November 22, 1988

Docket No. 50-458

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

Dear Mr. Deddens:

SUBJECT: RIVER BEND STATION, UNIT 1 - AMENDMENT NO. 31 TO FACILITY OPERATING LICENSE NO. NPF-47 (TAC NO. 67878)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 31 to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 6, 1988, as supplemented October 20, 1988.

The amendment revises the Technical Specifications to allow single recirculation loop operation.

A copy of our Safety Evaluation and Notice of Issuance are also enclosed.

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

Enclosures:
1. Amendment No. 31 to License No. NPF-47
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

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C/P 1

DOCUMENT NAME: RIVER BEND AMEND TAC 67878

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

November 22, 1988

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Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

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See next page

Mr. James C. Deddens
Gulf States Utilities Company

River Bend Nuclear Plant

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

GULF STATES UTILITIES COMPANY

DOCKET NO. 50-458

RIVER BEND STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 31
License No. NPF-47

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

   A. The application for amendment by Gulf States Utilities Company (the licensee) dated April 6, 1988, as supplemented October 20, 1988, 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, as amended, 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 license 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-47 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 31 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. GSU shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

José A. Calvo

Jose A. Calvo, Director
Project Directorate - IV
Division of Reactor Projects - III, IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 22, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 31

FACILITY OPERATING LICENSE NO. NPF-47

DOCKET NO. 50-458

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contains a vertical line indicating the area of change. Overleaf pages are provided to maintain document completeness.

REMOVE PAGES	INSERT PAGES
2-1	2-1
2-4	2-4
2-5	2-5
B 2-1	B 2-1
3/4 2-1	3/4 2-1
3/4 2-7	3/4 2-7
-	3/4 2-7a
3/4 3-62	3/4 3-62
3/4 4-1	3/4 4-1
-	3/4 4-1a
3/4 4-2	3/4 4-2
-	3/4 4-2a
3/4 4-3	3/4 4-3
3/4 4-4	3/4 4-4
-	3/4 4-4a
3/4 4-5	3/4 4-5
B 3/4 1-2	B 3/4 1-2
B 3/4 2-2	B 3/4 2-2
B 3/4 2-3	B 3/4 2-3
B 3/4 2-4	B 3/4 2-4
B 3/4 4-1	B 3/4 4-1
-	B 3/4 4-1a
B 3/4 4-2	B 3/4 4-2

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 1.07 with two recirculation loop operation and shall not be less than 1.08 with single recirculation loop operation with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With MCPR less than 1.07 with two recirculation loop operation or less than 1.08 with single recirculation loop operation and the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 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 above 1325 psig, as measured in the reactor vessel steam dome, 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, after depressurizing the reactor vessel, if required. Comply with the requirements of Specification 6.7.1.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-High, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-High		
1) Two Recirculation Loop Operation		
a) Flow Biased	≤ 0.66 W+48%, with a maximum of	≤ 0.66 W+51%, with a maximum of
b) High Flow Clamped	≤ 111.0% of RATED THERMAL POWER	≤ 113.0% of RATED THERMAL POWER
2) Single Recirculation Loop Operation		
a) Flow Biased	< 0.66 W+42.7%, with a maximum of	< 0.66 W+45.7%, with a maximum of
b) High Flow Clamped	< 111.0% of RATED THERMAL POWER	< 113.0% of RATED THERMAL POWER
c. Neutron Flux-High	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1064.7 psig	≤ 1079.7 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 9.7 inches above instrument zero*	≥ 8.7 inches above instrument zero
5. Reactor Vessel Water Level-High, Level 8	≤ 51.0 inches above instrument zero*	≤ 52.1 inches above instrument zero
6. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed

*See Bases Figure B 3/4 3-1.

TABLE 2.2.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
7. Main Steam Line Radiation - High	$\leq$ 3.0 x full power background	$\leq$ 3.6 x full power background
8. Drywell Pressure - High	$\leq$ 1.68 psig	$\leq$ 1.88 psig
9. Scram Discharge Volume Water Level - High		
a. Level Transmitter - LISN601A and B	$\leq$ 49"	$\leq$ 53"
LISN601C and D	$\leq$ 49"	$\leq$ 51.7"
b. Float Switches - LSN013A and B	$\leq$ 48.76"	$\leq$ 53.50"
LSN013C and D	$\leq$ 46.88"	$\leq$ 49.00"
10. Turbine Stop Valve - Closure	$\leq$ 5% closed	$\leq$ 7% closed
11. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	$\geq$ 530 psig	$\geq$ 465 psig
12. Reactor Mode Switch Shutdown Position	NA	NA
13. Manual Scram	NA	NA

*See Bases Figure B 3/4 3-1.

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit such that the MCPR is not less than 1.07 for two recirculation loop operation and 1.08 for single recirculation loop operation. MCPR greater than 1.07 for two recirculation loop operation and 1.08 for single recirculation loop operation represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the GE Critical Power correlation (Reference 1) is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28,000 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28,000 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.

2.1.2 THERMAL POWER, High Pressure and High Flow

The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using a statistical model that combines all of the uncertainties in the operating parameters and in the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical power correlation. Details of the fuel cladding integrity safety limit calculation are given in Reference 1. Reference 1 includes a tabulation of the uncertainties used in the determination of the Safety Limit MCPR and of the nominal values of parameters used in the Safety Limit MCPR statistical analysis.

Reference

1. "General Electric Standard Application for Reactor Fuel (GESTAR)," NEDE-24011-P-A-8.

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, and 3.2.1-6. The limits of Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5 and 3.2.1-6 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 or 3.2.1-6, 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 and 3.2.1-6:

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.

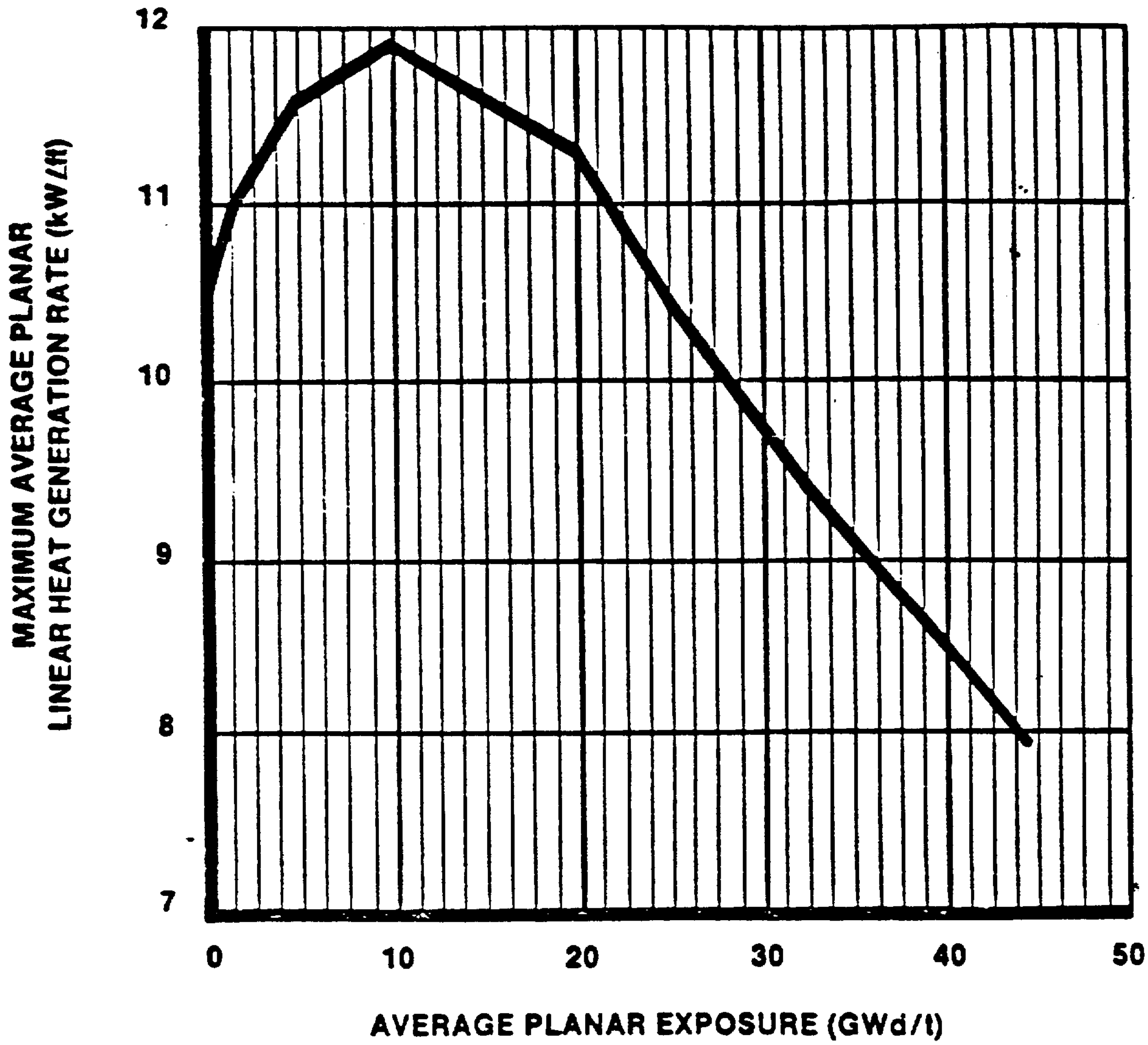


FIGURE 3.2.1-1
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR)
VERSUS AVERAGE PLANAR EXPOSURE BP8SRB094

3/4.2.2 APRM SETPOINTS

LIMITING CONDITION FOR OPERATION

3.2.2 The APRM flow biased simulated thermal power-high scram trip setpoint (S) and flow biased neutron flux-upscale control rod block trip setpoint (S_{RB}) shall be established according to the following relationships:

a. Two Recirculation Loop Operation

Trip Setpoint	Allowable Value
$S \leq (0.66W + 48\%)T$	$S \leq (0.66W + 51\%)T$
$S_{RB} \leq (0.66W + 42\%)T$	$S_{RB} \leq (0.66W + 45\%)T$

b. Single Recirculation Loop Operation

Trip Setpoint	Allowable Value
$S \leq (0.66W + 42.7\%)T$	$S \leq (0.66W + 45.7\%)T$
$S_{RB} \leq (0.66W + 36.7\%)T$	$S_{RB} \leq (0.66W + 39.7\%)T$

where: S and S_{RB} are in percent of RATED THERMAL POWER,
W = Loop recirculation flow as a percentage of the loop recirculation flow which produces a rated core flow of 84.5 million lbs/hr.
T = The ratio of FRACTION OF RATED THERMAL POWER (FRTP) divided by the CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD). T is applied only if less than or equal to 1.0.

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

ACTION:

With the APRM flow biased simulated thermal power-high scram trip setpoint and/or the flow biased neutron flux-upscale control rod block trip setpoint less conservative than the value shown in the Allowable Value column for S or S_{RB}, as above determined, initiate corrective action within 15 minutes and adjust S and/or S_{RB} to be consistent with the Trip Setpoint value * within 6 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

*With CMFLPD greater than the FRTP, rather than adjusting the APRM setpoints, the APRM gain may be adjusted such that the APRM readings are greater than or equal to 100% times CMFLPD, provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER, and a notice of the adjustment is posted on the reactor control panel.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRTP and CMFLPD shall be determined, the value of T calculated, and the most recent actual APRM flow biased simulated thermal power-high scram and flow biased neutron flux-upscale control rod block trip setpoints verified to be within the above limits or adjusted, as required:

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 CMFLPD greater than or equal to FRTP.

d. The provisions of Specification 4.0.4 are not applicable.

POWER DISTRIBUTION LIMITS

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 both $MCPR_f$ and $MCPR_p$ limits at indicated core flow and THERMAL POWER as shown in Figures 3.2.3-1 and 3.2.3-2.

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 limit shown in Figures 3.2.3-1 and 3.2.3-2, initiate corrective action within 15 minutes and restore MCPR to within the required limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.3 MCPR shall be determined to be equal to or greater than the MCPR limit determined from Figures 3.2.3-1 and 3.2.3-2:

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.

TABLE 3.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Declare the RPCS inoperable and take the ACTION required by Specification 3.1.4.2.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.#

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.#

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.#

NOTES

* With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

** OPERABLE channels must be associated with SRM required OPERABLE per Specification 3.9.2.

The provisions of Specification 3.0.4 are not applicable.

(a) This function shall be automatically bypassed if detector count rate is $\geq$ 100 cps or the IRM channels are on range 3 or higher.

(b) This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

(c) This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

(d) This function shall be automatically bypassed when the IRM channels are on range 1.

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD PATTERN CONTROL SYSTEM		
a. Low Power Setpoint	27.5 ± 3% of RATED THERMAL POWER	27.5 ± 7.5% of RATED THERMAL POWER
b. High Power Setpoint	62.5 ± 3% of RATED THERMAL POWER	62.5 ± 7.5% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux Upscale		
1) Two Recirculation Loop Operation	≤ 0.66W + 42%*	≤ 0.66W + 45%*
2) Single Recirculation Loop Operation	≤ 0.66W + 36.7%*	≤ 0.66W + 39.7%*
b. Inoperative	NA	NA
c. Downscale	≥5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	< 1×10^5 cps	< 1.6×10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 0.7 cps	≥ 0.5 cps**
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 division of full scale	≤ 110/125 division of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 division of full scale	≥ 3/125 division of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High - LISN602A	< 18.00"	< 21.12"
LISN602B	≤ 18.00"	≤ 21.60"
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ 108% of rated flow	≤ 111% of rated flow

*The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

**Provided signal to noise ratio is ≥ 2, otherwise setpoint of 3 cps and allowable 1.8 cps.

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 system recirculation loops shall be in operation and in Region I as specified in Figure 3.4.1.1-1 with either:

a. Two recirculation loops operating with limits and setpoints per Specifications 2.1.2, 2.2.1, 3.2.1, 3.2.2, 3.3.6, or

b. A single loop operating with:

   1. Volumetric recirculation loop flow rate less than or equal to 33,000 gpm, and

   2. The recirculation loop flow control system in the loop Manual (Position Control) Mode, and

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

   4. Limits and setpoints for single recirculation loop operation per Specifications 2.1.2, 2.2.1, 3.2.1, 3.2.2, and 3.3.6.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*

ACTION

a. During single loop operation, with volumetric recirculation loop flow rate greater than 33,000 gpm, immediately initiate corrective action to reduce flow to less than or equal to 33,000 gpm within 1 hour.

b. During single loop operation, with the recirculation flow control system not in the Loop Manual mode, immediately initiate corrective action to place the recirculation flow control system in the Loop Manual mode within 1 hour.

c. During single loop operation, with THERMAL POWER greater than 70% of RATED THERMAL POWER, immediately initiate corrective action to reduce THERMAL POWER to less than or equal to 70% of RATED THERMAL POWER within 1 hour.

d. Within 4 hours upon entry into single loop operation, verify that the operating limits in Specification 3.2.1 have been appropriately adjusted for single loop operation.

*See Special Exception 3.10.4

e. Within 12 hours upon entry into single loop operation, verify that the setpoints in Specifications 2.2.1, 3.2.2 and 3.3.6 are within appropriate limits.

f. During single loop operation with either THERMAL POWER ≤ 30% of RATED THERMAL POWER or recirculation loop flow in the operating loop is ≤ 50% of rated recirculation loop flow and temperature differences exceeding the limits in Surveillance Requirement 4.4.1.1.4, suspend THERMAL POWER or recirculation loop flow increases.*

g. With one or two reactor coolant system recirculation loops in operation and total core flow greater than 39% and less than 45% of rated core flow and THERMAL POWER greater than the limit specified in Region II of Figure 3.4.1.1-1:

   1. Determine the APRM and LPRM** noise levels (Surveillance 4.4.1.1.2):

      a) At least once per 8 hours, and

      b) Within 30 minutes after completion of a THERMAL POWER increase of at least 5% of RATED THERMAL POWER.

   2. With the APRM or LPRM** neutron flux noise levels greater than three times their established baseline noise levels, immediately initiate corrective action to restore the noise levels within the required limits within 2 hours by increasing core flow to greater than or equal to 45% of rated core flow or by reducing THERMAL POWER to less than or equal to the limit specified in Region II of Figure 3.4.1.1-1.

h. With one or two reactor coolant system recirculation loops in operation and total core flow less than 39% of rated core flow and THERMAL POWER greater than the limit specified in Region III of Figure 3.4.1.1-1, immediately within 15 minutes initiate corrective action to increase core flow to greater than or equal to 39% of rated core flow or reduce THERMAL POWER to less than the limit specified in Region III of Figure 3.4.1.1-1 within 4 hours.

---

*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 provisions of Specification 3.0.4 are not applicable with respect to Surveillance Requirement 4.4.1.1.4b and c.

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

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 c) 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

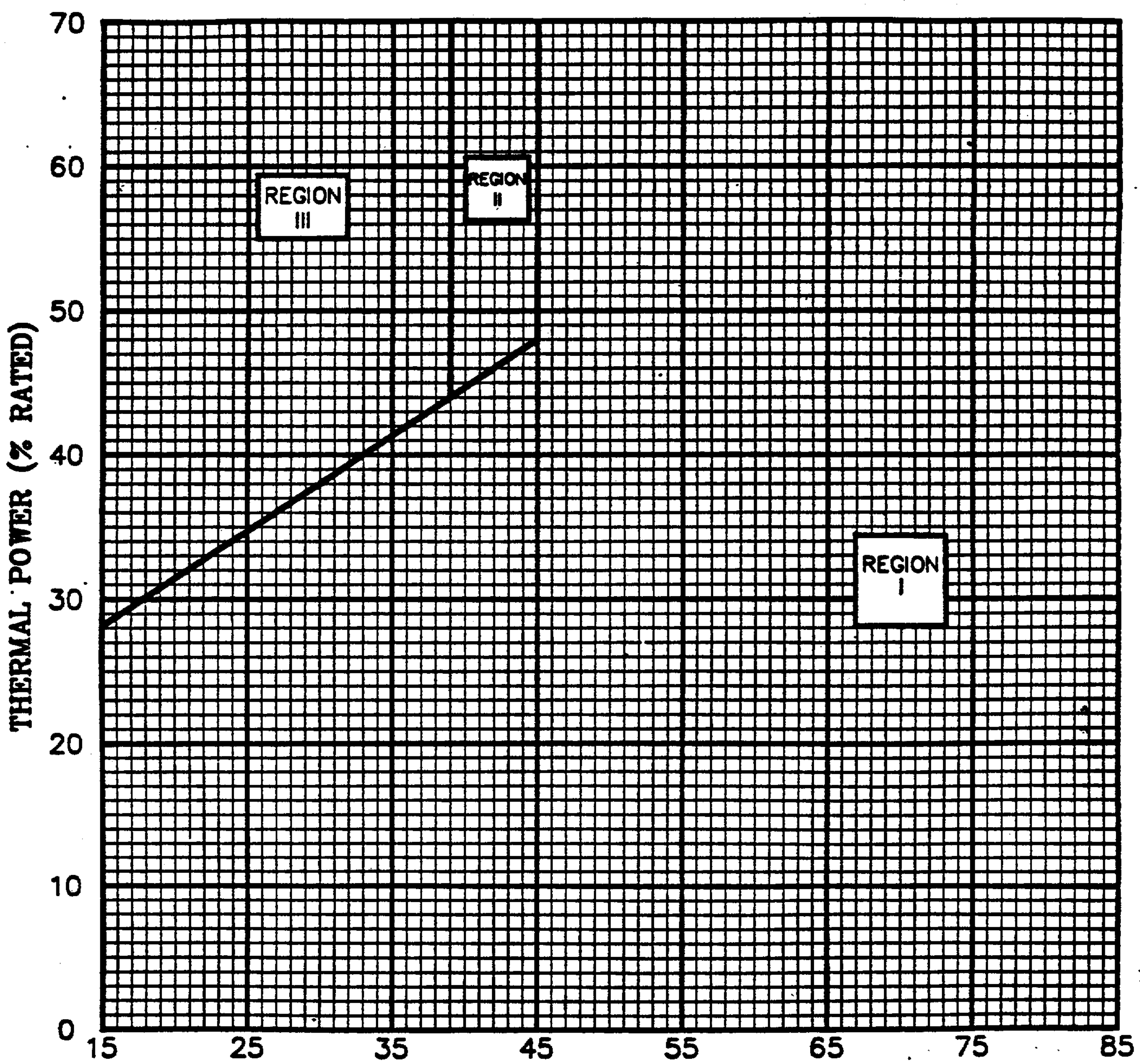
*Detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center of core should be monitored.

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.



CORE FLOW (% RATED)
FIGURE 3.4.1.1-1
THERMAL POWER VERSUS CORE FLOW

REACTOR COOLANT SYSTEM

JET PUMPS

LIMITING CONDITION FOR OPERATION

3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 During two recirculation loop operation each of the above required jet pumps shall be demonstrated OPERABLE prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER, and at least once per 24 hours while greater than 25% of RATED THERMAL POWER, by determining recirculation loop flow, total core flow and diffuser-to-lower plenum differential pressure for each jet pump and verifying that no two of the following conditions occur when both recirculation loop indicated flows are in compliance with Specification 3.4.1.3.

a. The indicated recirculation loop flow differs by more than 10% from the established flow control valve position-loop flow characteristics.

b. The indicated total core flow differs by more than 10% from the established total core flow value derived from recirculation loop flow measurements.

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established patterns by more than 10%.

d. The provisions of Specification 4.0.4 are not applicable provided that this surveillance is performed within 24 hours after exceeding 25% of RATED THERMAL POWER.

4.4.1.2.2 During single recirculation loop operation, each of the required jet pumps in the operating recirculation loop shall be demonstrated OPERABLE at least once per 24 hours while greater than 25% of RATED THERMAL POWER, by determining recirculation loop flow in the operating loop, total core flow and diffuser-to-lower plenum differential pressure for each jet pump in the operating loop and verifying that no two of the following conditions occur:

a. The indicated recirculation loop flow in the operating loop differs by more than 10% from the established* single recirculation flow control valve positions - loop flow characteristics.

*To be determined during initial use of single loop operation. Surveillance Requirements of 4.4.1.2 are not required to allow determination of characteristic curves.

SURVEILLANCE REQUIREMENTS (Continued)

b. The indicated total core flow differs by more than 10% from the established* jet pump flow/recirculation pump flow characteristic for the operating loop.

c. The individual diffuser-to-lower plenum differential pressure of any individual jet pump differs from established* single recirculation loop patterns by more than 10%.

d. The provisions of specification 4.0.4 are not applicable provided that this surveillance is performed within 24 hours after exceeding 25% of RATED THERMAL POWER.

*To be determined during initial use of single loop operation. Surveillance Requirements of 4.4.1.2 are not required to allow determination of characteristic curves.

RECIRCULATION LOOP FLOW

LIMITING CONDITION FOR OPERATION

3.4.1.3 Recirculation loop flow mismatch shall be maintained within:

a. 5% of rated recirculation flow with core flow greater than or equal to 70% of rated core flow.

b. 10% of rated recirculation flow with core flow less than 70% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2* during two recirculation loop operation.

ACTION:

With recirculation loop flows different by more than the specified limits, either:

a. Restore the recirculation loop flows to within the specified limit within 2 hours, or

b. Shutdown one of the recirculation loops and take the ACTION required by Specification 3.4.1.1.**

SURVEILLANCE REQUIREMENTS

4.4.1.3 Recirculation loop flow mismatch shall be verified to be within the limits at least once per 24 hours.

*See Special Test Exception 3.10.4.
**The provisions of Specification 3.0.4 are not Applicable.

REACTOR COOLANT SYSTEM

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

3.4.1.4 An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is less than or equal to 100°F,* and:

a. When both loops have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is less than or equal to 50°F, or

b. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is less than or equal to 50°F and the operating loop flow rate is less than or equal to 50% of rated loop flow.

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

ACTION:

With temperature differences and/or flow rates exceeding the above limits, suspend startup of any idle recirculation loop.

SURVEILLANCE REQUIREMENTS

4.4.1.4 The temperature differentials and flow rate shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop.

*Below 25 psig, this temperature differential is not applicable.

3/4.1.3 CONTROL RODS

The specifications of this section (1) ensure that the minimum SHUTDOWN MARGIN is maintained and the control rod insertion times are consistent with those used in the safety analyses, and (2) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem. Therefore, with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period that is long enough to permit determining the cause of the inoperability yet prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those not fully inserted are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shut down for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than the fuel cladding safety limit during the limiting power transient analyzed in Section 15.0 of the FSAR. This analysis shows that the negative reactivity rates, resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than the fuel cladding safety limit. The occurrence of scram times longer then those specified should be viewed as an indication of a systemic problem with the rod drives and, therefore, the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

3/4.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold, xenon-free condition and shall show the core to be subcritical by at least R + 0.38% delta k/k or R + 0.28% delta k/k, as appropriate. The value of R in units of % delta k/k is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of R must be positive or zero and must be determined for each fuel loading cycle.

Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of demonstration of the SHUTDOWN MARGIN. The highest worth rod may be determined analytically or by test. The SHUTDOWN MARGIN is demonstrated by an in-sequence control rod withdrawal at the beginning-of-life fuel cycle conditions and, if necessary, at any future time in the cycle if the first demonstration indicates that the required margin could be reduced as a function of exposure. Observation of subcriticality in this condition assures subcriticality with the most reactive control rod fully withdrawn.

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined any time a control rod is incapable of insertion.

3/4.1.2 REACTIVITY ANOMALIES

Since the SHUTDOWN MARGIN requirement for the reactor is small, a careful comparison of actual conditions to the predicted conditions is necessary, and the changes in reactivity can be inferred from these comparisons of rod patterns. Since the comparisons are easily done, frequent checks are not an imposition on normal operations. A 1% change is larger than is expected for normal operation so a change of this magnitude should be thoroughly evaluated. A change as large as 1% would not exceed the design conditions of the reactor and is on the safe side of the postulated transients.

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

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 Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5 and 3.2.1-6.

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 on Figures 3.2.1-1, 3.2.1-2, 3.2.1-3, 3.2.1-4, 3.2.1-5 and 3.2.1-6 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 3.2.1-6 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 $\geq$ 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.

Bases Table B 3.2.1-1

SIGNIFICANT INPUT PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters;

Core THERMAL POWER .... 3015 Mwt* which corresponds to 105% of rated steam flow

Vessel Steam Output .... 13.08×10^6 lbm/hr which corresponds to 105% of rated steam flow

Vessel Steam Dome Pressure.............. 1060 psia

Design Basis Recirculation Line Break Area for:

a. Large Breaks 2.2 ft^2.

b. Small Breaks 0.09 ft^2.

Fuel Parameters:

FUEL TYPE	FUEL ASSEMBLY GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kw/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core	8 x 8	13.4	1.4	1.17**

A more detailed listing of input of each model and its source is presented in Section II of NEDE 20566[1] and subsection 6.3.3 of the FSAR.

*This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes an assembly power consistent with operation of the highest powered rod at 102% of its Technical Specification LINEAR HEAT GENERATION RATE limit.

**For single recirculation loop operation, loss of nucleate boiling is assumed at 0.01 after LOCA regardless of initial MCPR.

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 of 1.07 and an analysis of abnormal operational transients. For any abnormal operating transient analysis, 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 settings 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 MCPR. When added to the Safety Limit MCPR of 1.07, the required minimum operating limit MCPR of Specification 3.2.3 is obtained and is presented in Figure 3.2.3-1. Analysis of transients occurring during single recirculation loop operation indicates that the maximum operating limit MCPR will be bounded by the limits in Specification 3.2.3. The power-flow map of Figure B 3/4 2.3-1 shows typical regions of plant operation.

The evaluation of a given transient begins with the system initial parameters identified in Reference 2 that are input to a GE core dynamic behavior transient computer program. The codes used to evaluate transients are described in Reference 2. The principal result of this evaluation is the reduction in MCPR caused by transient.

The purpose of the $MCPR_f$ and $MCPR_p$ of Figures 3.2.3-1 and 3.2.3-2 is to define operating limits at other than rated core flow and power conditions. At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_f$ and $MCPR_p$ at the existing core flow and power state. 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 $MCPR_f$s were calculated such that, for the maximum core flow rate and the corresponding THERMAL POWER along the 105%-of-rated steam flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along the 105%-of-rated steam flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow is defined as $MCPR_f$.

3/4.4.1 RECIRCULATION SYSTEM

The impact of single recirculation loop operation upon plant safety has been assessed and single recirculation loop operation is permitted if the MCPR fuel cladding safety limit is increased as noted by Specification 2.1.2; APRM scram and control rod block setpoints are adjusted as noted in Tables 2.2.1-1 and 3.3.6-2, respectively; MAPLHGR limits are decreased by the factor given in Specification 3.2.1 (Reference 3). MCPR operating limits are adjusted per specification 3/4.2.3, for both single and two recirculation loop operation.

Additionally, surveillances on the volumetric flow rate of the operating recirculation loop is imposed to exclude the possibility of excessive core internal vibration. The surveillance on differential temperatures below 30% THERMAL POWER or 50% rated recirculation loop flow is to mitigate the undue thermal stress on the vessel nozzles, recirculation pump and vessel bottom head during extended operation in the single recirculation loop mode.

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 single loop operation the jet pump operability surveillances are only performed for the jet pumps in the operating recirculation loop, as the loads on the inactive jet pumps are expected to be very low due to the low flow in the reverse direction through the jet pumps. Recirculation loop flow mismatch limits are in compliance with ECCS LOCA analysis design criteria for two recirculation loop operation. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA. In the case where the mismatch limits cannot be maintained during two recirculation loop operation, continued operation is permitted in a single recirculation loop operation mode.

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. Sudden equalization of a temperature difference >100°F between the reactor vessel bottom head coolant and the coolant in the upper region of the reactor vessel by increasing core flow rate would cause undue stress in the reactor vessel bottom head.

The objective of GE BWR plant and fuel design is to provide stable operation with margin over the normal operating domain. However, at the high power/low flow corner of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., rod pattern, power shape). To provide assurance that neutron flux limit cycle oscillations are detected and suppressed, APRM and LPRM neutron flux noise levels should be monitored while operating in this region.

RECIRCULATION SYSTEM (Continued)

Stability tests at operating BWRs were reviewed to determine a generic region of the power/flow map in which surveillance of neutron flux noise levels should be performed. A conservative decay ratio of 0.6 was chosen, as the basis for determining the generic region for surveillance, to account for the plant-to-plant variability of decay ratio with core and fuel designs. This generic region has been determined to correspond to a core flow of less than or equal to 45% of rated core flow and a thermal power greater than that specified in Figure 3.4.1.1-1 (Reference 1).

Plant-specific calculations can be performed to determine an applicable region for monitoring neutron flux noise levels. In this case the degree of conservatism can be reduced since plant-to-plant variability would be eliminated. In this case, adequate margin will be assured by monitoring the region which has a decay ratio greater than or equal to 0.8.

3/4.4 REACTOR COOLANT SYSTEM

BASES

RECIRCULATION SYSTEM (Continued)

Neutron flux noise limits are also established to ensure early detection of limit cycle neutron flux oscillations. BWR cores typically operate with neutron flux noise caused by random boiling and flow noise. Typical neutron flux noise levels of 1 to 12% of rated power (peak-to-peak) have been reported for the range of low to high recirculation loop flow during both single and dual recirculation loop operation. Neutron flux noise levels which significantly bound these values are considered in the thermal/mechanical design of GE BWR fuel and are found to be of negligible consequence (Reference 2). In addition, stability tests at operating BWRs have demonstrated that when stability related neutron flux limit cycle oscillations occur they result in peak-to-peak neutron flux limit cycles of 5 to 10 times the typical values. Therefore, actions taken to reduce neutron flux noise levels exceeding three times the typical value are sufficient to ensure early detection of limit cycle neutron flux oscillations.

Typically, neutron flux noise levels show a gradual increase in absolute magnitude as core flow is increased (constant control rod pattern) with two reactor recirculation loops in operation. Therefore, the baseline neutron flux noise level obtained at a specific core flow can be applied over a range of core flows. To maintain a reasonable variation between the low flow and high flow ends of the flow range, the range over which a specific baseline is applied should not exceed 20% of rated core flow with two recirculation loops in operation. Data from tests and operating plants indicate that a range of 20% of rated core flow will result in approximately a 50% increase in neutron flux noise level during operation with two recirculation loops. Baseline data should be taken near the maximum rod line at which the majority of operation will occur. However, baseline data taken at lower rod lines (i.e., lower power) will result in a conservative value since the neutron flux noise level is proportional to the power level at a given core flow.

References

(1) "BWR Core Thermal-Hydraulic Stability," Service Information Letter 380, Revision 1, February 1984.

(2) G. A. Watford, "Compliance of the General Electric Boiling Water Reactor Fuel Designs to Stability Licensing Criteria," December 1982 (NEDE 22277-P).

(3) "Single-Loop Operation Analysis for River Bend Station, Unit 1," NEDO-31441, May 1987.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves (SRV) is to prevent the reactor coolant system from being pressurized above the Safety Limit of 1375 psig, in accordance with the ASME Code. A total of 9 OPERABLE safety-relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient. Any combination of 4 SRVs operating in the relief mode and 5 SRVs operating in the safety mode is acceptable.

SAFETY/RELIEF VALVES (Continued)

Demonstration of the safety-relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

The low-low set system ensures that safety/relief valve discharges are minimized for a second opening of these valves, following any overpressure transient. This is achieved by automatically lowering the closing setpoint of 5 valves and lowering the opening setpoint of 2 valves following the initial opening. In this way, the frequency and magnitude of the containment blowdown duty cycle is substantially reduced. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973. In conformance with Regulatory Guide 1.45, the atmospheric gaseous radioactivity system will have a sensitivity of 10^{-6} μCi/cc.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage, due to equipment design and the detection capability of the instrumentation for determining system leakage, was also considered. The evidence obtained from experiments suggests that, for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shut down to allow further investigation and corrective action.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity, thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 31 TO FACILITY OPERATING LICENSE NO. NPF-47

GULF STATES UTILITIES COMPANY

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

ENCLOSURE 1

1.0 INTRODUCTION

By letter dated April 6, 1988, as supplemented October 20, 1988, Gulf States Utilities Company (GSU) (the licensee) requested an amendment to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1. The proposed amendment would modify the Technical Specifications to allow single recirculation loop operation.

Single recirculation loop operation (SLO) at reduced power is highly desirable when one loop becomes inoperative during maintenance or testing activities. This evaluation provides the results of the NRC staff's review of the licensee's evaluation of accidents and abnormal operational transients with only one recirculation pump operative. This evaluation is performed for a P8X8R fueled core on an equilibrium cycle basis up to a maximum power of approximately 70% of rated. The analysis and evaluation are applicable to both the initial fuel cycle and reload cycles.

Reference 1 also addresses Technical Specification changes related to Thermal-Hydraulic Stability considerations during SLO. The staff has reviewed the proposed changes and included an evaluation in Section 2 of this Safety Evaluation.

This evaluation also addresses the proposed recirculation flow and differential temperature limits to avoid thermal stratification that could result in unacceptable thermal stress levels in the bottom head region during SLO.

2.0 EVALUATION

The licensee provided a General Electric (GE) report entitled "Single Loop Operation Analysis for River Bend Station, Unit 1" (Ref. 2). The GE report evaluated the SLO safety issues pertaining to the River Bend Station to justify extended operation with one recirculation loop out of service. The staff evaluation of the SLO safety issues and the proposed Technical Specification changes follows.

2.1 MCPR FUEL CLADDING INTEGRITY SAFETY LIMIT

The net effect of increased uncertainties in the core total flow and Traversing In-Core Probe (TIP) readings for the single loop operation is a 0.01 incremental increase in the minimum critical power ratio (MCPR) fuel cladding integrity safety limit. Operating with one recirculation loop results in a maximum power output which is about 30% below that which is attainable for two-pump operation. Therefore, consequences of abnormal operation transients from one-loop operation will be less severe than those from a full power two-loop operational mode as provided in the River Bend Updated Safety Analysis Report (USAR).

The transient peak value results and Critical Power Ratio (CPR) results for the Load Rejection with Bypass Failure (LRBPF) and Feedwater Controller Failure (FWCF) with maximum demand are summarized in Table 1.

TABLE 1

SUMMARY OF TRANSIENT PEAK VALUE AND CPR RESULTS

	LRBPF	FWCF
Initial Power/Flow (% Rated)	70.2/53.6	70.2/53.6
Peak Neutron Flux (% NBR)	70.3	84.4
Peak Heat Flux (% Initial)	100.3	107.4
Peak Dome Pressure (psig)	1169	1153
Peak Vessel Bottom Pressure (psig)	1182	1165
Required Two Loop Initial MCPR		
Operating Limit at SLO Condition	1.39	1.39
delta-CPR	0.05	0.12
Transient MCPR	1.34	1.27
SLMCPR at SLO	1.07	1.07

This table shows that for the limiting transient events analyzed here, the MCPRs are all above the single-loop operation safety limit value of 1.07 so that there will be no fuel failure due to boiling transition. The peak vessel pressures are all below the ASME code value of 1375 psig. Therefore, the pressure barrier integrity is maintained under single-loop operation conditions. The staff finds this acceptable.

2.2 MCPR Operating Limit

2.2.1 Accidents (Other Than LOCA) and Transients Affected by One Recirculation Loop Out of Service

One Pump Seizure Accident

A plant specific analysis was not performed for this event. Previous analyses for the Grand Gulf plant has shown that the event results in a MCPR value significantly above the SLO safety limit MCPR. This has been confirmed for other BWRs (Refs. 3 and 4).

2.2.2 Abnormal Operating Transients

Although the increased uncertainties in core total flow and TIP readings resulted in a 0.01 increase in MCPR fuel cladding integrity safety limit during single-loop operation, the limiting transients analyzed in the GE report indicate that there is more than enough MCPR margin during single-loop operation to compensate for this increase in safety limit. For single-loop operation at off-rated conditions, the steady state operating MCPR limit is established by the power dependent and flow dependent MCPR curves. For the most limiting transient events analyzed, the GE report also shows that the present power dependent MCPR limits are bounding for single-loop operation. Further, the present flow dependent MCPR limits are also bounding for single-loop operation since the maximum core flow runout during single loop operation is only about 54% of rated. The transient consequence from one-loop operation is therefore bounded by previously submitted full power analyses. This is acceptable.

2.2.3 Rod Withdrawal Error

The rod withdrawal error at rated power is given in the USAR for the initial core and in cycle-dependent reload supplemental submittals. These analyses were performed to demonstrate that, even if the operator had ignored all instrument indications and alarms during the course of the transient, the rod block system would stop rod withdrawal at a minimum critical power ratio which is higher than the fuel cladding integrity safety limit. The GE report also shows that correction of the rod block equation for single-loop operation assures that the MCPR safety limit is not violated.

One-pump operation results in backflow through 10 of the 20 jet pumps while flow is being supplied to the lower plenum from the 10 active jet pumps. Because of this backflow through the inactive jet pumps, the present rod block equation and APRM settings were modified for use during one-pump operation. The staff has found them acceptable.

The staff finds that one-loop transients and accidents other than LOCA, which is discussed below, are bounded by the two-loop operation analyses and are, therefore, acceptable.

2.3 Stability Analysis

With one recirculation loop not in service, the primary contributing factors to the stability performance are the power/flow ratio and the recirculation loop characteristics. At forced circulation with one recirculation loop not in operation, the reactor core stability is influenced by the inactive recirculation loop. Staff evaluations have considered whether increased noise in SLO was being caused by reduced stability margin as SLO core flow was increased. Results of analyses and test indicates that the SLO stability characteristics are not significantly different from two-loop operation. At low core flows, SLO may be slightly less stable than two-loop operation but as core flow is increased and reverse flow is established, the stability performance is similar. At higher core flows with substantial reverse flow in the inactive recirculation loop, the effect of cross flow on the flow noise results in an increase in system noise (jet pump, core flow and neutron flux noise), but core thermal-hydraulic stability margin is very high, similar to two-loop operation. GE has developed a Service Information Letter-380, Revision 1 (Reference 5) informing plant operators how to recognize and suppress unanticipated oscillations when encountered during plant operation.

The NRC has approved the recommendation of SIL-380 for incorporation into BWR Plant Technical Specifications. The licensee has incorporated the surveillance requirements recommended by SIL-380 into the River Bend Technical Specifications and has proposed modifications applicable to the SLO mode. The staff finds this acceptable.

In a related matter, the NRC has identified generic safety implications regarding power oscillations in Boiling Water Reactors and has recently issued an NRC Bulletin No. 88-07 (Ref. 7) dealing with this subject. The licensee for River Bend has responded to the Bulletin by Reference 8 and has identified a revision to a station Abnormal Operating Procedure (AOP) in addition to confirmation of the bulletin action items. The NRC herein acknowledges the licensee's response and notes that the AOP revision will be reviewed under an NRC Regional Office inspection in accordance with a Temporary Instruction procedure.

2.4 Loss of Coolant Accident Analysis

SAFE/REFLOOD calculations were performed for a full spectrum of large break sizes for the recirculation suction line breaks for the single-loop operation mode. The small differences in uncovery time and reflood time for the limiting break size, i.e., 183 seconds for the single-loop vs. 184 seconds for the two-loop operation, would result in a small change in the calculated peak cladding temperature. The maximum average planar linear heat generation rate (MAPLHGR) reduction factor for the most limiting single-loop operation for P8X8R fuel is 0.84 which is conservative.

In the event of a small break LOCA, the slight increase (50°F) in peak clad temperature (PCT) is offset by the effect of the decreased MAPLHGR (equivalent to 300°F to 500°F PCT) for the single-loop operation. The calculated PCT values for small breaks will therefore be well below the 1547°F PCT value previously analyzed for small breaks. The LOCA analyses applicable to the River Bend SLO mode have been performed using methodology approved by the staff (Ref. 9) and the results are acceptable.

2.5 Containment Analysis

The GE analysis indicates that under SLO conditions limiting case accidents would result in peak containment pressures, containment temperatures, and suppression pool temperatures which are less severe than those estimated for design basis accidents under two-loop operation. GE also evaluated the chugging, condensation oscillation and pool swell loads under SLO conditions and stated that these loads slightly exceed those estimated for accidents during two-loop operation. In response to a staff request, the licensee provided additional discussion and quantification of the containment loads for SLO conditions. The information provided in Reference 10 verified that the evaluation specified in Appendix 6A of the River Bend Updated Safety Analysis Report (USAR) is applicable to SLO at the limiting operating point. The analysis was made using staff-approved methodology and a model based on the MARK III containment test program. The margins for pool swell loads and condensation loads were a few percent greater than two-loop limiting conditions and well within the design values. Chugging loads do not increase for SLO conditions. The staff finds this acceptable.

2.6 Miscellaneous Impact Evaluation

° Anticipated Transient Without Scram

Since the SLO initial power/flow condition is less than the rated condition used for the two-loop ATWS analysis, GE found the transient response less severe and therefore bounded by the FSAR analyses. This is acceptable.

° Fuel Mechanical Performance

Due to the substantial reverse flow established during SLO, both the Average Power Range Monitor (APRM) noise and core plate differential pressure noise are increased slightly. GE has stated that the APRM fluctuation should not exceed the fuel rod and assembly design bases. This is acceptable.

° Vessel Internal Vibration

GE imposed a recirculation pump drive flow limit for single-loop operation, which is about 33,000 gpm for rated reactor water temperature and pressure. This is based on measured prototypical value from the Kuo Sheng 1 plant which has been accepted by the staff as the valid prototype for River Bend. With maximum flow thus limited, vibration levels of the reactor internal components will be within acceptance limits during SLO at River Bend Station. This is acceptable.

° Jet Pump Operability

Jet pump surveillance is only required for the operating loop. The licensee has proposed modifications to the River Bend TS to accommodate the SLO mode. These changes are acceptable to the staff.

2.7 Thermal Stress Limits

The licensee considered the possibility that thermal stratification may occur in the bottom head of the reactor pressure vessel during single loop operation. Thermal stratification may occur if a stagnant layer of cold water forms near the bottom head. If the water suddenly mixes with warm water such that the temperature in the bottom head suddenly increases, then penetrations in the bottom head may expand at a rate different from the bottom head. This may result in the formation of cracks at the penetrations.

To avoid single loop operation at low power or low flow conditions that may allow thermal stratification to occur, the licensee has proposed limits of operation at greater than 30% rated thermal power and greater than 50% rated recirculation loop flow in the operating loop in order to increase power or flow. The licensee has stated that operation in the region of the power-flow map above these limits would not lead to thermal stratification. However, operation at or below the 30% rated thermal power or at or below the 50% rated recirculation loop flow would be permitted if the following differential temperature requirements are met within 15 minutes prior to an increase in thermal power or increase in recirculation loop flow:

a. Less than or equal to 100°F between the reactor vessel steam space coolant and bottom head drain line coolant; and

b. Less than or equal to 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel (not applicable if the loop is isolated); and

c. Less than or equal to 50°F between the reactor coolant within the loop not in operation and the operating loop (not applicable if the loop is isolated).

The licensee has proposed that these limits be incorporated in Technical Specification 3.4.1.1, action statement f, Surveillance Requirement 4.4.1.1.4, and Bases 3/4.4.1.

Current Technical Specification 3.4.1.4 contains similar temperature differential restrictions to prevent undue stress on the reactor vessel with regard to idle recirculation loop startup. This technical specification states that an idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel steam space and the bottom head drain line coolant is less than or equal to 100°F, and:

a. When both loops have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is less than or equal to 50°F, or

b. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is less than or equal to 50°F and the operating loop flow rate is less than or equal to 50% rated loop flow.

Based on its review, it is the staff's judgement that for greater than 50% rated recirculation loop flow, and greater than 30% rated thermal power, there will be adequate circulation of reactor coolant during singleloop operation to assure that there will not be thermal stratification that could lead to unacceptable stresses in the bottom head of the pressure vessel. The staff also finds that the proposed differential temperature limits for operation at or below 50% rated recirculation loop flow, or at or below 30% rated thermal power are consistent with the current Technical Specification 3.4.1.4.

The staff concludes that proposed Technical Specifications 3.4.1.1, action f, Surveillance Requirement 4.4.1.4, and Bases 3/4.4.1 are acceptable.

3.0 TECHNICAL SPECIFICATION CHANGES

The licensee has proposed to change TS Limiting Conditions of Operation, Bases Sections and the corresponding descriptive sections thereto so as to be in conformance with the GE analyses to implement SLO. The staff has reviewed the changes and finds them consistent with results of the GE analysis and also with TS changes approved for other BWR/6 facilities for single-loop operation. The staff concludes that these TS changes are acceptable.

3.1 Specification 2.1.2, page 2-1

The safety limit Minimum Critical Power Ratio will be increased by 0.01 to 1.08 for single-loop operation. This number is to account for core flow and TIP reading uncertainties, which are used in the statistical analysis of the safety limit.

3.2 Table 2.2.2-1, page 2-4

The APRM Reactor Protection System Instrumentation Trip Setpoints are modified to account for backflow through half the jet pumps. The setpoint equations will be changed in the RBS Technical Specifications. The changes are similar to other plant TS and are acceptable to the staff.

3.3 Specification 3/4.2.1, page 3/4 2-1

The Limiting Condition for Operation was changed to reflect the 84 percent reduction in APLHGR values for single-loop operation. The number is derived from LOCA analyses initiated from single-loop operation as discussed in Section 2.4 of this Safety Evaluation.

3.4 Specification 3/4.2.2, page 3/4 2-7

The setpoint equations for the APRM setpoint changes in Table 2.2.1-1 are identified.

3.5 Table 3.3.6-1, page 3/4 3-62

Control Rod Block Instrumentation Setpoints will be modified to account for back flow through the inactive jet pumps. These changes are similar to previously approved SLO Technical Specification changes on other plants and are acceptable to the staff.

3.6 Specification 3/4.4.1, pages 3/4 4-1 through 3/4 4-5

The Technical Specifications related to the Recirculation Loops are modified to reflect single-loop operation considerations discussed in this Safety Evaluation. This includes replacement of Figure 3.4.1.1-1 to identify the detect and suppress regions of the power-flow map associated with thermal-hydraulic stability.

3.7 Bases Section Changes

The Bases Section changes related to the proposed SLO mode and identified in the licensee's submittal were reviewed by the staff for consistency with the changes discussed above. The staff finds the bases discussions accurately reflect the bases for the changes and are acceptable as proposed.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51.35, an environmental assessment and finding of no significant impact was published in the Federal Register on November 17, 1988 (53 FR 46516).

Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

5.0 CONCLUSION

The staff 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, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public. The staff therefore concludes that the proposed changes are acceptable and they are hereby incorporated into the River Bend Unit 1 Technical Specifications.

Dated: November 22, 1988

Principal Contributors: M. McCoy, W. Paulson

5.0 REFERENCES

1. Letter, J. C. Deddens (GSU) to Document Control Branch (NRC), "Application for Amendment to Operating License No. NPF-47," April 6, 1988.

2. "Single Loop Operation Analysis for River Bend Station, Unit 1," NEDO-31441 (DRF No. A00-02463), General Electric Company, May 1987.

3. Letter, D. M. Musolf (NSP) to Director of ONRR (NRC), "Request for Amendment to Operating License No. DPR-22," March 24, 1986.

4. T. L. Riley (Clinton Power Station), "Pump Seizure During Single Loop Operation," L30-86(12-15)-6, December 15, 1986.

5. "BWR Core Thermal Hydraulic Stability," General Electric Company, February 10, 1984 (Service Information Letter-380, Revision 1).

6. Letter, C. O. Thomas (NRC) to H. C. Pfefferlen (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011, Rev. 6, Amendment 8, Thermal Hydraulic Stability Amendment to GESTAR II," April 24, 1985.

7. NRC Bulletin No. 88-07: Power Oscillations in Boiling Water Reactors (BWRs), June 15, 1988.

8. Letter, J. E. Booker (GSU) to Document Control Desk (NRC), Response to NRC Bulletin 88-07, September 8, 1988.

9. Letter, H.N. Berkow (NRC) to J.F. Quirk (GE), Safety Evaluation of General Electric ECCS Evaluation Methodology for Single Loop Operation, dated March 5, 1986.

10. Letter (RBG-29071), J. E. Booker (GSU) to Document Control Desk (NRC) dated October 20, 1988.

UNITED STATES NUCLEAR REGULATORY COMMISSION

GULF STATES UTILITIES COMPANY

DOCKET NO. 50-458

NOTICE OF ISSUANCE OF AMENDMENT

TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 31 to Facility Operating License No. NPF-47, issued to Gulf States Utilities Company, (the licensee), which revised the Technical Specifications for operation of the River Bend Station, Unit 1, located in West Feliciana Parish, Louisiana.

The amendment was effective as of the date of its issuance.

The amendment revised the Technical Specifications to allow single recirculation loop operation.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter 1, which are set forth in the license amendment.

The Notice of Consideration of Issuance of Amendment was published in the Federal Register on May 13, 1988 (53 FR 17131). No request for a hearing or petition for leave to intervene was filed following the notices.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment.

For further details with respect to the action, see: (1) the application for amendment dated April 6, 1988, as supplemented October 20, 1988; (2) Amendment No. 31 Facility Operating License No. NPF-47; and (3) the Commission's related Safety Evaluation and Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, 2120 L Street N. W., Washington, D.C. 20555; at Government Documents Department, Louisiana State University, Baton Rouge, Louisiana 70803. A copy of items (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects-III, IV, V and Special Projects.

Dated at Rockville, Maryland this 22nd day November, 1988.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter A. Paulson

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