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Jerry C. Roberts
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March 22, 2001

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

Attention: Document Control Desk

Subject: Grand Gulf Nuclear Station
Unit 1
Docket No. 50-416
License No. NPF-29
Response to NRC Request for Additional Information Regarding Cycle 12 Reload Proposed Amendment, LDC-2000-076

References:
1. GNRO-2000/00084, dated November 10, 2000, "Grand Gulf Nuclear Station Cycle 12 Reload Proposed Amendment to the Operating License, LDC-2000-076",
2. GNRO-2001-00011, dated February 15, 2001, "Correction to Cycle 12 Reload Proposed Amendment to the Operating License, LDC-2000-076",
3. GNRI-2001/00033, dated March 15, 2001, Request for Additional Information

GNRO-2001/00025

Ladies and Gentlemen:

Please find attached Entergy Operations, Inc. response to the NRC Request for Additional Information (RAI) regarding proposed changes to the Grand Gulf Nuclear Station (GGNS) Technical Specifications (TS) for Cycle 12 operation.

The original amendment request (Reference 1) had been reviewed and accepted by the Plant Safety Review Committee. The conclusions of the Significant Hazards Considerations for this response remain unchanged.

Based on the guidelines in 10CFR50.92, Entergy Operations has concluded that the response to the NRC request for additional information involves no additional significant hazards considerations.

The responses to this RAI introduce 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 operating 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 March 22, 2001.

Yours truly,

JCR/LFD/amt

attachments:
1. Response to Request for Additional Information
2. Approved Topical Reports for COLR References
3. Marked-up Current Technical Specification 5.6.5

cc:

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Levanway	D. E.	(Wise Carter)
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ATTACHMENT 1

TO

GNRO-2001/00025

Response to Request for Additional Information

Question 1

Provide the fuel types and numbers of assemblies used in Grand Gulf Nuclear Station, Unit 1 (GGNS), Cycle 12 operation and identify if they are fresh or irradiated fuel (once or twice burned, etc.) Also, provide the fuel loading pattern for Cycle 12 operation, identify its difference from Cycle 11, and the impact on the SLMCPR calculation.

Response:

The Cycle 12 core is composed of the following fuel types:

Fuel Type	Batch	Total Bundles	Irradiation History
Atrium 10	24, 25	204	Fresh
GE-11	23	228	Once Burned
GE-11	20, 21, 22	268	Twice Burned
GE-11	18, 19	100	Thrice Burned

The full core loading map is provided in Figure 1.1. 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 cycle 12 loading plan is very similar to the cycle 11 core loading. Both cycles use a conventional core loading in which fresh and irradiated fuel bundles are distributed in a checker board configuration throughout the interior region of the core with more depleted fuel loaded on the core periphery. The specific loading pattern is selected to meet fuel design limits while maximizing cycle energy. The potential fuel loading impact on the MCPR safety limit is illustrated by Figure 1.2 which shows the high power portion of the end-of-cycle power histograms. The Cycle 12 histogram is shifted to slightly higher powers due to the improved thermal performance of the Atrium 10 fuel but the shapes are very similar. Since the safety limit is more sensitive to the shape of the histogram, this indicates the core loading changes have little impact on the difference between the Cycle 11 and 12 MCPR safety limits. The difference is attributed primarily to the differences in methodology between the two analyses. With additional reload batches of Atrium 10 fuel, the radial power distribution is projected to flatten so the proposed safety limit includes margin to accommodate the anticipated impact of these changes.

Figure 1.1 - GGNS Cycle 12 Loading by Batch

I / J	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32
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5					18	22	23	23	23	23	25	22	25	22	25	22	22	25	22	25	22	25	23	23	23	23	22	18				
6				19	22	22	23	25	23	25	22	25	23	25	23	25	25	23	25	23	25	22	25	23	25	23	22	22	19			
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8				18	23	25	22	21	23	24	23	24	21	24	23	24	24	23	24	21	24	23	24	23	21	22	25	23	18			
9			18	18	23	23	25	23	24	20	24	20	23	23	20	20	20	20	23	23	20	24	20	24	23	25	23	23	18	18		
10		19	22	22	23	25	22	24	20	23	20	24	23	25	23	25	25	23	25	23	24	20	23	20	24	22	25	23	22	22	19	
11		18	22	23	25	22	23	23	24	20	24	20	24	23	20	21	21	20	23	24	20	24	20	24	23	23	22	25	23	22	18	
12		18	22	23	22	25	23	24	20	24	20	20	23	21	23	24	24	23	21	23	20	20	24	20	24	23	25	22	23	22	18	
13		18	22	23	25	23	21	21	23	23	24	23	25	23	25	21	21	25	23	25	23	24	23	23	21	21	23	25	23	22	18	
14		18	22	23	22	25	21	24	23	25	23	21	23	21	23	25	25	23	21	23	21	23	25	23	24	21	25	22	23	22	18	
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17	18	22	23	25	22	25	21	24	20	25	21	24	21	25	21	22	22	21	25	21	24	21	25	20	24	21	25	22	25	23	22	18
18	18	22	22	23	25	23	25	23	20	23	20	23	25	23	25	21	21	25	23	25	23	20	23	20	23	25	23	25	23	22	22	18
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23		19	22	22	23	25	22	24	20	23	20	24	23	25	23	25	25	23	25	23	24	20	23	20	24	22	25	23	22	22	19	
24			18	18	23	23	25	23	24	20	24	20	23	23	20	20	20	20	23	23	20	24	20	24	23	25	23	23	18	18		
25				18	23	25	22	21	23	24	23	24	21	24	23	24	24	23	24	21	24	23	24	23	21	22	25	23	18			
26				19	23	23	25	22	25	22	23	23	20	21	25	20	20	25	21	20	23	23	22	25	22	25	23	23	19			
27				19	22	22	23	25	23	25	22	25	23	25	23	25	25	23	25	23	25	22	25	23	25	23	22	22	19			
28					18	22	23	23	23	23	25	22	25	22	25	22	22	25	22	25	22	25	23	23	23	23	22	18				
29						18	19	18	18	22	23	23	23	23	23	25	25	23	23	23	23	23	22	18	18	19	18					
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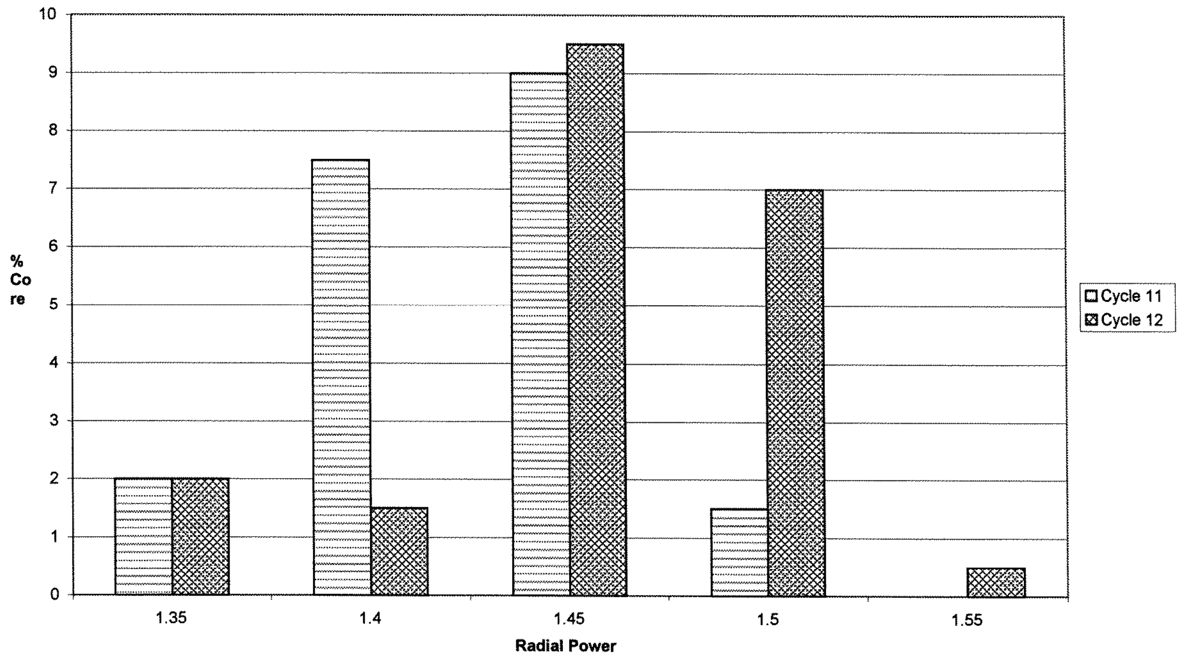
Figure 1.2 : EOC Radial Power Histrogram



Question 2

Provide a description of the test performed for the thermal-hydraulic characteristics of the GE11 fuel design evaluated in Siemens Power Corporation (Siemens) hydraulic test facility, and describe the details of the proper inputs determined for the Cycle 12 SLMCPR calculation based on these test results.

Response:

Thermal-hydraulic testing of a GE11 fuel assembly was performed at FRA-ANP in order to characterize the component pressure drop coefficients of the inlet region (including inlet hardware and lower tie plate), the exit region (including the exit hardware and upper tie plate) and the grid spacers. Data used to determine the hydraulic resistance of the lower tie plate spring seals was also obtained. Differential pressure measurements were taken over a wide range of temperatures, flows and Reynolds Numbers.

The test data reduction process develops the applicable parameters that Framatome uses to model the steady-state and transient thermal-hydraulic behavior of the GE11 fuel design consistent with their thermal hydraulic codes and NRC-approved pressure drop methodology (Reference: XN-NF-79-59(P)(A), *Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies*, Exxon Nuclear Company, November 1983.). Framatome applies the exact same process to develop the thermal hydraulic models for their own fuel designs.

The GE11 and ATRIUM-10 thermal hydraulic models are used in the MCPR safety limit analysis to determine the flow through each of the fuel assemblies.

Question 3

GE 11 fuel is dominant in Grand Gulf Cycle 11 core, in which there are only 36 thrice burned Siemens 9x9-5 fuel assemblies which may not contribute to the difference in calculated SLMCPR value, and Cycle 12 operation is a mixed core of 204 fresh ATRIUM-10 fuel bundles and once and twice burned GE11 fuel bundles. It appears that the two recirculation loop operation for Cycle 12 has less SLMCPR value than that in Cycle 11 operation by 0.01. Please describe the calculation methods in detail, and justify that the decrease of the SLMCPR value for the two recirculation loop operation and no change of SLMCPR for single loop operation still provide enough margin for Cycle 12 operation. Please also provide a description of the methodology which resulted in the data cited in references A.1, A.4, and A.7 of Attachment to GNRO-2000/00084.

Response:

To clarify the question above, Cycle 12 core design contains GE11 once, twice, and thrice burned bundles as described in the answer to Question 1 above.

MCPR safety limit method description:

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 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 Grand Gulf Cycle 12 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 Grand Gulf Nuclear Station Unit 1 Cycle 12 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.

References A.1, A.4, and A.7 of Attachment to GNRO-2000/00084 are summarized as follows:

A.1 Letter, K.V. Walters (SPC) to J.B. Lee (Entergy), "Grand Gulf Unit 1 Cycle 12 Final Core Design Report," GEXI 2000/00076, August 29, 2000.

A.4 Letter, J.B. Lee (Entergy) to K.V. Walters (SPC), "Grand Gulf Nuclear Station Unit 1 and River Bend Station Unit 1, Reload Transition Data-GE11 Additive Constants," CEXO-2000-00293, July 25, 2000.

A.7 Letter, J. Lee (Entergy) to K.V. Walters (SPC), "Plant Parameters to Support MCPR Safety Limit Analysis-Grand Gulf," August 28, 2000.

Reference A.1 provided a description of the Cycle 12 core design of which portions were used for the Safety Limit MCPR calculations. Specifically, this reference provided a neutronic description of the Cycle 12 fresh fuel including batch size and split, axial enrichments, gad loading, etc. This reference also provided the reference loading pattern, which describes the location of each fuel bundle in the core. The response to question 1 above provides the Cycle 12 core composition by fuel type and batch number.

Reference A.4 provided to Siemens Power Corporation information to be used for the modeling of the co-resident GE11 fuel. Of particular interest are the GE11 additive constants to be used with SPC's ANFB10 critical power correlation and the GE11 critical power correlation additive constant uncertainty which is used in the statistical convolution process of the Safety Limit MCPR evaluation. This information was developed by EOI in accordance with the NRC approved methodology given in EMF-2245, "Application of Siemens Power Corporation's Critical Power Correlations to Co-Resident Fuel." The GE11 critical power correlation additive constant uncertainty is given in Table 1 of Attachment 4 to GNRO 2000-00084.

Reference A.7 provided the GGNS plant specific inputs to the Safety Limit calculation. Of particular interest are the heat balance parameters and the associated plant measurement uncertainties. These reactor system uncertainties, which are given in Table 1 of Attachment 4 to GNRO 2000-00084, are identical to those applied in the Cycle 11 Safety Limit MCPR calculations and have been confirmed to be applicable to GGNS. These generic BWR parameters are identical to those previously approved for General Electric's MCPR safety limit calculations in NEDC-32601. These uncertainties represent a portion of the uncertainties that are statistically convolved in accordance with SPC's NRC approved methodology in the development of the Safety Limit MCPR.

Question 4

Technical Specifications (TS) for the Core Operating Limits Report (COLR) include the TSs to be removed to the COLR and the list of approved methodologies to support those TSs to be removed to the COLR. Those methods should provide the calculation of the cycle-specific core operating limits specified in the COLR TS and in Generic Letter 88-16. Provide the justification that those proposed 26 approved topical reports satisfy the COLR TS criteria. Also, for TS 5.6.5 COLR reference a.5, provide a more definitive reference location for the actual cycle-specific core operating limit parameters which are contained within the TS specified.

Response:

See Attachment 2 of this response to the Request for Additional Information for a brief summary of the justification of the proposed references. It has been determined that proposed reference number 5 (i.e. the fuel channel reference [(EMF-93-177(P)(A)]) and proposed reference number 26, (i.e. attachment 4 to GNRO-2000-00084) do not need to be referenced in the Technical Specifications. Attachment 2 to this letter includes the applicable LCO for each of the proposed Framatome-ANP references. Attachment 3 to this letter contains the revised TS section 5.6.5 with the two references above deleted. 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.

TS 5.6.5 COLR references a.4, a.5, and a.6 were added as part of the solution to the neutronic/thermal-hydraulic instability issues as identified in NRC IE Bulletin 88-7, Supplement 1 and GL 94-02. Grand Gulf Nuclear Station (GGNS) has chosen to implement Enhanced Option 1-A (E1A) Core Stability to address these issues.

As part of the GGNS E1A solution approved by the Staff in Amendment Number 141 (TAC No. MA3406), Technical Specification (TS) section 5.6.5 was modified to indicate that the Core Operating Limits Report (COLR) contained limits associated with LCO 3.2.4, "Fraction of Core Boiling Boundary (FCBB)", LCO 3.3.1.1, "RPS Instrumentation", and LCO 3.3.1.3, "Period Based Detection System (PBDS)". Reference a.5 (LCO 3.3.1.1) includes the Allowable Values of the APRM flow biased scram (see Table 3.3.1.1-1 Function 2.d).

The E1A APRM flow biased scram function provides a preemptive automatic reactor scram upon entry into the Exclusion Region of the power operating curve. Since the APRM flow biased scram function is a feature of the E1A stability solution necessary to ensure compliance with 10 CFR 50 Appendix A, General Design Criterion 12, it was added to TS Table 3.3.1.1-1 as RPS Function 2.d. The cycle specific core operating limits parameters are the high flow-biased scram allowable values based on aligned drive flow and simulated thermal power. NEDO-32339-A, Supplement 4, Revision 1 places the Allowable Value of the APRM flow biased scram function in the COLR.

Question 5

A statement made in CRNO-2000-0024, in the section titled Application Procedures, reads "...As this is expected to be a one-time analysis with no ongoing applications of the methodology, a separate calculation procedure was not developed..." Please clarify its real intention for this cycle specific application.

Response:

CNRO-2000-00024 discusses the technology transfer concerning the development of the additive constants for the GE11 fuel at the EOI BWR plants (Grand Gulf Nuclear Station (GGNS) and River Bend Station (RBS)). The additive constants and their uncertainties were developed for the use of the ANFB10-Edge CPR correlation with GE11 fuel and do not change on a plant- or cycle-specific basis. Since both GGNS and RBS will be transitioning to Framatome fuel, EOI expects that no further analysis concerning the development of additive constants for the GE11 fuel in either EOI BWR plant is needed for subsequent cycles.

ATTACHMENT 2

TO

GNRO-2001/00025

BWR Approved Topical Reports For
GGNS and RBS Tech Spec and COLR References

**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.

Report	Applicable LCO	Justification
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[1] fuel. The ANFB-10 critical power correlation will be used for the GE11 fuel.
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.

[1] ATRIUM is a trademark of Framatome ANP.

Report	Applicable LCO	Justification
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.
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.	---	---

ATTACHMENT 3

TO

GNRO-2001/00025

Marked-up Current Technical Specification 5.6.5

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.

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