UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY OPERATIONS, INC.

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

ENTERGY MISSISSIPPI, INC.

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 141
License No. NPF-29

1. The Nuclear Regulatory Commission (the Commission) has found that:

A. The application for amendment by Entergy Operations, Inc. (the licensee) dated July 20, 1998, as supplemented by letter dated June 29, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications, as indicated in the attachment to this license amendment; and paragraph 2.C.(2) of Facility Operating License No. NPF-29 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 141, are hereby incorporated into this license. Entergy Operations, Inc. shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 120 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A Gramm

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: January 19, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 141

FACILITY OPERATING LICENSE NO. NPF-29

DOCKET NO. 50-416

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
1.0-3	1.0-3
---	1.0-3a
---	3.2-4
---	3.2-5
---	3.2-6
3.3-4	3.3-4
---	3.3-4a
3.3-5	3.3-5
---	3.3-5a
3.3-6	3.3-6
---	3.3-13a
---	3.3-13b
3.4-1	3.4-1
3.4-2	3.4-2
3.4-3	3.4-3
3.4-4	3.4-4
3.4-5	---
5.0-18	5.0-18
5.0-20	5.0-20

1.1 Definitions

DOSE EQUIVALENT I-131 (continued)	be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites."
EMERGENCY CORE COOLING SYSTEM (ECCS) RESPONSE TIME	The ECCS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ECCS initiation setpoint at the channel sensor until the ECCS equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) SYSTEM RESPONSE TIME	The EOC-RPT SYSTEM RESPONSE TIME shall be that time interval from initial movement of the associated turbine stop valve or the turbine control valve to complete suppression of the electric arc between the fully open contacts of the recirculation pump circuit breaker. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured, except for the breaker arc suppression time, which is not measured but is validated to conform to the manufacturer's design value.
FRACTION OF CORE BOILING BOUNDARY (FCBB)	The FCBB shall be the ratio of the power generated in the lower 4 feet of the active reactor core to the power required to produce bulk saturated boiling of the coolant entering the fuel channels. The core boiling boundary is the axial elevation of core average bulk saturation above the bottom of the active reactor core.
ISOLATION SYSTEM RESPONSE TIME	The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. The response time

(continued)

1.1 Definitions

ISOLATION SYSTEM RESPONSE TIME	may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
L_a	The maximum allowable primary containment leakage rate, L_a, shall be 0.437% of primary containment air weight per day at the calculated peak containment pressure (P_a).

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Fraction of Core Boiling Boundary (FCBB)

LCO 3.2.4 The FCBB shall be ≤ 1.0.

APPLICABILITY: THERMAL POWER and core flow in the Restricted Region as Specified in the COLR,
MODE 1 when RPS Function 2.d, APRM Flow Biased Simulated Thermal Power - High, Allowable Value is "Setup" as specified in the COLR.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. FCBB not within limit for reasons other than an unexpected loss of feedwater heating or unexpected reduction in core flow.	A.1 Restore FCBB to within limit.	2 hours
B. Required Action and associated Completion Time of Condition A not met. OR	B.1 Initiate action to exit the Restricted Region. AND B.2 Initiate action to return APRM Flow Biased Simulated Thermal Power - High Allowable Value to "non-Setup" value.	Immediately Immediately following exit of Restricted Region

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued) --------NOTE--------- Required Action B.1 and Required Action B.2 shall be completed if this Condition is entered due to an unexpected loss of feedwater heating or unexpected reduction in core flow. --------------------- FCBB not within limit due to an unexpected loss of feedwater heating or unexpected reduction in core flow.		

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	------------------NOTE------------------ Not required to be performed until 15 minutes after entry into the Restricted Region if entry was the result of an unexpected transient. -- Verify FCBB ≤ 1.0.	24 hours AND Once within 15 minutes following unexpected transient

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.6	------------------NOTE------------------ Only required to be met during entry into MODE 2 from MODE 1. --------------------------------------- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.7	Calibrate the local power range monitors.	1000 MWD/T average core exposure
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	Calibrate the trip units.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.10	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 3. For Function 2.d, APRM recirculation flow transmitters are excluded. 4. For Function 2.d, the digital components of the flow control trip reference cards are excluded. -- Perform CHANNEL CALIBRATION.	184 days

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.11	Perform CHANNEL FUNCTIONAL TEST.	18 months
SR 3.3.1.1.12	-----NOTES----- 1. Neutron detectors are excluded. 2. For IRMs, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ---------- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.1.13	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months
SR 3.3.1.1.14	Verify Turbine Stop Valve Closure, Trip Oil Pressure — Low and Turbine Control Valve Fast Closure Trip Oil Pressure — Low Functions are not bypassed when THERMAL POWER is $\geq$ 40% RTP.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.15	-----------------NOTES----------------- 1. Neutron detectors are excluded. 2. For Functions 3, 4, and 5 in Table 3.3.1.1-1, the channel sensors may be excluded. 3. For Function 6, "n" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. --- Verify the RPS RESPONSE TIME is within limits.	18 months on a STAGGERED TEST BASIS
SR 3.3.1.1.16	Verify the simulated thermal power time constant.	18 months
SR 3.3.1.1.17	Perform APRM recirculation flow transmitter calibration.	18 months
SR 3.3.1.1.18	Adjust the flow control trip reference card to conform to reactor flow.	Once within 7 days after reaching equilibrium conditions following refueling outage

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux – High	2	3	H	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.12 SR 3.3.1.1.13	≤ 122/125 divisions of full scale
	5(a)	3	I	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.12 SR 3.3.1.1.13	≤ 122/125 divisions of full scale
b. Inop	2	3	H	SR 3.3.1.1.3 SR 3.3.1.1.13	NA
	5(a)	3	I	SR 3.3.1.1.4 SR 3.3.1.1.13	NA
2. Average Power Range Monitors					
a. Neutron Flux – High, Setdown	2	3	H	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 20% RTP
b. Fixed Neutron Flux – High	1	3	G	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 120% RTP
c. Inop	1,2	3	H	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.13	NA
d. Flow Biased Simulated Thermal Power - High	1	3	G	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.17 SR 3.3.1.1.18	(b)

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.
(b) Allowable Values specified in the COLR. Allowable Value modification required by the COLR due to reductions in feedwater temperature may be delayed for up to 12 hours.

3.3 INSTRUMENTATION

3.3.1.3 Period Based Detection System (PBDS)

LCO 3.3.1.3 One channel of PBDS instrumentation shall be OPERABLE.

AND

Each OPERABLE channel of PBDS instrumentation shall not indicate Hi-Hi DR alarm.

APPLICABILITY: THERMAL POWER and core flow in the Restricted Region specified in the COLR,
THERMAL POWER and core flow in the Monitored Region specified in the COLR.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Any OPERABLE PBDS channel indicating Hi-Hi DR Alarm	A.1	Place the reactor mode switch in the shutdown position.	Immediately
B. Required PBDS channel inoperable while in the Restricted Region.	B.1	---------NOTE-------- Only applicable if RPS Function 2.d, APRM Flow Biased Simulated Thermal Power - High, Allowable Value is "Setup". -------------------- Initiate action to exit the Restricted Region.	 Immediately
	OR		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.2	Place the reactor mode switch in the shutdown position.	Immediately
C. Required PBDS channel inoperable while in the Monitored Region.	C.1	Initiate action to exit the Monitored Region.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.1.3.1	Verify each OPERABLE channel of PBDS instrumentation not in Hi-Hi DR Alarm.	12 hours
SR 3.3.1.3.2	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.3.3	Perform CHANNEL FUNCTIONAL TEST.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation.

OR

One recirculation loop shall be in operation with the required limits modified for single loop operation as specified in the COLR.

-------------------------NOTE-------------------------
Required limit modifications for single recirculation loop operation may be delayed for up to 12 hours after transition from two recirculation loop operation to single recirculation loop operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Recirculation loop jet pump flow mismatch not within limits.	A.1 Shutdown one recirculation loop.	2 hours

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ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. No recirculation loops in operation.	B.1	Be in MODE 3.	12 hours
C. Required limit modifications not performed.	C.1	Declare associated limit(s) not met.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	------------------NOTE------------------ Not required to be performed until 24 hours after both recirculation loops are in operation. -- Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation is: a. ≤ 10% of rated core flow when operating at < 70% of rated core flow; and b. ≤ 5% of rated core flow when operating at ≥ 70% of rated core flow.	24 hours

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),

3) LCO 3.2.3, Linear Heat Generation Rate (LHGR),

4) LCO 3.2.4, Fraction of Core Boiling Boundary (FCBB).

5) LCO 3.3.1.1, RPS Instrumentation, and

6) LCO 3.3.1.3, Period Based Detection System (PBDS).

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5.6 Reporting Requirements

5.6.5 Core Operating Limits Report (COLR) (continued)

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

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

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

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

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

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

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

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

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

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

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

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

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