Docket No. 50-416

January 21, 1993

Mr. William T. Cottle
Vice President, Operations GGNS
Entergy Operations, Inc.
Post Office Box 756
Port Gibson, Mississippi 39150

Dear Mr. Cottle:

SUBJECT: ISSUANCE OF AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. NPF-29 - GRAND GULF NUCLEAR STATION, UNIT 1 (TAC NO. M84829)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 106 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment revises the Technical Specifications (TS) in response to your application dated October 29, 1992.

The amendment changes the TS to remove cycle-specific reactor physics parameters and incorporate them into a new document called the Core Operating Limits Report. The amendment was requested in response to Generic Letter 88-16.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:
Paul W. O'Connor, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 106 to NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION

Docket File	NRC/Local PDR	PD4-1 Reading	P. Noonan
M. Virgilio	OPA(2G5)	J. Larkins	PD4-1 Plant File
P. O'Connor(2)	OGC (15B18)	D. Hagan (3206)	T. Gody, Jr. (13E21)
G. Hill(4)	Wanda Jones (7103)	C. Grimes (11E22)	D. Pickett (13H15)
ACRS(10)	OC/LFMB (4503)	D. Verrelli, RII	R. Hall (13E21)
J. Roe (13A2)	C. Hoxie	H. Rathbun	

OFC	LA:PD4-1	I:PD4-1	PM:PD4-1	SRXB	OGC	D:PD4-1
NAME	PNoonan/v	HRathbun	PO'Connor	RJones	MYoung	GHubbard
DATE	12/1 /92	12/23/92	12/23/92	12/23/92	1/15/93	1/21/93

OFFICIAL RECORD COPY

Document Name: GG84829.AMD

930129

9301290166 930121
PDR ADOCK 05000416
P PDR

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

January 21, 1993

Docket No. 50-416

Mr. William T. Cottle
Vice President, Operations GGNS
Entergy Operations, Inc.
Post Office Box 756
Port Gibson, Mississippi 39150

Dear Mr. Cottle:

SUBJECT: ISSUANCE OF AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. NPF-29 - GRAND GULF NUCLEAR STATION, UNIT 1 (TAC NO. M84829)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 106 to Facility Operating License No. NPF-29 for the Grand Gulf Nuclear Station, Unit 1. This amendment revises the Technical Specifications (TS) in response to your application dated October 29, 1992.

The amendment changes the TS to remove cycle-specific reactor physics parameters and incorporate them into a new document called the Core Operating Limits Report. The amendment was requested in response to Generic Letter 88-16.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Paul W. O'Connor

Paul W. O'Connor, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.106 to NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

cc:

Mr. Donald C. Hintz, President
and Chief Executive Officer
Entergy Operations, Inc.
P. O. Box 31995
Jackson, Mississippi 39286-1995

Robert B. McGehee, Esquire
Wise, Carter, Child & Caraway
P. O. Box 651
Jackson, Mississippi 39205

Nicholas S. Reynolds, Esquire
Winston & Strawn
1400 L Street, N.W. - 12th Floor
Washington, D.C. 20005-3502

Mr. Jack McMillan, Director
Division of Solid Waste Management
Mississippi Department of Natural
Resources
P. O. Box 10385
Jackson, Mississippi 39209

President,
Claiborne County Board of Supervisors
Port Gibson, Mississippi 39150

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta St., Suite 2900
Atlanta, Georgia 30323

Mr. Michael J. Meisner
Director, Nuclear Licensing
Entergy Operations, Inc.
P. O. Box 756
Port Gibson, Mississippi 39150

Mr. D. L. Brannen, Vice President
Bechtel Power Corporation
P. O. Box 2166
Houston, Texas 77252-2166

Mr. Rudolph H. Bernhard
Senior Resident Inspector
U.S. Nuclear Regulatory Commission
Route 2, Box 399
Port Gibson, Mississippi 39150

Mr. C. R. Hutchinson
GGNS General Manager
Entergy Operations, Inc.
P. O. Box 756
Port Gibson, Mississippi 39150

The Honorable William J. Guste, Jr.
Attorney General
Department of Justice
State of Louisiana
P. O. Box 94005
Baton Rouge, Louisiana 70804-9005

Alton B. Cobb, M.D.
State Health Officer
State Board of Health
P. O. Box 1700
Jackson, Mississippi 39205

Office of the Governor
State of Mississippi
Jackson, Mississippi 39201

Mike Morre, Attorney General
Frank Spencer, Asst. Attorney General
State of Mississippi
Post Office Box 22947
Jackson, Mississippi 39225

Mr. John P. McGaha
Vice President, Operations Support
Entergy Operations, Inc.
P.O. Box 31995
Jackson, Mississippi 39286-1995

Mr. S. S. Chan
Project Engineer
Bechtel Power, Corporation
P.O. Box 2166
Houston, Texas 77252-2166

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

ENTERGY OPERATIONS, INC.

SYSTEM ENERGY RESOURCES, INC.

SOUTH MISSISSIPPI ELECTRIC POWER ASSOCIATION

MISSISSIPPI POWER AND LIGHT COMPANY

DOCKET NO. 50-416

GRAND GULF NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 106
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 October 29, 1992, 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. 106 , 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 90 days from its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

George T. Hubbard, Acting Director
Project Directorate IV-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the
Technical Specifications

Date of Issuance: January 21, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 106

FACILITY OPERATING LICENSE NO. NPF-29

DOCKET NO. 50-416

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE PAGES	INSERT PAGES
i	i
iv	iv
xx	xx
1-2	1-2
-	1-2a
2-1	2-1
3/4 2-1	3/4 2-1
3/4 2-2	----
3/4 2-3	3/4 2-2
3/4 2-4	3/4 2-3
3/4 2-5	----
3/4 2-6	----
3/4 2-6a	----
3/4 2-7	3/4 2-4
3/4 2-7a	----
3/4 2-7b	----
3/4 2-7c	----
B 3/4 2-1	B 3/4 2-1
B 3/4 2-7	B 3/4 2-7
5-5	5-5
6-19	6-19
-	6-19a
-	6-19b

INDEX

DEFINITIONS

SECTION

SECTION

DEFINITIONS (Continued) PAGE

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

SECTION PAGE

INDEX

ADMINISTRATIVE CONTROLS

CORE ALTERATION

1.7 CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement of the SRMs, IRMs, LPRMs, TIPs, or special movable detectors is not considered to be CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of the movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT (COLR)

1.7a The COLR is the Grand Gulf Nuclear Station specific document that provides core operating limits for the current reload cycle. These cycle-specific operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.11. Plant operation within these operating limits is addressed in individual Specifications.

CRITICAL POWER RATIO

1.8 The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the ANFB correlation to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.

DOSE EQUIVALENT I-131

1.9 DOSE EQUIVALENT I-131 shall be that concentration of I-131, microcuries per gram, which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AVERAGE PLANAR EXPOSURE

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

DRYWELL INTEGRITY

1.10 DRYWELL INTEGRITY shall exist when:

a. All drywell penetrations required to be closed during accident conditions are either:

   1. Capable of being closed by an OPERABLE drywell automatic isolation system, or

   2. Closed by at least one manual valve, blind flange, or deactivated automatic valve secured in its closed position, except for valves that are opened under administrative controls as permitted by Specification 3.6.4.

b. The drywell equipment hatch is closed and sealed.

c. The drywell airlock is in compliance with the requirements of Specification 3.6.2.3.

d. The drywell leakage rates are within the limits of Specification 3.6.2.2.

e. The suppression pool is in compliance with the requirements of Specification 3.6.3.1.

f. The sealing mechanism associated with each drywell penetration; e.g., welds, bellows or O-rings, is OPERABLE.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.1 THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

THERMAL POWER, High Pressure and High Flow

2.1.2 The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.06 during two loop operation and 1.07 during single loop operation with the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than the above limits and the reactor vessel steam dome pressure greater than 785 psig and core flow greater than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.1.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1325 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1325 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1325 psig within 2 hours and comply with the requirements of Specification 6.7.1.

SAFETY LIMITS (Continued)

REACTOR VESSEL WATER LEVEL

2.1.4 The reactor vessel water level shall be above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL CONDITIONS 3, 4 and 5

ACTION:

With the reactor vessel water level at or below the top of the active irradiated fuel, manually initiate the ECCS to restore the water level. Depressurize the reactor vessel as necessary for ECCS operation. Comply with the requirements of Specification 6.7.1.

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

During two loop operation or single loop operation, with an APLHGR exceeding the limits, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the required limits:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.

3/4.2.2 [DELETED]

3/4.2.3 MINIMUM CRITICAL POWER RATIO

LIMITING CONDITION FOR OPERATION

3.2.3 The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With MCPR less than the applicable MCPR limits, initiate corrective action within 15 minutes and restore MCPR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR shall be determined to be equal to or greater than the applicable MCPR limits.

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.

d. The provisions of Specification 4.0.4 are not applicable.

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 LHGR's shall be determined to be equal to or less than their allowable limits:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER,

c. Initially and at least once per 12 hours when the reactor is operating on a LIMITING CONTROL ROD PATTERN for LHGR, and

d. The provisions of Specification 4.0.4 are not applicable.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod to rod power distribution within an assembly. The Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for two loop operation are specified in the CORE OPERATING LIMITS REPORT (COLR).

For single-loop operation, a MAPLHGR limit corresponding to the product of the two loop MAPLHGR and a reduction factor specified in the COLR can be conservatively used to ensure that the PCT for single loop operation is bounded by the PCT for two loop operation.

The daily requirement for calculating APLHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control

MINIMUM CRITICAL POWER RATIO (Continued)

During initial start-up testing of the plant, a MCPR evaluation will be made at 25% of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate MCPR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating MCPR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that MCPR will be known following a change in THERMAL POWER or power shape, that could place operation exceeding a thermal limit.

3/4.2.4 LINEAR HEAT GENERATION RATE

This specification assures that the Linear Heat Generation Rate (LHGR) in any rod is less than the design linear heat generation even if fuel pellet densification is postulated.

The LHGR limits are multiplied by the smaller of either the flow dependent LHGR factor ($LHGRFAC_f$) or the power dependent LHGR factor ($LHGRFAC_p$) corresponding to the existing core flow and power state to ensure adherence to the fuel mechanical design bases during the limiting transient. $LHGRFAC_f$'s are generated to protect the core from slow flow runout transients. A curve is provided based on the maximum credible flow runout transient for Loop Manual operation. The result of a single failure or single operator error during operation in Loop Manual is the runout of only one loop because both recirculation loops are under independent control. $LHGRFAC_p$'s are generated to protect the core from plant transients other than core flow increases.

The daily requirements for calculating LHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate LHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating LHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that LHGR will be known following a change in THERMAL POWER or power shape that could place operation exceeding a thermal limit.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 800 fuel assemblies. Each fuel assembly shall contain fuel rods and water rods clad with Zircaloy cladding. Each fuel rod shall have a design nominal active fuel length of 150 inches. Reload fuel shall have mechanical, thermal-hydraulic and neutronic characteristics compatible with the initial core loading. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC Staff-approved codes and methods, shown to comply with all safety design bases, and are identified in the Core Operating Limits Report.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 193 control rod assemblies, each consisting of a cruciform array of stainless steel tubes containing a design nominal 143.7 inches of boron carbide, B_4C, powder surrounded by a cruciform shaped stainless steel sheath.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of:

1. 1250 psig on the suction side of the recirculation pump.

2. 1650 psig from the recirculation pump discharge to the outlet side of the discharge shutoff valve.

3. 1550 psig from the discharge shutoff valve to the jet pumps.

c. For a temperature of 575°F.

VOLUME

5.4.2 The total water and steam volume of the reactor vessel and recirculation system is approximately 22,000 cubic feet at a nominal T_{ave} of 533°F.

MONTHLY OPERATING REPORTS

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to main steam system 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.

CORE OPERATING LIMITS REPORT (COLR)

6.9.1.11 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 CORE OPERATING LIMITS REPORT (COLR) for the following:

a. The Average Planar Linear Heat Generation Rate (APLHGR) for Technical Specification 3.2.1.

b. The Minimum Critical Power Ratio (MCPR) for Technical Specification 3.2.3.

c. The Linear Heat Generation Rate (LHGR) for Technical Specification 3.2.4.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in the following documents. The appropriate revision/supplement number for each document shall be identified in the Core Operating Limits Report.

1) XN-NF-79-71(P), Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, Exxon Nuclear Company, Inc., Richland, WA. Approved by NRC letter dated October 24, 1986.

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

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

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

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

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

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

CORE OPERATING LIMITS REPORT (COLR) (Continued)

8) XN-NF-573(P), RAMPEX Pellet-Clad Interaction Evaluation Code for Power Ramps, Exxon Nuclear Company, Inc., Richland, WA. Approved by NRC letter dated August 28, 1990.

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

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-NR-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. Approved by NRC letter dated October 24, 1986.

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 of Pressure Drop in BWR Fuel Assemblies, Exxon Nuclear Company, Inc., Richland, WA.

The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, thermal-hydraulic limits, Emergency Core Cooling System (ECCS) limits, Nuclear limits such as shutdown margin, transient analysis limits, and accident limits) of the safety analysis are met.

The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Nuclear Regulatory Commission pursuant to Section 50.4 of 10 CFR Part 50 within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS.

6.10 RECORD RETENTION (Continued)

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7.1-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the Operational Quality Assurance Manual not listed in Section 6.10.1.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PSRC and the SRC.

l. Records of the service lives of all hydraulic and mechanical snubbers including the date at which the service life commences and associated installation and maintenance records.

m. Records of analyses required by the radiological environmental monitoring program.

n. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. NPF-29

ENTERGY OPERATIONS, INC., ET AL.

GRAND GULF NUCLEAR STATION, UNIT 1

DOCKET NO. 50-416

1.0 INTRODUCTION

By letter dated October 29, 1992, the licensee (Entergy Operations, Inc.), submitted a request for changes to the Grand Gulf Nuclear Station, Unit 1 (GGNS) Technical Specifications (TS). The proposed changes would modify specifications having cycle-specific parameter limits by replacing the values of those limits with a reference to the Core Operating Limits Report (COLR). The proposed changes also include the addition of the COLR to the Definitions Section and to the reporting requirements of the Administrative Controls Section of TS. Guidance on the proposed changes was developed by NRC on the basis of the review of a lead-plant proposal submitted on the Oconee plant docket that was endorsed by the Babcock and Wilcox Owners Group. This guidance was provided to all power reactor licensees and applicants by Generic Letter 88-16, dated October 4, 1988.

2.0 EVALUATION

The licensee's proposed changes to the TS are in accordance with the guidance provided by Generic Letter 88-16 and are addressed below.

(A) The Definitions Section of the TS was modified to include a definition of the COLR that requires cycle/reload-specific parameter limits to be established on a unit-specific basis in accordance with an NRC-approved methodology that maintains the limits of the safety analysis. The definition notes that plant operation within these limits is addressed by individual specifications.

(B) The following specifications were revised to replace the values of cycle-specific parameter limits with a reference to the COLR that provides these limits.

(1) Specification 3.2.1 and Bases 3/4.2.1

The Average Planar Linear Heat Generation Rate (APLHGR) limits for this specification and for these Bases are specified in the COLR.

(2) Specification 3/4.2.3

The Minimum Critical Power Ratio (MCPR) limits for this specification are specified in the COLR.

(3) Specification 3/4.2.4 and Bases 3/4.2.4

The Linear Heat Generation Rate (LHGR) limits for this specification and for these Bases are specified in the COLR.

(C) Specification 6.9.1.11, COLR, was added to the reporting requirements of the Administrative Controls Section of the TS. This specification requires that the COLR be submitted, upon issuance, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector. The report provides the values of cycle-specific parameter limits that are applicable for the current fuel cycle. Furthermore, this specification requires that the values of these limits be established using NRC-approved methodologies and be consistent with all applicable limits of the safety analysis. The approved methodologies are the following:

(1) XN-NF-79-71(P), Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors, Exxon Nuclear Company, Inc., Richland, WA. Approved by NRC letter dated October 24, 1986.

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

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

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

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

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

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

(8) XN-NF-573(P), RAMPEX Pellet Clad Interaction Evaluation Code for Power Ramps, Exxon Nuclear Company, Inc., Richland, WA. Approved by NRC letter dated August 28, 1990.

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

(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, MCPRp 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. Approved by NRC letter dated October 24, 1986.

(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 of Pressure Drop in BWR Fuel Assemblies, Exxon Nuclear Company, Inc., Richland, WA.

References (1), (8), and (15) were not published by the vendor as approved topical reports; therefore, the references have been clarified by adding the NRC approval dates.

Finally, the specification requires that all changes in cycle-specific parameter limits be documented in the COLR before each reload cycle or remaining part of a reload cycle and submitted upon issuance to NRC, prior to operation with the new parameter limits.

In addition, the licensee has revised the description of fuel assembly design features in TS Section 5.3, Reactor Core. The reference to the initial core design is deleted and that core designs be developed and analyzed using NRC-approved codes and methods. Also, the design description requires fuel assembly types to be identified in the COLR.

On the basis of the review of the above items, the NRC staff concludes that the licensee provided an acceptable response to those items as addressed in the NRC guidance in Generic Letter 88-16 on modifying cycle-specific parameter limits in TS. Because plant operation continues to be limited in accordance with the values of cycle-specific parameter limits that are established using an NRC-approved methodology, the NRC staff concludes that this change is administrative in nature and there is no impact on plant safety as a consequence. Accordingly, the staff finds the proposed changes acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Mississippi State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding (57 FR 55578). This amendment also changes recordkeeping, reporting or administrative procedures or requirements. Accordingly, with respect to these items, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: T. Huang, SRXB/DEST
T. Dunning, OTSB/DOEA
H. Rathbun, PDIV-1/DRPW

Date: January 21, 1993