DOCKET FILE

Docket No. 50-416 March 9, 1992

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. 93 TO FACILITY OPERATING LICENSE NO. NPF-29 - GRAND GULF NUCLEAR STATION, UNIT 1 (TAC NO. M80590)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 93 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 May 30, 1991.

The amendment deletes references in the Technical Specifications to operation of the Reactor Recirculation System in the Non-Loop Manual (automatic) mode of flow control. This mode of operation was eliminated by a 10 CFR 50.59 evaluation and an approved design change implemented during Refueling Outage 4 in 1990.

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. 93 to NPF-29
2. Safety Evaluation

cc w/enclosures:
See next page

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

March 9, 1992

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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. 93
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 May 30, 1991, 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. 93 , 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.

FOR THE NUCLEAR REGULATORY COMMISSION

John T. Larkins

John T. Larkins, 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: March 9, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 93

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.

REMOVE PAGES	INSERT PAGES
3/4 2-5	3/4 2-5
3/4 2-7b	3/4 2-7b
3/4 4-1	3/4 4-1
3/4 4-1a	3/4 4-1a
B 3/4 2-4	B 3/4 2-4
B 3/4 2-4a	B 3/4 2-4a
B 3/4 2-6	B 3-4 2-6
B 3/4 2-7	B 3-4 2-7
B 3/4 2-7a	B 3/4 2-7a
B 3/4 4-1	B 3/4 4-1
B 3/4 4-1a	B 3/4 4-1a

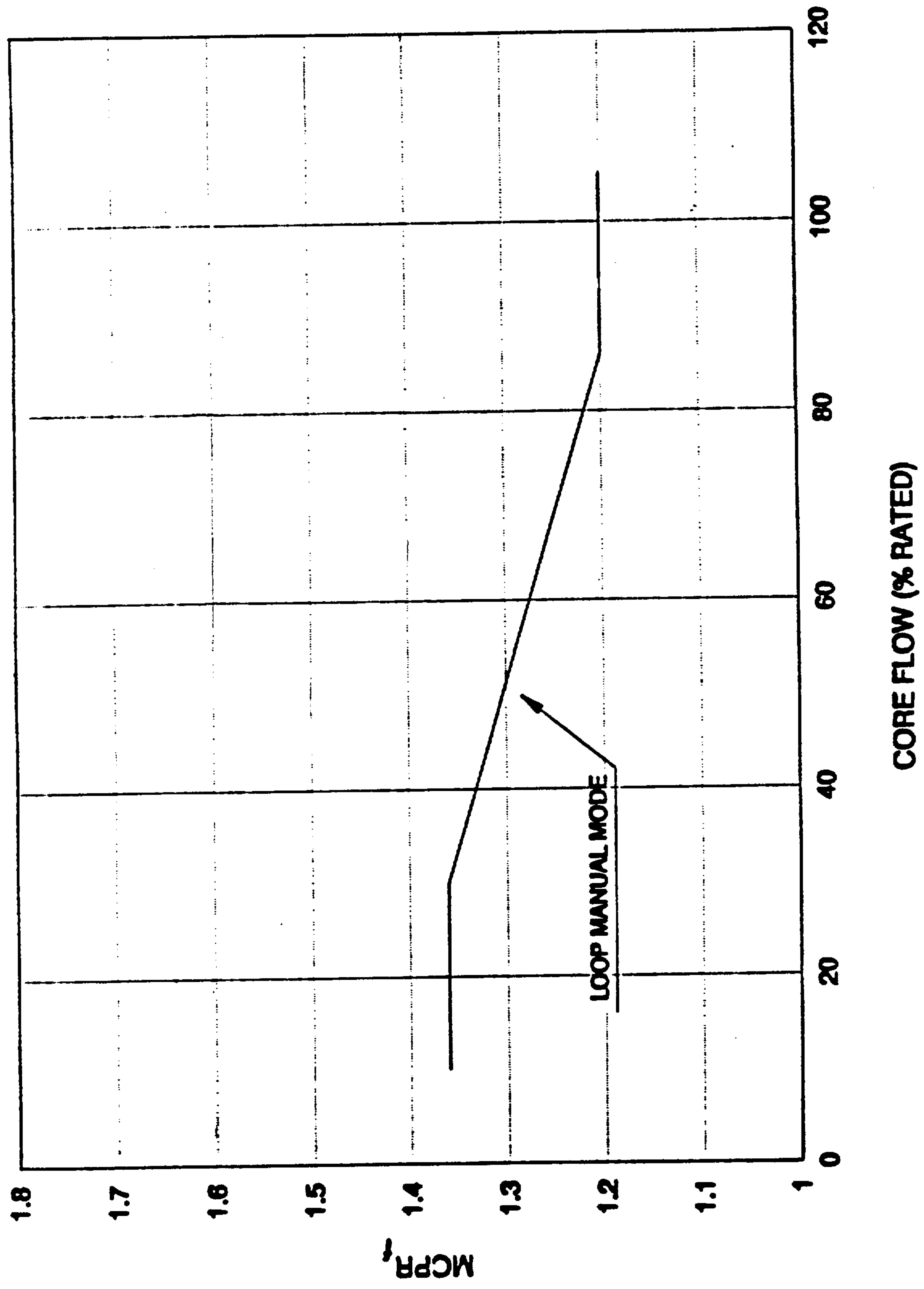

FIGURE 3.2.3-1 MCPR$_f$

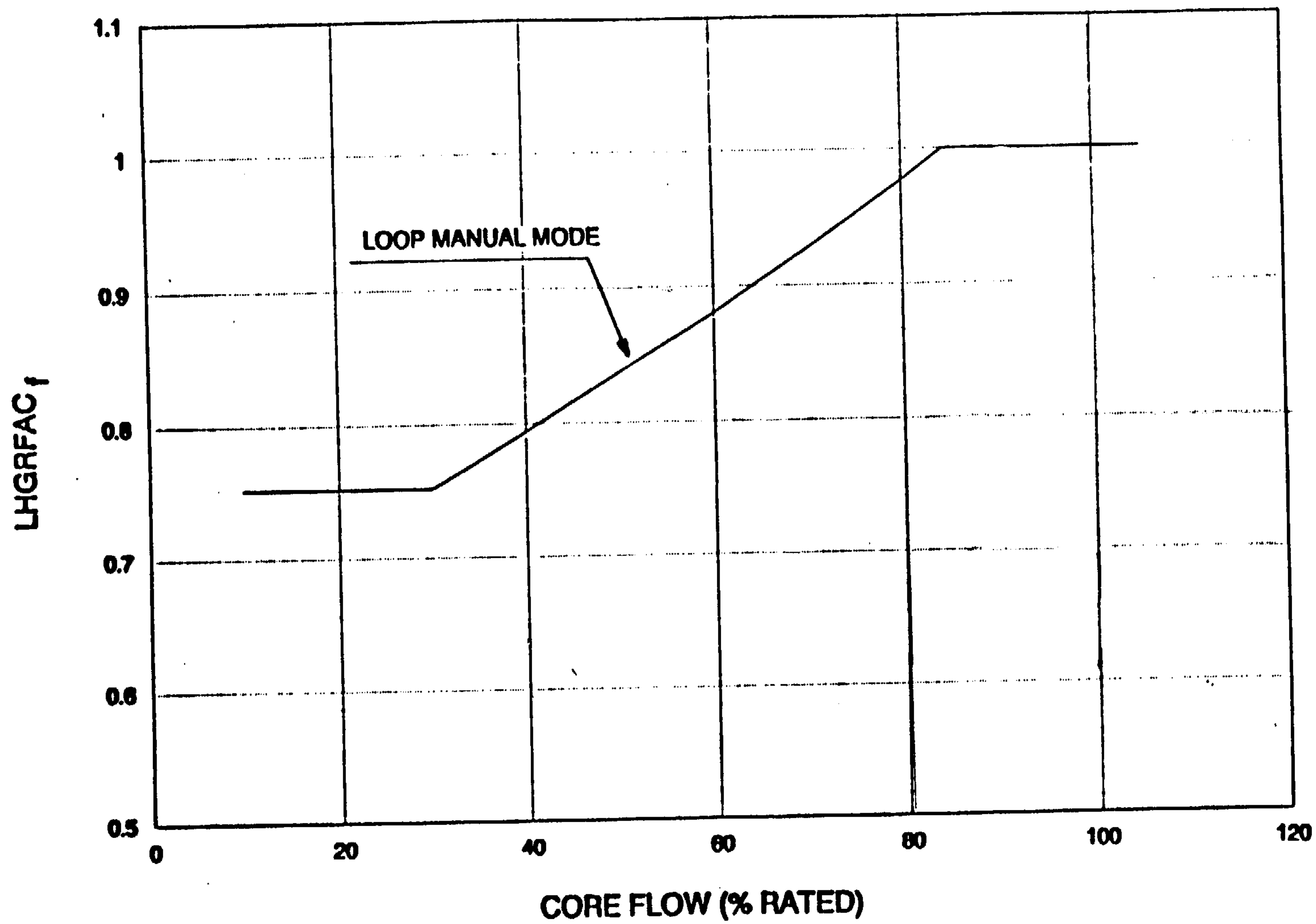

FIGURE 3.2.4-2 LHGRFAC$_f$

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 The reactor coolant recirculation system shall be in operation with either:

a. Two recirculation loops operating with limits and setpoints per Specifications 2.2.1, 3.2.1, and 3.3.6, or

b. A single recirculation loop operating with:

   1. A volumetric loop flow rate less than 44,600 gpm, and

   2. Limits and setpoints per Specifications 2.2.1, 3.2.1, and 3.3.6.

Operation is not permissible in Regions A, B or C as specified in Figure 3.4.1.1-1 except that operation in Region C is permissible during control rod withdrawals for startup.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With no reactor coolant system recirculation loops in operation and the reactor mode switch in the run position, immediately place the reactor mode switch in the shutdown position.

b. With operation in Region A as specified in Figure 3.4.1.1-1, immediately place the reactor mode switch in the shutdown position.

c. With operation in regions B or C as specified in Figure 3.4.1.1-1, observe the indicated APRM, neutron flux noise level. With a sustained APRM neutron flux noise level greater than 10% peak-to-peak of RATED THERMAL POWER, immediately place the reactor mode switch in the shutdown position.

d. With operation in Region B as specified in Figure 3.4.1.1-1, immediately initiate action to either reduce THERMAL POWER by inserting control rods or increase core flow if one or more recirculation pumps are on fast speed by opening the flow control valve to within Region D of Figure 3.4.1.1-1 within 2 hours.

e. With operation in Region C as specified in Figure 3.4.1.1-1, unless operation in this region is for control rod withdrawals during startup, immediately initiate action to either reduce THERMAL POWER or increase core flow to within Region D of Figure 3.4.1.1-1 within 2 hours.

f. During single loop operation, with the volumetric loop flow rate greater than the above limit, immediately initiate corrective action to reduce flow to within the above limit within 30 minutes.

*See Special Test Exception 3.10.4.

LIMITING CONDITION FOR OPERATION (Continued)

g. During single loop operation, with temperature differences exceeding the limits of SURVEILLANCE REQUIREMENT 4.4.1.1.5, suspend the THERMAL POWER or recirculation loop flow increase.

h. With a change in reactor operating conditions, from two recirculation loops operating to single loop operation, or restoration of two loop operation, the limits and setpoints of Specifications 2.2.1, 3.2.1, and 3.3.6 shall be implemented within 8 hours or declare the associated equipment inoperable (or the limits to be "not satisfied"), and take the ACTIONS required by the referenced specifications.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 At least once per 24 hours, the reactor coolant recirculation system shall be verified to be in operation and not in Regions A, B or C as specified in Figure 3.4.1.1-1 except that operation in Region C is permissible during control rod withdrawals for startup.

4.4.1.1.2 Each reactor coolant system recirculation loop flow control valve in an operating loop shall be demonstrated OPERABLE at least once per 18 months by:

a. Verifying that the control valve fails "as is" on loss of hydraulic pressure at the hydraulic unit, and

b. Verifying that the average rate of control valve movement is:

    1. Less than or equal to 11% of stroke per second opening, and

    2. Less than or equal to 11% of stroke per second closing.

4.4.1.1.3 During single loop operation, verify that the volumetric loop flow rate of the loop in operation is within the limit at least once per 24 hours.

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limit MCPRs at steady state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta CPR. When added to the Safety Limit MCPR, the required operating limit MCPR of Specification 3.2.3 is obtained. The power-flow map of Figure B 3/4 2.3-1 defines the analytical basis for generation of the MCPR operating limits (References 2 and 3).

MCPR operating limits are defined as functions of exposure ($MCPR_e$), flow ($MCPR_f$), and power ($MCPR_p$). The limit to be used at a given operating state is the highest of these three limits.

The purpose of the $MCPR_e$ is to define operating limits for all anticipated exposures during the Cycle. The $MCPR_e$ limits are established for a set of exposure intervals. The limiting transients are analyzed at the limiting exposure for each interval.

The $MCPR_e$ operating limits are established based on the largest delta-CPR calculated at the limiting exposure and ensure that the MCPR safety limit will not be exceeded during the most limiting transient in each of the exposure intervals.

The purpose of the $MCPR_f$ and $MCPR_p$ is to define operating limits at other than rated core flow and power conditions for all exposures during the cycle.

The $MCPR_f$s are established to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured. The reference core flow increase event used to establish the $MCPR_f$ is a hypothesized slow flow runout to maximum, that does not result in a scram from neutron flux overshoot exceeding the APRM neutron flux-high level (Table 2.2.1-1 item 2). The result of a single failure or single operator error during Loop Manual operation is the runout of one loop because the two recirculation loops are under independent control. With this basis, the $MCPR_f$ curve was generated from

MINIMUM CRITICAL POWER RATIO (Continued)

a series of steady state core thermal hydraulic calculations performed at several core power and flow conditions along the steepest flow control line. In the actual calculations a conservative highly steep generic representation of the 105% steam flow rodline flow control line has been used. Assumptions used in the original calculations of this generic flow control line were consistent with a slow flow increase transient duration of several minutes: (a) the plant heat balance was assumed to be in equilibrium, and (b) core xenon concentration

MINIMUM CRITICAL POWER RATIO (Continued)

was assumed to be constant. The generic flow control line is used to define several core power/flow states at which to perform steady-state core thermal-hydraulic evaluations.

The Loop Manual mode of operation was analyzed. Consistent with the single failure/single operator error criterion, one loop runout was postulated for Loop Manual operation. The maximum core flow at loop runout was assumed to be 110% of rated flow. Peaking factors were selected such that the MCPR for the bundle with the least margin of safety would not decrease below the MCPR Safety Limit.

The $MCPR_p$ is established to protect the core from plant transients other than core flow increase including the localized rod withdrawal error event. Core power dependent setpoints are incorporated (incremental control rod withdrawal limits) in the Rod Withdrawal Limiter (RWL) System Specification (3.3.6). These setpoints allow greater control rod withdrawal at lower core powers where core thermal margins are large. However, the increased rod withdrawal requires higher initial MCPR's to assure the MCPR safety limit Specification (2.1.2) is not violated. The analyses that establish the power dependent MCPR requirements that support the RWL system are presented in Reference 4. For core power below 40% of RATED THERMAL POWER, where the EOC-RPT and the reactor scrams on turbine stop valve closure and turbine control valve fast closure are bypassed, separate sets of $MCPR_p$ limits are provided for high and low core flows to account for the significant sensitivity to initial core flows. For core power above 40% of RATED THERMAL POWER, bounding power-dependent MCPR limits were developed. The abnormal operating transients analyzed for single loop operation are discussed in Reference 5 and the appropriate cycle-specific documents. No change to the MCPR operating limit is required for single loop operation.

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin.

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 of Figure 3.2.4-1 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 requirement 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.

References:

1. XN-NF-80-19(A), Volume 2, "Exxon Nuclear Methodology for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model," Exxon Nuclear Company, September 1982.

2. General Electric Company, "Maximum Extended Operating Domain Analysis," March 1986.

3. AECM-86/0066, "Final Summary Startup Test Report 12," Letter, O.D. Kingsley, MP&L, to J. N. Grace, NRC, February 1986.

4. XN-NF-825(P)(A), Supplement 2, "BWR/6 Generic Rod Withdrawal Analysis; $MCPR_p$ for All Plant Operations Within the Extended Operation Domain," Exxon Nuclear Company, October 1986.

5. GGNS Reactor Performance Improvement Program, Single Loop Operation Analysis, General Electric Final Report, February 1986.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor core coolant recirculation loop inoperable has been evaluated and found to remain within design limits and safety margins provided certain limits and setpoints are modified. The "GGNS Single Loop Operation Analysis" identified the applicable fuel thermal limits and APRM setpoint modifications necessary to maintain the same margin of safety for single loop operation as is available during two loop operation. Additionally, loop flow limitations are established to ensure vessel internal vibration remains within limits.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation. During two loop operation, recirculation loop flow mismatch limits are in compliance with ECCS LOCA analysis design criteria. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. In cases where the mismatch limits cannot be maintained, continued operation is permitted with one loop in operation.

The power/flow operating map is divided into four (4) regions. Regions A and B are restricted from operations. They include the operating area above the 80% rod-line and below 40% core flow. Region C includes the operating area above the 80% rod-line and between 40% and 45% core flow. Operation in Region C is allowed only for control rod withdrawals during startup for required fuel preconditioning. Region D consists of the rest of the operating map. No core thermal-hydraulic stability related restrictions are applied to Region D since the potential onset of core thermal-hydraulic instabilities is not predicted within Region D.

The definition of Regions A, B and C is based on BWR stability operational data and required operator actions. Although a large margin to onset of instability was observed in Regions A, B and C during GGNS stability tests for typical operating configuration, a conservative approach is adopted in the specification.

With no reactor coolant system recirculation loops in operation, and the reactor mode switch in the Run position, an immediate reactor shutdown is required. Reactor shutdown is not required when recirculation pump motors are de-energized during recirculation pump speed transfers. Upon entry to Region A an immediate reactor shutdown is required. Upon entry to Region B or Region C, unless operation in Region C is for control rod withdrawals during startup, either a reduction of THERMAL POWER to below the 80% rod-line by control rod insertion or an increase in core flow to exit the region by opening the recirculation loop FCV is required.

Per the specification, the APRM neutron flux noise level should be observed while in Regions B and C. In the unlikely event in which a sustained

3/4.4.1 RECIRCULATION SYSTEM (Continued)

APRM neutron flux noise level exceeding 10% peak-to-peak of RATED THERMAL POWER is observed, an immediate reactor shutdown is required.

The APRM neutron flux noise level of 10% peak-to-peak of RATED THERMAL POWER is established to ensure early detection of core thermal-hydraulic instabilities. APRM neutron flux noise levels in the range of 2% to 6% peak-to-peak of RATED THERMAL POWER were observed for the Grand Gulf Reactor during its first three operating cycles and at different power/flow operating conditions. This represents the typical APRM neutron flux noise level for stable operations of the Grand Gulf Reactor.

The 10% peak-to-peak of RATED THERMAL POWER noise level provides adequate margin to thermal limits in the unlikely event of uncontrolled limit cycle oscillations while in Regions B and C, including the even less likely event of regional oscillations. The required operator action of an immediate reactor shutdown upon entry to Region A and upon detection of sustained APRM neutron flux noise level greater than the 10% peak-to-peak of RATED THERMAL POWER assures that an adequate margin to thermal limits will be maintained at all times.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 100°F. During single loop operation, the condition may exist in which the coolant in the bottom head of the vessel is not circulating. These differential temperature criteria are also to be met prior to power or flow increases from this condition.

The recirculation flow control valves provide regulation of individual recirculation loop drive flows; which, in turn, will vary the flow rate of coolant through the reactor core over a range consistent with the rod pattern and recirculation pump speed. The recirculation flow control system consists of the electronic and hydraulic components necessary for individual positioning of the two hydraulically actuated flow control valves. Solid state control logic will generate a flow control valve "motion inhibit" signal in response to any one of several hydraulic power unit or analog control circuit failure signals. The "motion inhibit" signal causes hydraulic power unit shutdown and hydraulic isolation such that the flow control valve fails "as is." This design feature insures that the flow control valves do not respond to potentially erroneous control signals.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 93 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 May 30, 1991, the licensee (Entergy Operations, Inc.), submitted a request for changes to the Grand Gulf Nuclear Station, Unit 1 (GGNS) Technical Specifications (TS). The requested changes would delete from the TS several references to operation of the Reactor Recirculation System in the Non-Loop Manual (automatic) mode of flow control.

2.0 EVALUATION

The Reactor Flow Control system was originally capable of controlling the two-loop flow control valves either individually in Loop Manual (manual) flow control or together in Non-Loop Manual (automatic) flow control. The automatic mode was comprised of circuitry that adjusted total core flow by simultaneous signals to both valves. The adjustments maintained a desired turbine-generator output, reactor neutron flux level, or total recirculation drive flow as selected by the operator.

During the fourth Refueling Outage, in accordance with 10 CFR 50.59, the licensee implemented a design change that permanently disabled the automatic mode of flow control. The automatic mode was removed by installing wiring in the circuitry that prevents transferring the flow control system out of the Loop Manual mode under any circumstances.

There are no TS operating conditions or actions requiring the availability of the automatic mode of flow control. Both operating modes are unique and function independently of each other. The choice of operating in the automatic or manual modes is based on the preference of the operator. As this is a non-safety related system and with the automatic mode permanently disabled, references to and inferences from this mode are unnecessary and potentially misleading. The revisions to the TS remove these references and inferences and include removal of thermal operating limit curves strictly associated with the automatic mode of operation.

The staff concludes that these changes are not safety-significant and are 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

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. 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 the amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 31433). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 FR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the 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 Contributor: R. Twigg

Date: March 9, 1992