Docket

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

March 29, 1989

Docket No. 50-458

Gulf States Utilities
ATTN: Mr. James C. Deddens
Senior Vice President (RBNG)
Post Office Box 220
St. Francisville, Louisiana 70775

Dear Mr. Deddens:

SUBJECT: CORRECTIONS TO AMENDMENT NOS. 31 AND 33 TO FACILITY OPERATING LICENSE NO. NPF-47 - RIVER BEND STATION, UNIT 1

On November 22, 1988 and January 30, 1989, the Commission issued Amendment Nos. 31 and 33 respectively to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. Amendment No. 31 revised the Technical Specifications (TS) to allow single recirculation loop operation. Amendment No. 33 changed the TS to provide the bundle Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and the higher Linear Heat Generation Rate (LHGR) limits for the new fuel being added for the second reload. Amendment No. 33 also revised TS Section 5, Design Features, Item 5.3.1 to generalize the fuel details to allow referencing future fuel designs included in the "General Electric Standard Application for Reactor Fuels" (GESTAR), TS Bases 3/4.2.1 to reference TS 3.2.1 instead of referencing individual MAPLHGR curves, and Figure 3.2.3-1 to show the revised flow dependent Minimum Critical Power Ratio (MCPR) operating limit curve.

TS page 3/4 4-2 of Amendment No. 31 contained a typographical error in which reference was made to "(Specification 3.4.1.1 ACTION c)." This correction now references ACTION g.

TS page 3/4 2-1 issued with Amendment No. 33 inadvertently omitted changes to this page issued with Amendment No. 31. Also, TS page B 3/4 2-2 issued as an overleaf page to B 3/4 2-1 did not contain changes issued in Amendment No. 31.

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This page should have had a change issued with Amendment No. 33. Corrected pages 3/4 2-1, 3/4 4-2 and B 3/4 2-1 and their corresponding overleaf pages are enclosed.

Please accept our apology for any inconvenience this error may have caused you.

Sincerely,

/s/

Walter A. Paulson, Project Manager
Project Directorate - IV
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

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As stated

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Walter A. Paulson

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Division of Reactor Projects - III,
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SURVEILLANCE REQUIREMENTS

4.4.1.1.1 Each reactor coolant system recirculation loop flow control valve 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 control 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

4.4.1.1.2 Establish a baseline APRM and LPRM* neutron flux noise valve within the regions for which monitoring is required (Specification 3.4.1.1 ACTION g) within 2 hours of entering the region for which monitoring is required unless baselining has previously been performed in the region since the last refueling outage.

4.4.1.1.3 Initially, within 1 hour upon entry into single loop operation and once per 12 hours thereafter, verify that:

a. THERMAL POWER is less than or equal to 70% of RATED THERMAL POWER, and

b. The recirculation flow control system is in the Loop Manual (Position Control) mode, and

c. The volumetric recirculation flow rate is less than or equal to 33,000 gpm.

4.4.1.1.4 With one reactor coolant system recirculation loop not in operation, and either THERMAL POWER less than or equal to 30% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is less than or equal to 50% of rated recirculation loop flow, within 15 minutes prior to an increase in THERMAL POWER or recirculation loop flow, verify that the following differential temperature requirements are met:

a. $\leq$ 100°F between reactor vessel steam space coolant and bottom head drain line coolant, and

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*Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of core should be monitored.

SURVEILLANCE REQUIREMENTS (Continued)

b. ≤ 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel**, and

c. ≤ 50°F between the reactor coolant within the loop not in operation and the operating loop.**

**With one recirculation loop not in operation and isolated, the differential temperature requirements of Surveillance Requirement 4.4.1.1.4b and c are not applicable and the provision of Surveillance Requirement 4.0.4 are not applicable with respect to Surveillance Requirement 4.4.1.1.4b and c.

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) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits shown in Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, 3.2.1-7 and 3.2.1-8.* The limits of Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, 3.2.1-7 and 3.2.1-8 shall be reduced to a value of 0.84 times the two recirculation loop operation limit when in single loop operation.

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

ACTION:

With an APLHGR exceeding the limits of Figure 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, 3.2.1-7 or 3.2.1-8, 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 limits determined from Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5, 3.2.1-6, 3.2.1-7 and 3.2.1-8:

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.

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*The limits on Figures 3.2.1-7 and 3.2.1-8 are to be used only for manual calculations.

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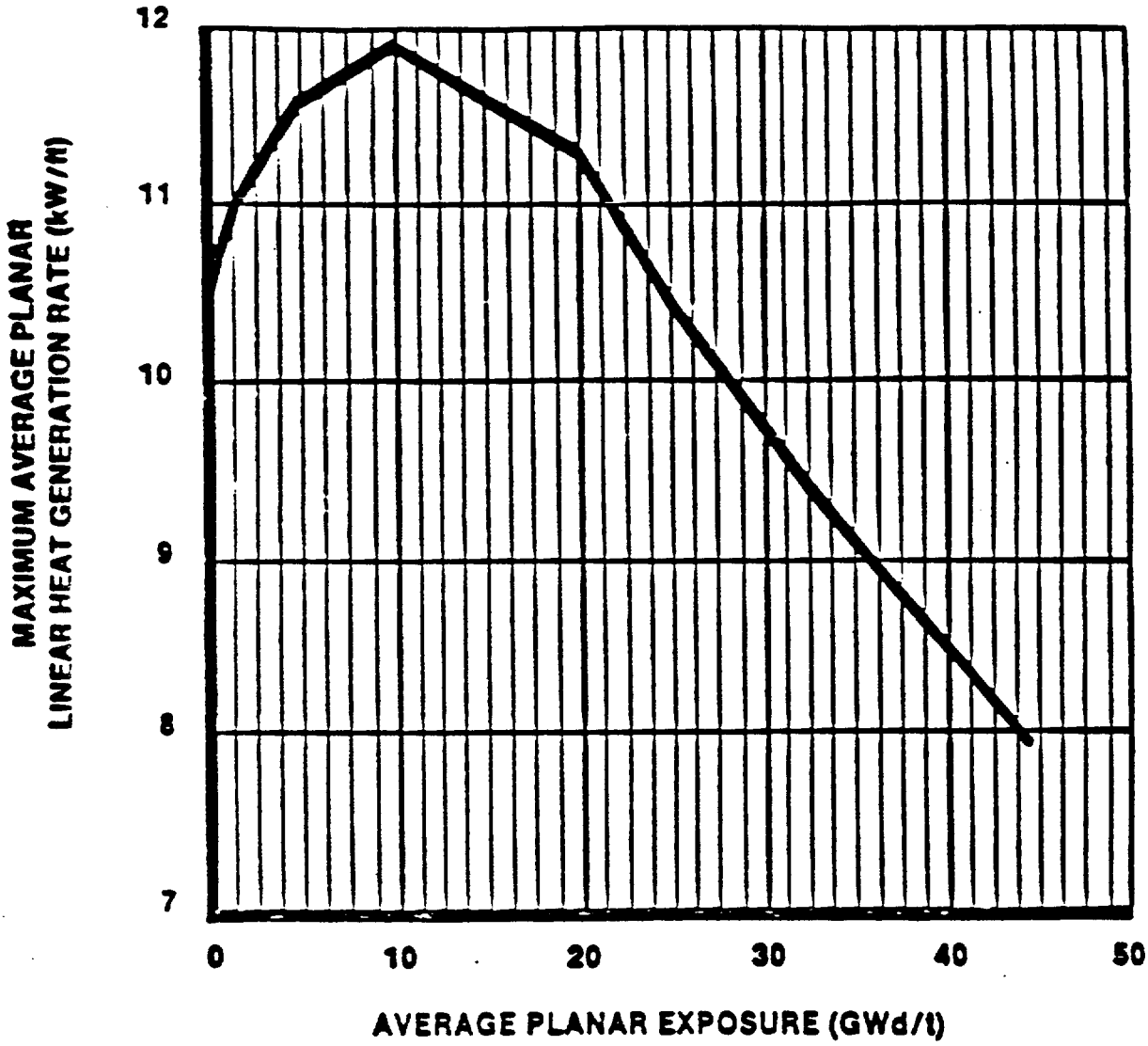

FIGURE 3.2.1-1
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE BP8SRB094

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

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 peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure-dependent steady state gap conductance and rod-to-rod local peaking factor. The Technical Specification AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in Specification 3.2.1.

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 rod changes. The requirement to calculate APLHGR 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 APLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that APLHGR will be known following a change in THERMAL POWER or power shape that could place operation into a condition exceeding a thermal limit.

The calculational procedure used to establish the APLHGR shown in Specification 3.2.1 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in NEDE-20566[1]. Differences in this analysis compared to previous analyses can be broken down as follows.

a. Input Changes

1. Corrected Vaporization Calculation - Coefficients in the vaporization correlation used in the REFLOOD code were corrected.

2. Incorporated more accurate bypass areas - The bypass areas in the top guide were recalculated using a more accurate technique.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

3. Corrected guide tube thermal resistance.

4. Correct heat capacity of reactor internals heat nodes.

b. Model Change

1. Core CCFL pressure differential - 1 psi - Incorporate the assumption that flow from the bypass to lower plenum must overcome a 1 psi pressure drop in core.

2. Incoporate NRC pressure transfer assumption - The assumption used in the SAFE-REFLOOD pressure transfer when the pressure is increasing was changed.

A few of the changes affect the accident calculation irrespective of CCFL. These changes are listed below.

a. Input Change

1. Break Areas - The DBA break area was calculated more accurately.

b. Model Change

1. Improved Radiation and Conduction Calculation - Incorporation of CHASTE-05 for heatup calculation.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B 3.2.1-1.

For plant operation with a single recirculation loop, the MAPLHGR limits of figures 3.2.1-1 through 3.2.1-8 are multiplied by 0.84. The constant factor 0.84 is derived from LOCA analyses initiated from single recirculation loop operation to account for earlier boiling transition at the limiting fuel mode compared to the standard LOCA evaluations.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased simulated thermal power-high scram trip setpoint and the flow biased neutron flux-upscale control rod block trip setpoints of the APRM instruments must be adjusted for both two recirculation loop operation and single recirculation loop operation to ensure that MCPR does not become less than the fuel cladding safety limit or that > 1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification, when the combination of THERMAL POWER and CMFLPD indicates a peak power distribution, to ensure that an LHGR transient would not be increased in degraded conditions.