Entergy

Entergy
P.O. Box 756
Port Gibson, MS 39150
Tel 601 437 6409

George A. Williams
Vice President-Operations
Grand Gulf Nuclear Station

GNRO-2006/00015

May 8, 2006

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

SUBJECT: License Amendment Request
Changes to the Analytical Methods Referenced in Technical Specification 5.6.5, "Core Operating Limits Report (COLR)"
Grand Gulf Nuclear Station, Unit 1
Docket No. 50-416
License No. NPF-29

Dear Sir or Madam:

Pursuant to 10 CFR 50.90, Entergy Operations, Inc. (Entergy) hereby requests the following amendment for Grand Gulf Nuclear Station, Unit 1 (GGNS). The proposed change will add a NRC previously approved topical report to the analytical methods referenced in Technical Specification (TS) Section 5.6.5, "Core Operating Limits Report (COLR)." TS Section 5.6.5 requires core operating limits to be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, using the analytical methods previously approved by the NRC and referenced in Section 5.6.5.b. Certain core operating limits are established based upon design basis Loss-of-Coolant Accident (LOCA) analyses. The current method of performing the LOCA analyses will be replaced by an updated method described in Framatome Advanced Nuclear Power (FRA-ANP) topical report, "EXEM BWR-2000 ECCS Evaluation Model".

GGNS plans to use the updated methodology beginning with operating Cycle 16, currently scheduled to begin spring 2007. EXEM BWR-2000 has been approved by the NRC and is applicable to the GGNS plant design and the fuel being used at GGNS. GGNS currently uses FRA-ANP ATRIUM-10 fuel and will continue to do so during operating Cycle 16.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that this change involves no significant hazards consideration. The bases for these determinations are included in the attached submittal.

The proposed change does not include any new commitments.

The NRC has approved similar TS changes to allow the use of EXEM BWR-2000 for LOCA analysis at Susquehanna Units 1 and 2, Browns Ferry Units 2 and 3, LaSalle County Station Units 1 and 2, and Columbia Generating Station.

Entergy requests approval of the proposed amendment by February 1, 2007 to support Refueling Outage 15. Once approved, the amendment shall be implemented prior to Cycle 16 operation. Although this request is neither exigent nor emergency, your prompt review is requested.

If you have any questions or require additional information, please contact Ron Byrd at 601-368-5792.

I declare under penalty of perjury that the foregoing is true and correct. Executed on May 8, 2006.

Sincerely,

George A. Williams

GAW/RWB/amt
Attachments:
1. Analysis of Proposed Technical Specification Change
2. Proposed Technical Specification Changes (mark-up)

cc: (See Next Page)

cc: NRC Senior Resident Inspector
Grand Gulf Nuclear Station
Port Gibson, MS 39150

Dr. Bruce S. Mallett
Regional Administrator, Region IV
U. S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-4005

U. S. Nuclear Regulatory Commission
ATTN: Mr. Bhalchandra Vaidya, NRR/DORL (w/2)
ATTN: ADDRESSEE ONLY
Mail Stop OWFN/O-7D1A
Washington, DC 20555-0001

Mr. Brian W. Amy, MD, MHA, MPH
Mississippi Department of Health
P. O. Box 1700
Jackson, MS 39215-1700

Mr. D. E. Levanway (Wise Carter)
Mr. L. J. Smith (Wise Carter)
Mr. N. S. Reynolds
Mr. J. N. Compton

Attachment 1

GNR0-2006/00015

Analysis of Proposed Technical Specification Change

1.0 DESCRIPTION

This letter is a request to amend Operating License NPF-29 for Grand Gulf Nuclear Station, Unit 1 (GGNS).

The proposed change will add a NRC previously approved topical report to the analytical methods referenced in Technical Specification (TS) Section 5.6.5, "Core Operating Limits Report (COLR)." TS Section 5.6.5 requires core operating limits to be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, using the analytical methods previously approved by the NRC and referenced in Section 5.6.5.b. Certain core operating limits are established based upon design basis Loss-of-Coolant Accident (LOCA) analyses. The current method of performing the LOCA analyses will be replaced by an updated method described in Framatome Advanced Nuclear Power (FRA-ANP) topical report, "EXEM BWR-2000 ECCS Evaluation Model".

GGNS plans to use the updated LOCA analytical method to determine specific core operating limits for operating Cycle 16 currently scheduled to begin spring 2007.

2.0 PROPOSED CHANGE

TS Section 5.6.5.b lists the analytical methods previously reviewed and approved by the NRC that are used to determine the core operating limits. The list includes the following topical reports:

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

These references are proposed to be revised as follows:

18. EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP Richland, Inc.

19. Deleted

20. Deleted

21. Deleted

The TS change is intended to allow GGNS to use the EXEM BWR-2000 methodology for performing LOCA analyses beginning with operating Cycle 16. The LOCA analyses will be used to establish the Average Planar Linear Heat Generation Rate (APLHGR) operating limits, imposed by TS 3.2.1. The NRC staff approved the EXEM BWR-2000 topical report by letter dated May 29, 2001 and found it to be acceptable for referencing in license applications. GGNS currently performs LOCA analyses using the EXEM BWR Evaluation Model in conjunction with the RELAX and HUXY codes (References 19, 20, and 21). The EXEM BWR ECCS Evaluation Model (current Reference 18) is no longer used and is being deleted.

References to the topical reports for the RELAX and HUXY codes (References 20 and 21) no longer need to be listed because they are incorporated by reference into the EXEM BWR-2000 ECCS Evaluation Model topical report (new Reference 18). This is consistent with other licensee amendments that approved the use of EXEM BWR-2000.

3.0 BACKGROUND

Core operating limits are established each operating cycle in accordance with TS 3.2, "Power Distribution" and TS 5.6.5, "Core Operating Limits Report (COLR)". These operating limits ensure that the fuel design limits are not exceeded during any conditions of normal operation or in the event of any Anticipated Operational Occurrence (AOO). In addition, the APLHGR operating limits imposed by TS 3.2.1 also ensure that the Peak Cladding Temperature (PCT) during the postulated design basis LOCA does not exceed the 2200°F limit specified in 10 CFR 50.46.

The methods used to determine the operating limits are those previously found acceptable by the NRC and listed in TS Section 5.6.5.b. The analytical methods currently listed support the determination of core operating limits by using those methods applicable to fuel supplied by General Electric (GE, currently known as Global Nuclear Fuels) or Framatome Advanced Nuclear Power (FRA-ANP, formerly known as Siemens). GGNS has employed fuel supplied by GE or FRA-ANP since it began commercial operation but is only using FRA-ANP ATRIUM-10 fuel in the current operating cycle. GGNS also plans to continue using ATRIUM-10 fuel in the next operating cycle, Cycle 16.

The requested TS change will add a previously approved FRA-ANP topical report to the references listed in TS Section 5.6.5.b. The Topical Report, EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model," describes an updated method of performing ECCS evaluations under design basis LOCA conditions. GGNS currently performs the LOCA analysis using the EXEM BWR methodology described in Topical Report ANF-91-048(P)(A), "Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model." The LOCA analysis is used to establish the APLHGR limits required by Technical Specification 3.2.1.

One of the principal reasons for the updated methodology was to address some issues raised during a 1997 NRC Core Performance Inspection of FRA-ANP. The inspection included an assessment of the EXEM BWR evaluation model. The assessment determined that the benchmarking and validation of certain computer codes used to support the evaluation model needed to be improved. These concerns were addressed by the development and approval of the EXEM BWR-2000 ECCS evaluation model. As part of the resolution of the NRC assessment issues, the older EXEM BWR evaluation model will no longer be used. This necessitates the need for the GGNS LOCA analysis to be reanalyzed with the EXEM BWR-2000 model. The proposed TS change, if approved, will be implemented beginning with GGNS operating Cycle 16.

4.0 TECHNICAL ANALYSIS

The APLHGR limits required by Technical Specification 3.2.1 are specified in the COLR and are the result of fuel design, Design Basis Accident (DBA), and transient analyses. The APLHGR is a measure of the average linear heat generation rate of all the fuel rods in a fuel assembly at any axial location. Limits on the APLHGR are specified to ensure that the fuel design limits are not exceeded during anticipated operational occurrences (AOOs) and that the peak cladding temperature (PCT) during the postulated design basis LOCA does not exceed the 2200°F limit specified in 10 CFR 50.46.

GGNS currently uses the NRC approved EXEM BWR evaluation model for the LOCA analysis. GGNS proposes to use an updated FRA-ANP LOCA analytical method beginning in operating Cycle 16. The updated method is described in Topical Report EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model." The NRC found the topical report to be acceptable for referencing in license applications and issued its safety evaluation providing the basis for acceptance by letter dated May 29, 2001.

The NRC review of the EXEM BWR-2000 model concluded the following:

> The staff notes that from validation against test data, the large-break Design Basis Accident (DBA) Peak Clad Temperatures (PCTs) were conservatively calculated. The test results for small breaks show low temperatures, and the EXEM BWR-2000 model using evaluation model options bounds the temperature data. Furthermore, the EXEM BWR-2000 model adequately predicts the important LOCA phenomena.
>
> The staff therefore concludes that the proposed EXEM BWR-2000 ECCS EM, as documented in References 1, 2, 4 and 5, is acceptable for referencing in BWR LOCA analyses, with the limitation that application of the revised evaluation model will be limited to jet pump applications.

Entergy has determined that the FRA-ANP EXEM BWR-2000 evaluation model is acceptable for GGNS LOCA analyses. The GGNS plant design meets the limitation stipulated in the NRC safety evaluation since it is a BWR-6 plant which incorporates jet pumps in its design. GGNS uses FRA-ANP ATRIUM-10 fuel in its current operating cycle (Cycle 15) and plans to continue using ATRIUM-10 fuel in Cycle 16.

A GGNS plant specific LOCA analysis was performed by FRA-ANP using the EXEM BWR-2000 evaluation model. The analysis assumed a full core of ATRIUM-10 fuel and used a generic ATRIUM-10 neutronic design that is expected to be conservative relative to actual cycle-specific designs. A cycle specific evaluation is performed each cycle to confirm that the generic fuel design remains bounding.

The results of the new GGNS LOCA analysis were compared with the current licensing basis analysis, which uses the older EXEM BWR evaluation model. Tables 1 through 5 included in this attachment provide the key input parameters used in both the current licensing basis analysis and the new analysis. The current calculated LOCA PCT for ATRIUM-10 fuel, using the older EXEM BWR model, is 1797°F. The LOCA PCT for ATRIUM-10 fuel using the new EXEM BWR-2000 model is 1895°F. As such, there is an increase in PCT of 98°F associated with the change in the LOCA analysis. However, it should be noted that the new analysis assumes that the plant is operating at a lower Minimum Critical Power Ratio (MCPR) at the start of the event. This lower MCPR input value results in an increase in PCT but does not impact the current MCPR Safety Limits provided in Technical Specification 2.1.1.2. The new analysis also conservatively assumed that only 6 of 8 Automatic Depressurization System (ADS) valves were available; whereas, the current analysis assumed 7 were available. The assumption is conservative since the TS do not allow plant operation with less than 7 operable ADS valves. The resulting PCT of 1895°F, using the EXEM BWR-2000 model for the LOCA analyses, still affords adequate margin to the 2200°F limit of 10 CFR 50.46.

In summary, the EXEM BWR-2000 evaluation model is an improved method of evaluating ECCS performance with LOCA analyses. The model has been reviewed and approved by the NRC and is applicable to the GGNS plant design and the fuel being used at GGNS. The application of the LOCA analysis model will continue to ensure that the APLHGR operating limits are established to protect the fuel cladding integrity during normal operation, AOOs, and the design basis LOCA.

5.0 REGULATORY ANALYSIS

5.1 Applicable Regulatory Requirements/Criteria

The proposed changes have been evaluated to determine whether applicable regulations and requirements continue to be met.

Entergy has determined that the proposed changes do not require any exemptions or relief from regulatory requirements, other than the TS, and do not affect conformance with any General Design Criterion (GDC) differently than described in the Updated Final Safety Analysis Report (UFSAR).

10 CFR 50.36, Paragraph (c)(5), states that the TS will include administrative controls that address the provisions relating to organization and management, procedures, record keeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. The COLR is required as a part of the reporting requirements specified in the GGNS TS Administrative Controls section. The TS requires the core operating limits to be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and to be documented in the COLR. In addition, it requires the analytical methods used to determine the core operating limits to be approved by the NRC and described in the Administrative Controls section of the TS. The proposed TS changes ensure that these requirements are met.

Generic Letter (GL) 88-16, "Removal of Cycle-Specific parameters from Technical Specifications" and Technical Specification Task Force (TSTF) traveler TSTF-363, "Revise Topical Report References in ITS 5.6.5, COLR" provide guidance on the method of referencing topical reports in the TS and the COLR. TSTF-363, which was approved by the NRC on July 6, 2000, requires the titles of the analytical methods to be included in TS 5.6.5.b and the complete identification (report number, title, revision, date, and any supplements) to be included in the COLR. The proposed TS changes are consistent with this guidance.

10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors" requires each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical zircaloy or ZIRLO cladding to be provided with an ECCS designed so that its calculated cooling performance following postulated LOCAs conforms to explicit criteria. Specifically related to this proposed TS change is a criterion that the calculated maximum fuel element cladding temperature shall not exceed 2200 F. In addition, the ECCS cooling performance must be calculated in accordance with an acceptable evaluation model. The proposed TS change allows use of an acceptable evaluation model that ensures conformance with these requirements.

5.2 No Significant Hazards Consideration

Entergy proposes to use an updated Loss-of-Coolant Accident (LOCA) analysis method for Grand Gulf Nuclear Station (GGNS) beginning with operating Cycle 16. The Framatome Advanced Nuclear Power (FRA-ANP) method of analysis, EXEM BWR-2000, will be used to determine the Average Planar Linear Heat Generation Rate (APLHGR) core operating limits imposed by Technical Specification (TS) 3.2.1. TS Section 5.6.5.b must be revised to include a reference to the topical report that describes the updated method prior to the method being used to establish the APLHGR limits. Therefore Entergy proposes to revise TS Section 5.6.5.b to replace references to the current method, EXEM BWR, with a reference to the updated method, EXEM BWR-2000.

Entergy Operations, Inc. has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

   Core operating limits are established each operating cycle in accordance with TS 3.2, "Power Distribution" and TS 5.6.5, "Core Operating Limits Report (COLR)". These core operating limits ensure that the fuel design limits are not exceeded during any conditions of normal operation or in the event of any Anticipated Operational Occurrence (AOO). In addition, the Average Planar Linear Heat Generation Rate (APLHGR) operating limits imposed by Technical Specification 3.2.1 also ensure that the Peak Cladding Temperature (PCT) during the postulated design basis LOCA does not exceed the 2200°F limit specified in 10 CFR 50.46. The APLHGR is a measure of the average linear heat generation rate of all the fuel rods in a fuel assembly at any axial location.

   The methods used to determine the operating limits are those previously found acceptable by the NRC and listed in TS Section 5.6.5.b. A change to TS Section 5.6.5.b is requested to include an updated LOCA analysis method, EXEM BWR-2000. The updated method will be used to determine the APLHGR operating limits imposed by Technical Specification 3.2.1. EXEM BWR-2000 has been reviewed and approved by the NRC and is applicable to the GGNS plant design and the FRA-ANP fuel being used at GGNS. The application of the LOCA analytical model will continue to ensure that the APLHGR operating limits are established to protect the fuel cladding integrity during normal operation, AOOs, and the design basis LOCA.

The requested TS changes concern the use of analytical methods and do not involve any plant modifications or operational changes that could affect any postulated accident precursors or accident mitigation systems and do not introduce any new accident initiation mechanisms.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   Response: No.

   The proposed TS amendment will not change the design function, reliability, performance, or operation of any plant systems, components, or structures. It does not create the possibility of a new failure mechanism, malfunction, or accident initiators not considered in the design and licensing bases. Plant operation will continue to be within the core operating limits that are established using NRC approved methods that are applicable to the GGNS design and the GGNS fuel.

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

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The ECCS performance analysis methods are used to establish the APLHGR limits required by Technical Specification 3.2.1. The APLHGR limits are specified in the COLR and are the result of fuel design, design basis accident (DBA), and transient analyses. Limits on the APLHGR are specified to ensure that the fuel design limits are not exceeded during anticipated operational occurrences (AOOs) and that the peak cladding temperature (PCT) during the postulated design basis LOCA does not exceed the 2200°F limit specified in 10 CFR 50.46.

   The EXEM BWR-2000 evaluation model is an updated LOCA analytical method that has been approved by the NRC and is applicable to the GGNS plant design and the fuel being used at GGNS. A GGNS plant specific ECCS performance analysis has been performed with the EXEM BWR-2000 evaluation model. This evaluation concluded that the resulting PCT still afforded adequate margin to the 2200°F limit of 10 CFR 50.46

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

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

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 PRECEDENCE

The NRC has approved the use of the EXEM BWR-2000 ECCS Evaluation Model, Topical Report EMF-2361(P)(A), for use at the following plants.

PLANT NAME	AMENDMENT No.
1. Columbia Generating Station	Amendment No. 185 (ADAMS Accession No.: ML031340744)
2. Susquehanna Steam Electric Station Unit 1	Amendment No. 231 (ADAMS Accession No.: ML060730355)
3. Susquehanna Steam Electric Station Unit 2	Amendment No. 194 (ADAMS Accession No.: ML050590044)
4. Browns Ferry Units 2 and 3	Amendment Nos. 287 and 245 (ADAMS Accession No.: ML033650142)
5. LaSalle County Station Units 1 and 2	Amendment Nos. 174 and 160 (ADAMS Accession No.: ML060120391)

Table 1
Initial Conditions

Parameter	EXEM BWR Value	EXEM BWR-2000 Value
Reactor Power (MWt)	4105.5	4105.5
Total Core Flow (Mlb/hr)	118.1	118.1
Hot Assembly MAPLHGR (kW/ft)	12.5	12.5
Hot Assembly MCPR	1.19	1.16

Table 2
High Pressure Core Spray (HPCS) Parameters

Parameter	EXEM BWR Value	EXEM BWR-2000 Value
Maximum Coolant Temperature (°F)	120	120
Initiating Signals and Setpoints		
Water Level (in)[1]	475	475
High Drywell Pressure (psig)	Not Used	Not Used
Time Delays		
Time for HPCS pump to reach rated speed and injection valve wide open (sec)	35	35
Coolant Flow Rate Versus Pressure		
Vessel to Drywell ΔP (psid)	Flow Rate (gpm)	
0	7000	7000
200	6300	6300
1147	1485	1485
1177	495	495
>1177	0	0

[1] Relative to vessel zero

Table 3
Low Pressure Coolant Injection (LPCI) Parameters

Parameter	EXEM BWR Value	EXEM BWR-2000 Value
Reactor pressure permissive for opening valves – analytical (psia)	450	450
Maximum Coolant Temperature (°F)	120	120
Initiating Signals and Setpoints		
Water Level (in)[2]	366.3	366.3
High Drywell Pressure (psig)	Not Used	Not Used
Time Delays		
Maximum time for LPCI pumps to reach rated speed (sec)	47	47
LPCI injection valve stroke time (sec)	35	35
Coolant Flow Rate Versus Pressure		
Vessel to Drywell ΔP (psid)	Flow Rate (gpm)	
0	7333.33	7333.33
20	7333.33	7333.33
225	0	0

[2] Relative to vessel zero

Table 4
Low Pressure Core Spray (LPCS) Parameters

Parameter	EXEM BWR Value	EXEM BWR-2000 Value
Reactor pressure permissive for opening valves – analytical (psia)	450	450
Maximum Coolant Temperature (°F)	120	120
Initiating Signals and Setpoints		
Water Level (in)[3]	366.3	366.3
High Drywell Pressure (psig)	Not Used	Not Used
Time Delays		
Maximum time for LPCS pump to reach rated speed (sec)	47	47
LPCS injection valve stroke time (sec)	35	35
Coolant Flow Rate Versus Pressure		
Vessel to Drywell ΔP (psid)	Flow Rate (gpm)	
0	7000	7000
122	6300	6300
289	0	0

[3] Relative to vessel zero

Table 5
Automatic Depressurization System (ADS) Parameters

Parameter	EXEM BWR Value [1]	EXEM BWR-2000 Value [2]
Number of Valves Installed	8	8
Number of Valves Available	7	6[4]
Minimum flow capacity of available valves (Mlbm/hr at psig)	6.475 at 1241	5.550 at 1241
Initiating Signals and Setpoints		
Water Level (in)[5]	366.3	366.3
High Drywell Pressure (psig)[6]	2	2
Time Delays		
Delay Time (from initiating signal to time valves are open) (sec)[6]	120	120

[4] Only 6 valves are assumed operable in the analyses to support one ADSVOOS operation and to eliminate the need to evaluate the single failure of an ADS valve as a separate single failure (total of 2 inoperable ADS valves)
[5] Relative to vessel zero
[6] The drywell high-pressure setpoint is exceeded before the water level setpoint is reached. Therefore, the ADS timer is assumed to start when the water level setpoint is reached.

Attachment 2

GNRO-2006/00015

Proposed Technical Specification Changes (mark-up includes affected Operating License page)

5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR) (continued)

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), "BWR/6 Generic Rod Withdrawal Error Analysis, $MCPR_p$ for Plant Operations within the Extended Operating Domain", Exxon Nuclear Company, Inc., Richland, WA.

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

EMF-2361(P)(A), "EXEM BWR-2000 ECCS Evaluation Model," Framatome ANP Richland, Inc.

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

Deleted

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.~~

(continued)

5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR) (continued)

Deleted

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) 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.

(continued)

INSERT "A"
move to page 5.0-21

Delete
page

5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR) (continued)

INSERT "A" from page 5.0-20a

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.

(b) SERI is required to notify the NRC in writing prior to any change in (i) the terms or conditions of any new or existing sale or lease agreements executed as part of the above authorized financial transactions, (ii) the GGNS Unit 1 operating agreement, (iii) the existing property insurance coverage for GGNS Unit 1 that would materially alter the representations and conditions set forth in the Staff's Safety Evaluation Report dated December 19, 1988 attached to Amendment No. 54. In addition, SERI is required to notify the NRC of any action by a lessor or other successor in interest to SERI that may have an effect on the operation of the facility.

C. The license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Entergy Operations, Inc. is authorized to operate the facility at reactor core power levels not in excess of 3898 megawatts thermal (100 percent power) in accordance with the conditions specified herein.

(2) *Technical Specifications*

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. ~~169~~ are hereby incorporated into this license. Entergy Operations, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

Insert new Amendment No.

The Surveillance Requirements (SRs) for Diesel Generator 12 contained in the Technical Specifications and listed below, are not required to be performed immediately upon implementation of Amendment No. 169. The SRs listed below shall be successfully demonstrated at the next regularly scheduled performance.

SR 3.8.1.9,
SR 3.8.1.10, and
SR 3.8.1.14