 50.46 and Appendix K criteria to 10 CFR 50. RELAX and FLEX, which are key computer codes in the methodology, have been subsequently modified.
ANF-91-048(P)(A), *Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model*, Advanced Nuclear Fuels Corporation, January 1993.	3.2.1	Describes updates to the RELAX system blowdown code and FLEX refill codes.
ANF-91-048(P)(A) Supplements 1 and 2, *BWR Jet Pump Model Revision for RELAX*, Siemens Power Corporation, October 1997.	3.2.1	Describes modifications to the jet pump model in the RELAX blowdown code that better predict jet pump performance.
XN-CC-33(A) Revision 1, *HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option Users Manual*, Exxon Nuclear Company, November 1975.	3.2.1	Develops a planar heat transfer model which is used to calculate peak cladding temperatures as part of the evaluation model methodology.
EMF-2292(P)(A) Revision 0, *ATRIUM™-10: Appendix K Spray Heat Transfer Coefficients*, Siemens Power Corporation, September 2000.	3.2.1	Provides measured cladding temperatures from spray heat transfer tests to justify the use of Appendix K coefficients for ATRIUM-10 fuel LOCA analyses.
EMF-CC-074(P)(A) Volume 4 Revision 0, *BWR Stability Analysis - Assessment of STAIF with Input from MICROBURN-B2*, Siemens Power Corporation, August 2000.	3.2.4	Describes methodology for stability analysis with input from the MICROBURN-B2 reactor core simulator.

Report	Applicable LCO	Justification
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.	---	---