May 5, [illegible]9

Mr. Randall K. Edington
Vice President - Operations
Entergy Operations, Inc.
River Bend Station
P. O. Box 220
St. Francisville, LA 70775

SUBJECT: RIVER BEND STATION, UNIT 1 - ISSUANCE OF AMENDMENT RE: REACTOR STABILITY LONG-TERM SOLUTION (TAC NO. MA3886)

Dear Mr. Edington:

The Commission has issued the enclosed Amendment No. 106 to Facility Operating License No. NPF-47 for the River Bend Station, Unit 1 (RBS). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated October 8, 1998, as supplemented by letter dated April 15, 1999.

The amendment implements the Boiling Water Reactor Owners Group (BWROG) Enhanced Option I-A (EIA) for the reactor stability long-term solution to the neutronic and thermal hydraulic instability that is documented in NEDO-32339, Revision 1, "Reactor Stability Long-Term Solution, Enhanced Option I-A." The amendment deletes the limits on power and flow conditions in the specification on recirculating loops operating, adds two new specifications to the TSs to establish limits for fraction of core boiling boundary and the period based detection system, modifies the surveillance requirements for reactor protection instrumentation, and adds the NEDO document to the core operating limits report. The associated Bases pages are also revised and are included with the new TSs.

As stated in the enclosed Safety Evaluation Report, the staff would like to arrange an on-site review of the plant-specific installation of the BWROG reactor stability long-term solution EIA at RBS in the near future.

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

Sincerely,

ORIG. SIGNED BY
Robert J. Fretz, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-458
Enclosures: 1. Amendment No.106 to NPF-47
2. Safety Evaluation
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 5, 1999

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Vice President - Operations
Entergy Operations, Inc.
River Bend Station
P. O. Box 220
St. Francisville, LA 70775

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Sincerely,

Robert J. Fretz, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosures: 1. Amendment No. 106 to NPF-47
2. Safety Evaluation

cc w/encls: See next page

Mr. Randall K. Edington
Entergy Operations, Inc.

River Bend Station

cc:

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

ENTERGY GULF STATES, INC. **

AND

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

RIVER BEND STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 106
License No. NPF-47

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

   A. The application for amendment by Entergy Gulf States, Inc.* (the licensee) dated October 8, 1998, as supplemented by letter dated April 15, 1999, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, 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

---

* EOI is authorized to act as agent for Entergy Gulf States, Inc, and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

**Entergy Gulf States, Inc., has merged with a wholly owned subsidiary of Entergy Corporation. Entergy Gulf States, Inc. was the surviving company in the merger.

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. 106 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. EOI 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 and shall be implemented during refueling outage 8.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Gramm

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

Attachment: Changes to the Technical Specifications

Date of Issuance: May 5, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 106

FACILITY OPERATING LICENSE NO. NPF-47

DOCKET NO. 50-458

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

Remove	Insert
---	3.2-4
---	3.2-5
---	3.2-6
3.3-3	3.3-3
3.3-4	3.3-4
3.3-5	3.3-5
3.3-6	3.3-6
3.3-7	3.3-7
---	3.3-14a through 3.3-14c
3.4-1	3.4-1
3.4-2	3.4-2
3.4-3	3.4-3
5.0-18	5.0-18
5.0-19	5.0-19
---	B 3.2-12 through B 3.2-18
B 3.3-8 through B 3.3-10	B 3.3-8 through B 3.3-10
B 3.3-25	B 3.3-25
---	B 3.3-25a
B 3.3-26	B 3.3-26
B 3.3-29 through B 3.3-31	B 3.3-29 through B 3.3-31a
---	B 3.3-39a through B 3.3-39h
B 3.4-1 through B 3.4-8	B 3.4-1 through B 3.4-8

3.2 POWER DISTRIBUTION LIMITS

3.2.4 Fraction of Core Boiling Boundary (FCBB)

LCO 3.2.4 The FCBB shall be ≤ 1.0.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. FCBB not within limit for reasons other than an unexpected loss of feedwater heating or unexpected reduction in core flow.	A. Restore FCBB to within limit.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR --------NOTE-------- Required Action B.1 and Required Action B.2 shall be completed if this Condition is entered due to an unexpected loss of feedwater heating or unexpected reduction in core flow. -------------------- FCBB not within limit due to an unexpected loss of feedwater heating or unexpected reduction in core flow.	B.1 Initiate action to exit the Restricted Region. AND B.2 Initiate action to return APRM Flow Biased Simulated Thermal Power-High Allowable Value to "non-Setup" value.	Immediately Immediately following exit of Restricted Region

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------

1. Refer to Table 3.3.1.1-1 to determine which SRs apply for each RPS Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains RPS trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.1.2	------------------NOTE------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 25% RTP. -- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power ≤ 2% RTP.	7 days
SR 3.3.1.1.3	Adjust the flow control trip reference card to conform to reactor flow.	Once within 7 days after reaching equilibrium conditions following refueling outage.
SR 3.3.1.1.4	------------------NOTE------------------ Not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	7 days

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
		(continued)
SR 3.3.1.1.5	Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.6	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.7	-------------------NOTE------------------- Only required to be met during entry into MODE 2 from MODE 1. -- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.8	Calibrate the local power range monitors.	1000 MWD/T average core exposure
SR 3.3.1.1.9	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.10	Calibrate the trip units.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.11	-----NOTES----- 1. Neutron detectors and flow reference transmitters are excluded. 2. For Function 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 3. For Function 2.b, the digital components of the flow control trip reference cards are excluded. ---------- Perform CHANNEL CALIBRATION.	184 days
SR 3.3.1.1.12	Perform CHANNEL FUNCTIONAL TEST.	18 months
SR 3.3.1.1.13	-----NOTES----- 1. Neutron detectors are excluded. 2. For IRMs, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ---------- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.1.14	Verify the APRM Flow Biased Simulated Thermal Power—High time constant is within the limits specified in the COLR.	18 months
SR 3.3.1.1.15	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
		(continued)
SR 3.3.1.1.16	Verify Turbine Stop Valve Closure and Turbine Control Valve Fast Closure Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ 40% RTP.	18 months
SR 3.3.1.1.17	Calibrate the flow reference transmitters.	18 months
SR 3.3.1.1.18	------------------NOTES------------------ 1. Neutron detectors are excluded. 2. For Functions 3, 4, and 5 in Table 3.3.1.1-1, the channel sensors are excluded. 3. For Function 6, "n" equals 4 channels for the purpose of determining the STAGGERED TEST BASIS Frequency. --- Verify the RPS RESPONSE TIME is within limits.	18 months on a STAGGERED TEST BASIS

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux - High	2	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 122/125 divisions of full scale
	5(a)	3	I	SR 3.3.1.1.1 SR 3.3.1.1.5 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 122/125 divisions of full scale
b. Inop	2	3	H	SR 3.3.1.1.4 SR 3.3.1.1.15	NA
	5(a)	3	I	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
2. Average Power Range Monitors					
a. Neutron Flux - High, Setdown	2	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.11 SR 3.3.1.1.15	≤ 20% RTP
b. Flow Biased Simulated Thermal Power - High	1	3	G	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.17 SR 3.3.1.1.18	(b)

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Allowable values specified in COLR. Allowable value modification required by the COLR due to reduction in feedwater temperature may be delayed for up to 12 hours.

3.3 INSTRUMENTATION

3.3.1.3 Period Based Detection System (PBDS)

LCO 3.3.1.3 One channel of PBDS instrumentation shall be OPERABLE.

<u>AND</u>

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

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

THERMAL POWER and core flow in the Monitored Region specified in the COLR.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any OPERABLE PBDS channel indicating Hi-Hi DR Alarm.	A.1 Place the reactor modeswitch in the shutdown position.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required PBDS channel inoperable while in the Restricted Region.	B.1 --------NOTE---------- Only applicable if RPS Function 2.b, APRM Flow Biased Simulated Thermal Power High, Allowable Value is "Setup". ----------------------	
	Initiate action to exit the Restricted Region.	Immediately
	<u>OR</u>	
	B.2 Place the reactor mode switch in the shutdown position.	Immediately
C. Required PBDS channel inoperable while in the Monitored Region.	C.1 Initiate action to exit the Monitored Region.	15 minutes

SURVEILLANCE REQUIREMENTS

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1

A. Two recirculation loops shall be in operation with matched flows.

OR

B. One recirculation loop shall be in operation with:

1. THERMAL POWER ≤ 83% RTP;
2. Total core flow within limits;
3. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR.
4. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR; and
5. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power- High), Allowable Value for single loop operation as specified in the COLR.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Recirculation loop jet pump flow mismatch not within limits.	A.1 Shutdown one recirculation loop.	2 hours
B. THERMAL POWER > 83% RTP during single loop operation.	B.1 Reduce THERMAL POWER to ≤ 83% RTP.	1 hour

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Requirements B.3, B.4, or B.5 of the LCO not met.	C.1 Satisfy the requirements of the LCO.	24 hours
D. Required actions and associated completion times of conditions A, B, or C not met. <u>OR</u> No recirculation loops in operation.	D.1 Be in Mode 3.	12 hours

SURVEILLANCE REQUIREMENTS

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

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5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and process control program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the main steam safety/relief valves, shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

   1) LCO 3.2.1, Average Planar Linear Heat Generation Rate (APLHGR),
   2) LCO 3.2.2, Minimum Critical Power Ratio (MCPR) (including power and flow dependent limits),
   3) LCO 3.2.3, Linear Heat Generation Rate (LHGR) (including power and flow dependent limits).
   4) LCO 3.2.4, Fraction of Core Boiling Boundary (FCBB)
   5) LCO 3.3.1.1, RPS Instrumentation (RPS)
   6) LCO 3.3.1.3, Periodic Based Detection System (PBDS)

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

1) NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel" (latest approved version);

2) NEDC-32489P (April 1996), "T-Factor Setdown Elimination Analysis for River Bend Station" (for power and flow dependent limits methodology only as evaluated and approved by Safety Evaluation and License Amendment 100 dated October 10, 1997).

3) NEDO-32339P-A, "Reactor Stability Long-Term Solution: Enhanced Option I-A," including Supplements 1 through 4 (latest revisions dated through April 1998).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 Fraction of Core Boiling Boundary (FCBB)

BASES

BACKGROUND

General Design Criterion 12 requires protection of fuel thermal safety limits from conditions caused by neutronic/ thermal hydraulic instability. Neutronic/thermal hydraulic instabilities result in power oscillations, which could result in exceeding the MCPR Safety Limit (SL). The MCPR SL ensures that at least 99.9% of the fuel rods avoid boiling transition during normal operation and during an anticipated operational occurrence (AOO) (refer to the Bases for SL 2.1.1.2).

The FCBB is the ratio of the power generated in the lower 4 feet of the active reactor core to the power required to produce bulk saturated boiling of the coolant entering the fuel channels. The value of 4 feet above the bottom of the active fuel is set as the boiling boundary limit based on analysis described in Section 9 of Reference 1. The boiling boundary limit is established to ensure that the core will remain stable during normal reactor operations in the Restricted Region of the power and flow map defined in the COLR which may otherwise be susceptible to neutronic/thermal hydraulic instabilities and therefore the MCPR SL remains protected.

Planned operation in the Restricted Region is accommodated by manually establishing the "Setup" values for the APRM flow-biased Simulated Thermal Power- High scram and control rod block functions. The "Setup" Allowable Values of the APRM Flow-Biased Thermal Power-High Function (refer to LCO 3.3.1.1, Table 3.3.1.1-1, Function 2.b.) are consistent with assumed operation in the Restricted Region with FCBB ≤ 1.0. Operation with the "Setup" values enables entry into the Restricted Region without a control rod block that would otherwise occur. Plant operation with the "Setup" values is limited as much as practical due to the effects on plant operation required to meet the FCBB limit.

APPLICABLE SAFETY ANALYSES

The analytical methods and assumptions used in establishing the boiling boundary limit are presented in Section 9 of Reference 1. Operation with the FCBB ≤ 1.0 (i.e., a bulk saturated boiling boundary ≥ 4 feet) is expected to ensure that operation within the Restricted Region will not result

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

in neutronic/thermal hydraulic instability due to either steady-state operation or as the result of an AOO which initiates and terminates entirely within the Restricted Region. Analysis also confirms that AOOs initiated from outside the Restricted Region (i.e., without an initial restriction on FCBB) which terminate in the Restricted Region are not expected to result in instability. The types of transients specifically evaluated are loss of flow and coolant temperature decrease which are limiting for the onset of instability (Ref. 1).

Although the onset of instability does not necessarily occur if the FCBB is greater than 1.0 in the Restricted Region, bulk saturated boiling at the 4 foot boiling boundary limit has been adopted to preclude neutronic/thermal hydraulic instability during operation in the Restricted Region. The effectiveness of this limit is based on the demonstration (Ref. 1) that with the limit met large margin to the onset of neutronic/thermal hydraulic instability exists and all major state parameters that affect stability have relatively small impacts on stability performance.

The FCBB satisfies Criterion 2 of the NRC Policy Statement.

LCO

Requiring FCBB ≤ 1.0 ensures the bulk coolant boiling boundary is ≥ 4 feet from the bottom of the active core. Analysis (Ref. 1) has shown that for anticipated operating conditions of core power, core flow, axial and radial power shapes, and inlet enthalpy, a boiling boundary of 4 feet ensures variations in these key parameters do not have a significant impact on stability performance.

Neutronic/thermal hydraulic instabilities result in power oscillations, which could result in exceeding the MCPR Safety Limit (SL). The MCPR SL ensures that at least 99.9% of the fuel rods avoid boiling transition during normal operation and during an AOO (refer to the Bases for SL 2.1.1.2).

APPLICABILITY

The FCBB limit is used to prevent core conditions necessary for the onset of instability and thereby preclude neutronic/thermal hydraulic instability while operating in the Restricted Region defined in the COLR.

(continued)

BASES

APPLICABILITY (continued)

The boundary of the Restricted Region in the Applicability of this LCO is analytically established in terms of thermal power and core flow. The Restricted Region is defined by the APRM Flow Biased Control Rod Block upscale alarm setpoints, which are a function of reactor recirculation drive flow. The Restricted Region Entry Alarm (RREA) signal is generated by the Flow Control Trip Reference (FCTR) card using the APRM Flow Biased Control Rod Block upscale alarm setpoints. As a result, the RREA is coincident with the Restricted Region boundary when the setpoints are not "Setup," and provides indication of entry into the Restricted Region. However, APRM Flow Biased Control Rod Block upscale alarm signals provided by the FCTR card, that are not coincident with the Restricted Region boundary, do not generate a valid RREA. The Restricted Region boundary for this LCO Applicability is specified in the COLR.

The FCBB limit is also used to ensure that core conditions, while operating with "Setup" values, remain consistent with analyzed transients initiated from inside and outside the Restricted Region.

When the APRM Flow Biased Control Rod Block upscale alarm setpoints are "Setup" the applicable setpoints used to generate the RREA are moved to the interior boundary of the Restricted Region to allow controlled operation within the Restricted Region. While the setpoints are "Setup" the Restricted Region boundary remains defined by the normal APRM Flow Biased Control Rod Block upscale alarm setpoints.

Parameters such as reactor power and core flow available at the reactor controls, may be used to provide immediate confirmation that entry into the Restricted Region could reasonably have occurred.

Operation outside the Restricted Region is not susceptible to neutronic/thermal hydraulic instability when applicable thermal power distribution limits such as MCPR are met.

(continued)

BASES

ACTIONS

A.1

If FCBB is not within the required limit, core conditions necessary for the onset of neutronic/hydraulic thermal instability may result. Therefore, prompt action should be taken to restore the FCBB to within the limit such that the stability of the core can be assured. Following uncontrolled entry into the Restricted Region, prompt restoration of FCBB within limit can be expected if FCBB is known to not significantly exceed the limit. Therefore, efforts to restore FCBB within limit following an uncontrolled entry into the Restricted Region are appropriate if operation prior to entry was consistent with planned entry or the potential for entry was recognized as demonstrated by FCBB being monitored and known to not significantly exceed the limit. Actions to exit the Restricted Region are appropriate when FCBB can not be expected to be restored in a prompt manner.

Actions to restart an idle recirculation loop, withdraw control rods, or reduce recirculation flow may result in approaching unstable reactor conditions and are not allowed to be used to comply with this Required Action. The 2-hour Completion Time is based on engineering judgment as to a reasonable time to restore the FCBB to within limit. The 2-hour Completion Time is acceptable based on the availability of the PBDS per Specification 3.3.1.3, "Period Based Detection System" and the low probability of a neutronic/thermal hydraulic instability event.

B.1 and B.2

Changes in reactor core state conditions resulting from an unexpected loss of feedwater heating or reduction in core flow (e.g., any unexpected reduction in feedwater temperature, recirculation pump trip, recirculation pump down shift to slow speed, or flow control valve closure) require immediate initiation of action to exit the Restricted Region and return the APRM Flow Biased Simulated Thermal Power- High Function (refer to LCO 3.3.1.1, Table 3.3.1.1-1, Function 2.b.) to the "non-Setup" value. Condition B is modified by a Note that specifies that Required Actions B.1 and B.2 must be completed if this Condition is entered due to an unexpected loss of feedwater heating or reduction in core flow.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

This action to exit the Restricted Region is required following unplanned events that occur while operating in the region and can result in significant loss of stability margin. During such unplanned events, adherence to the FCBB limit cannot be assured. Therefore, continued operation in the Restricted Region is not appropriate. The completion of Required Actions B.1 and B.2 is required even though FCBB may be calculated and determined to be within limit.

Core conditions continue to change after an unexpected loss of feedwater heating or reduction in core flow due to transient induced changes with the potential that the FCBB may change and the limit not be met. The potential for changing core conditions, with FCBB not met, is not consistent with operation in the Restricted Region or with the APRM Flow Biased Simulated Thermal Power-High Function "Setup". Therefore, actions to exit the Restricted Region and return the APRM Flow Biased Simulated Thermal Power-High Function to the "non-Setup" value are required to be completed in the event Condition B is entered due to an unexpected loss of feedwater heating or an unexpected reduction in core flow.

If operator actions to restore the FCBB to within limit are not successful within the specified Completion Time of Condition A, reactor operating conditions may be changing and may continue to change such that core conditions necessary for the onset of neutronic/thermal hydraulic instability may be met. Therefore, in the event the Required Action and associated Completion Time of Condition A is not met, immediate action to exit the Restricted Region and return the APRM Flow Biased Simulated Thermal Power-High Function to the "non-Setup" value is required.

Exit of the Restricted Region can be accomplished by control rod insertion and/or recirculation flow increases. Actions to restart an idle recirculation loop, withdraw control rods or reduce recirculation flow may result in approaching unstable reactor conditions and are not allowed to be used to comply with this Required Action. The time required to exit the Restricted Region will depend on existing plant conditions. Provided efforts are begun without delay and continued until the Restricted Region is exited, operation is acceptable.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

Verifying FCBB ≤ 1.0 is required to ensure the reactor is operating within the assumptions of the safety analysis. The boiling boundary limit is established to ensure that the core will remain stable during normal reactor operations in the Restricted Region of the power and flow map defined in the COLR which may otherwise be susceptible to neutronic/thermal hydraulic instabilities.

FCBB is required to be verified every 24 hours while operating in the Restricted Region defined in the COLR. The 24 hour Frequency is based on both engineering judgment and recognition of the slow rate of change in power distribution during normal operation.

The second Frequency requires FCBB to be within the limit within 15 minutes following an unexpected transient. The verification of the FCBB is required as a result of the possibility that the unexpected transient results in the limit not being met. The 15 minute frequency is based on both engineering judgment and the availability of the PBDS to provide the operator with information regarding the potential imminent onset of neutronic/thermal hydraulic instability. The 15 minute Frequency for this SR is not to be used to delay entry into Condition B following an unexpected reduction in feedwater heating, recirculation pump trip, recirculation pump down shift to slow speed, or significant flow control valve closure (small changes in flow control valve position are not considered significant). The action to exit the Restricted Region in Condition B is required following unplanned events that occur while operating in the region and can result in significant loss of stability margin. During such unplanned events, adherence to the FCBB limit cannot be assured. Therefore, continued operation in the restricted Region is not appropriate.

(continued)

BASES (continued)

This Surveillance is modified by a Note which allows 15 minutes to verify FCBB following entry into the Restricted Region if the entry was the result of an unexpected transient (i.e., an unintentional or unplanned change in core thermal power or core flow). The 15 minute allowance is based on both engineering judgment and the availability of the PBDS to provide the operator with information regarding the potential imminent onset of neutronic/thermal hydraulic instability. The 15 minute allowance of the Note is not to be used to delay entry into Condition B if the entry into the Restricted Region was the result of an unexpected reduction in feedwater heating, recirculation pump trip, recirculation pump down shift to slow speed, or significant flow control valve closure (small changes in flow control valve position are not considered significant).

REFERENCES

1. NEDO 32339-A, Revision 1, "Reactor Stability Long Term Solution: Enhanced Option I-A," April 1998.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power–High

The Average Power Range Monitor Flow Biased Simulated Thermal Power–High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow and is clamped at an upper limit that is always lower than the Average Power Range Monitor Fixed Neutron Flux–High Function Allowable Value. The Average Power Range Monitor Flow Biased Simulated Thermal Power–High Function provides a general definition of the licensed core power/core flow operating domain.

During continued operation with only one recirculation loop in service, the APRM flow biased setpoint is required to be conservatively set (refer to the Bases for LCO 3.4.1, "Recirculation Loops Operating" for more detailed discussion). The setpoint modification may be delayed in accordance with the allowances of LCO 3.4.1. After this time, the LCO 3.3.1.1 requirement for APRM OPERABILITY will enforce the more conservative setpoint.

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function is not associated with a limiting safety system setting. Operating limits established for the licensed operating domain are used to develop the Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function Allowable Values to provide pre-emptive reactor scram and prevent gross violation of the licensed operating domain. Operation outside the licensed operating domain may result in anticipated operational occurrences and postulated accidents being initiated from conditions beyond those assumed in the safety analysis. Operation within the licensed operating domain also ensures compliance with General Design Criterion 12.

General Design Criterion 12 requires protection of fuel thermal safety limits from conditions caused by neutronic/ thermal hydraulic instability. Neutronic/thermal hydraulic instabilities result in power oscillations, which could result in exceeding the MCPR SL.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power - High (continued)

The area of the core power and flow operating domain susceptible to neutronic/thermal hydraulic instability can be affected by reactor parameters such as reactor inlet feedwater temperature. Two complete and independent sets of Average Power Range Monitor Flow Biased Simulated Thermal Power High Function Allowable Values are specified in the COLR. Set 1 (Normal Trip Reference Set) provides protection against neutronic/thermal hydraulic instability during expected reactor operations. Set 2 (Alternate Trip Reference Set) provides protection against neutronic/thermal hydraulic instability during reactor operating conditions requiring added stability protection and is conservative with respect to Set 1. Feedwater temperature values requiring transition between flow control trip reference card sets are specified in the COLR.

In the event of a feedwater temperature reduction, Allowable Value modification (from the Normal Trip Reference Set to the Alternate Trip Reference Set) is required to preserve the margin associated with the potential for the onset of neutronic/thermal hydraulic instability which existed prior to the feedwater temperature reduction. The Allowable Value modification required by the COLR may be delayed up to 12 hours to allow time to adjust and check the adjustment of each flow control trip reference card. At the end of the 12 hour period, the Allowable Value modifications must be complete for all of the required channels or the applicable Condition(s) must be entered and the Required Actions taken. The 12 hour time period is acceptable based on the low probability of a neutronic/hydraulic instability event and the continued protection provided by the flow control trip reference card. In addition, when the feedwater temperature reduction results in operation in either the Restricted Region or Monitored Region, the requirements for the Period Based Detection System (LCO 3.3.1.3, Period Based Detection System (PBDS)) provide added protection against neutronic/ thermal hydraulic instability during the 12 hour time period.

The area of the core power and flow operating domain susceptible to neutronic/thermal hydraulic instability is affected by the value of Fraction of Core Boiling Boundary (LCO 3.2.4, FCBB). "Setup" and normal ("non-Setup") Average Power Range Monitor Flow Biased Simulated Thermal Power High Function Allowable Values are specified in the COLR. The normal ("non-Setup") value provides protection

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power - High (continued)

against neutronic/thermal hydraulic instability by preventing operation in the susceptible area of the operating domain when operating outside the Restricted Region specified in the COLR with the FCBB limit not required to be met. When the "Setup" value is selected, meeting the FCBB limit provides protection against instability.

"Setup" and "non-Setup" values are selected by operator manipulation of a Setup button on each flow control trip reference card. Selection of the "Setup" value is intended only for planned operation in the Restricted Region as specified in the COLR. Operation in the Restricted Region with the Average Power Range Monitor Flow Biased Simulated Thermal Power- High Function "Setup" requires the FCBB limit to be met and is not generally consistent with normal power operation.

The Average Power Range Monitor Flow Biased Simulated Thermal Power-High Function uses a trip level generated by the flow control trip reference card based on recirculation loop drive flow. Proper trip level generation as a function of drive flow requires drive flow alignment. This is accomplished by selection of appropriate dip switch positions on the flow control trip reference cards (Refer to SR 3.3.1.1.3). Changes in the core flow to drive flow functional relationship may vary over the core flow operating range. These changes can result from both gradual changes in recirculation system and core components over the reactor life time as well as specific maintenance performed on these components (e.g., jet pump cleaning).

The APRM System is divided into two groups of channels with four APRM inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one Average Power Range Monitor channel in a trip system can cause the associated trip system to trip.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power - High (continued)

Six channels of Average Power Range Monitor Flow Biased Simulated Thermal Power - High, with three channels in each trip system arranged in one-out-of-three logic, are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 11 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.

Each APRM channel receives one total drive flow signal. The recirculation loop drive flow signals are generated by eight flow units. One flow unit from each recirculation loop is provided to each APRM channel. Total drive flow is determined by each APRM by summing up the flow signals provided to the APRM from the two recirculation loops.

The THERMAL POWER time constant of 6.6 seconds is based on the fuel heat transfer dynamics and provides a signal proportional to the THERMAL POWER.

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function is required to be OPERABLE in MODE 1 when there is the possibility of neutronic/thermal hydraulic instability. The potential to exceed the SL applicable to high pressure and core flow conditions (MCPR SL), which provides fuel cladding integrity protection, exists if neutronic/thermal hydraulic instability can occur. During MODES 2 and 5, other IRM and APRM Functions provide protection for fuel cladding integrity.

2.c. Average Power Range Monitor Fixed Neutron Flux - High

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux increases. The Average Power Range Monitor Fixed Neutron Flux - High Function is capable of generating a trip signal to prevent fuel damage or excessive RCS pressure. For the overpressurization protection analysis of Reference 2, the Average Power Range Monitor Fixed Neutron Flux - High Function is assumed to terminate the main steam isolation valve (MSIV) closure event and, along with the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.c. Average Power Range Monitor Fixed Neutron Flux —High (continued)

safety/relief valves (S/RVs), limits the peak reactor pressure vessel (RPV) pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis (Ref. 7) takes credit for the Average Power Range Monitor Fixed Neutron Flux —High Function to terminate the CRDA.

The APRM System is divided into two groups of channels with four APRM channels inputting to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Six channels of Average Power Range Monitor Fixed Neutron Flux—High with three channels in each trip system arranged in a one-out-of-three logic are required to be OPERABLE to ensure that no single instrument failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least 11 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.

The Allowable Value is based on the Analytical Limit assumed in the CRDA analyses.

The Average Power Range Monitor Fixed Neutron Flux —High Function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the SLs (e.g., MCPR and RCS pressure) being exceeded. Although the Average Power Range Monitor Fixed Neutron Flux —High Function is assumed in the CRDA analysis that is applicable in MODE 2, the Average Power Range Monitor Neutron Flux —High, Setdown Function conservatively bounds the assumed trip and, together with the assumed IRM trips, provides adequate protection. Therefore, the Average Power Monitor Fixed Neutron Flux—High Function is not required in MODE 2.

2.d. Average Power Range Monitor —Inop

This signal provides assurance that a minimum number of APRMs are OPERABLE. Anytime an APRM mode switch is moved to any position other than Operate, an APRM module is unplugged, or the APRM has too few LPRM inputs (<11) as indicated by low voltage an inoperable trip signal will be

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.2

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoints," allows the APRMs to be reading greater than actual THERMAL POWER to compensate for localized power peaking. When this adjustment is made, the requirement for the APRMs to indicate within 2% RTP of calculated power is modified to require the APRMs to indicate 100% of calculated MFLPD. The Frequency of once per 7 days is based on minor changes in LPRM sensitivity, which could affect the APRM reading between performances of SR 3.3.1.1.8.

A restriction to satisfying this SR when < 25% RTP is provided that requires the SR to be met only at 25% RTP because it is difficult to accurately maintain APRM indication of core THERMAL POWER consistent with a heat balance when < 25% RTP. At low power levels, a high degree of accuracy is unnecessary because of the large inherent margin to thermal limits (MCPR and APLHGR). At 25% RTP, the Surveillance is required to have been satisfactorily performed within the last 7 days in accordance with SR 3.0.2. A Note is provided which allows an increase in THERMAL POWER above 25% if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after reaching or exceeding 25% RTP. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

SR 3.3.1.1.3

The Average Power Range Monitor Flow Biased Simulated Thermal Power High Function uses a trip level generated by the flow control trip reference card based on the recirculation loop drive flow. The drive flow is adjusted by a digital algorithm according to selected drive flow alignment dip switch settings. This SR sets the flow control trip reference card to ensure the drive flow alignment used results in the appropriate trip level being generated from the digital components of the card.

The Frequency of once following a refueling outage is based on the expectation that any change in the core flow to drive flow functional relationship during power operation would be gradual and

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.3 (continued)

that maintenance on recirculation system and core components which may impact the relationship is expected to be performed during refueling outages. The completion time of 7 days after reaching equilibrium conditions is based on plant conditions required to perform the test and engineering judgment of the time required to collect and analyze the necessary flow data and the time required to adjust and check the adjustment of each flow control trip reference card. The completion time of 7 days after reaching equilibrium conditions is acceptable based on the low probability of a neutronic/hydraulic instability event.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.4

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

As noted, SR 3.3.1.1.4 is not required to be performed when entering MODE 2 from MODE 1 since testing of the MODE 2 required IRM and APRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This allows entry into MODE 2 if the 7 day Frequency is not met per SR 3.0.2. In this event, the SR must be performed within 12 hours after entering MODE 2 from MODE 1. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

A Frequency of 7 days provides an acceptable level of system average availability over the Frequency interval and is based on reliability analysis (Ref. 9).

SR 3.3.1.1.5

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended Function. A Frequency of 7 days provides an acceptable level of system average availability over the Frequency and is based on the reliability analysis of Reference 9. (The Manual Scram Function's CHANNEL FUNCTIONAL TEST Frequency was credited in the analysis to extend many automatic scram Functions' Frequencies.)

SR 3.3.1.1.6 and SR 3.3.1.1.7

These Surveillances are established to ensure that no gaps in neutron flux indication exist from subcritical to power operation for monitoring core reactivity status. The overlap between SRMs and IRMs is required to be demonstrated to ensure that reactor power will not be increased into a region without adequate neutron flux indication. This is

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.11, SR 3.3.1.1.13, and SR 3.3.1.1.17

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

For functions 9 and 10, the CHANNEL CALIBRATION shall include the turbine first stage pressure instruments.

Note 1 states that neutron detectors and flow reference transmitters are excluded from CHANNEL CALIBRATION because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 MWD/T LPRM calibration against the TIPs (SR 3.3.1.1.8). Calibration of the flow reference transmitters is performed on an 18 month Frequency (SR 3.3.1.1.17). A second Note is provided that requires the APRM and IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR. The Frequency of SR 3.3.1.1.11, SR 3.3.1.1.13, and SR 3.3.1.1.17 is based upon the assumption of the magnitude of equipment drift in the setpoint analysis.

Note 3 states that the digital components of the flow control trip reference card are excluded from CHANNEL CALIBRATION of Function 2.b, Average Power Range Monitor Flow Biased Simulated Thermal Power-High. The analog output potentiometers of the flow control trip reference card are not excluded. The flow control trip reference card has an automatic self-test feature, which periodically tests the hardware which performs the digital algorithm. Exclusion of the digital components of the flow control trip reference card from CHANNEL CALIBRATION of Function 2.b is based on the conditions required to perform the test and the likelihood of a change in the status of these components not being detected.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.14

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function uses an electronic filter circuit to generate a signal proportional to the core THERMAL POWER from the APRM neutron flux signal. This filter circuit is representative of the fuel heat transfer dynamics that produce the relationship between the neutron flux and the core THERMAL POWER. The filter time constant is specified in the COLR and must be verified to ensure that the channel is accurately reflecting the desired parameter. The Frequency of 18 months is based on engineering judgment and reliability of the components.

SR 3.3.1.1.15

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required trip logic for a specific channel. The functional testing of control rods, in LCO 3.1.3, "Control Rod OPERABILITY," and SDV vent and drain valves, in LCO 3.1.8, "Scram Discharge Volume (SDV) Vent and Drain Valves," overlaps this Surveillance to provide complete testing of the assumed safety function.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.1.1.16

This SR ensures that scrams initiated from the Turbine Stop Valve Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low Functions will not be inadvertently bypassed when THERMAL POWER is ≥40% RTP. This involves calibration of the bypass channels. Adequate margins for the instrument setpoint methodology are incorporated into the actual setpoint. Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from turbine first stage pressure), the main turbine bypass valves must remain closed at THERMAL POWER ≥40% RTP to ensure that the calibration remains valid.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.16 (continued)

If any bypass channel setpoint is nonconservative (i.e., the Functions are bypassed at ≥40% RTP, either due to open main turbine bypass valve(s) or other reasons), then the affected Turbine Stop Valve Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure—Low Functions are considered inoperable. Alternatively, the bypass channel can be placed in the conservative condition (nonbypass). If placed in the nonbypass condition, this SR is met and the channel is considered OPERABLE.

The Frequency of 18 months is based on engineering judgment and reliability of the components.

SR 3.3.1.1.18

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. The RPS RESPONSE TIME acceptance criteria are included in Reference 10.

As noted, neutron detectors are excluded from RPS RESPONSE TIME testing because the principles of detector operation virtually ensure an instantaneous response time.

RPS RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. Note 2 requires STAGGERED TEST BASIS Frequency to be determined based on 4 channels per trip system, in lieu of the 8 channels specified in Table 3.3.1.1-1 for the MSIV Closure Function. This Frequency is based on the logic interrelationships of the various channels required to produce an RPS scram signal. Therefore, staggered testing results in response time verification of these devices every 18 months. This Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience, which shows that random failures of instrumentation components causing serious time degradation, but not channel failure, are infrequent.

REFERENCES

1. USAR, Figure 7.2-1.

2. USAR, Section 5.2.2.

3. USAR, Section 6.3.3.

(continued)

BASES

REFERENCES (continued)

4. USAR, Chapter 15.

5. USAR, Section 15.4.1.

6. NEDO-23842, "Continuous Control Rod Withdrawal in the Startup Range," April 18, 1978.

7. USAR, Section 15.4.9.

8. Letter, P. Check (NRC) to G. Lainas (NRC), "BWR Scram Discharge System Safety Evaluation," December 1, 1980, as attached to NRC Generic Letter dated December 9, 1980.

9. NEDO-30851-P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System," March 1988.

10. RBS Technical Requirements Manual.

11. NEDO-32291-A, "System Analysis for Elimination of Selected Response Time Testing Requirements," January 1994.

B 3.3 INSTRUMENTATION

B 3.3.1.3 Period Based Detection System (PBDS)

BASES

BACKGROUND

General Design Criterion 12 requires protection of fuel thermal safety limits from conditions caused by neutronic/ thermal hydraulic instability. Neutronic/thermal hydraulic instabilities can result in power oscillations which could result in exceeding the MCPR Safety Limit (SL). The MCPR SL ensures that at least 99.9% of the fuel rods avoid boiling transition during normal operation and during an anticipated operational occurrence (AOO) (refer to the Bases for SL 2.1.1.2).

The PBDS provides the operator with an indication that conditions consistent with a significant degradation in the stability performance of the reactor core has occurred and the potential for imminent onset of neutronic/thermal hydraulic instability may exist. Indication of such degradation is cause for the operator to initiate an immediate reactor scram if the reactor is being operated in either the Restricted Region or Monitored Region. The Restricted Region and Monitored Region are defined in the COLR.

The PBDS instrumentation of the Neutron Monitoring System consists of two channels. Each of the PBDS channels includes input from a minimum of 8 local power range monitors (LPRMs) within the reactor core. These inputs are continually monitored by the PBDS for variations in the neutron flux consistent with the onset of neutronic/thermal hydraulic instability. Each channel includes separate local indication, but share a common control room Hi-Hi DR Alarm. While this LCO specifies OPERABILITY requirements only for one monitoring and indication channel of the PBDS, if both are OPERABLE, a Hi-Hi DR Alarm from either channel results in the need for the operator to take actions.

The primary PBDS component is a card in the Neutron Monitoring System with analog inputs and digital processing. The PBDS card has an automatic self-test feature to periodically test the hardware circuit. The self-test functions are executed during their allocated portion of the executive loop sequence. Any self-test failure indicating loss of critical function results in a control room alarm. The inoperable condition is also displayed

(continued)

BASES

BACKGROUND (continued)

by an indicating light on the card front panel. A manually initiated internal test sequence can be actuated via a recessed push button. This internal test consists of simulating alarm and inoperable conditions to verify card OPERABILITY. Descriptions of the PBDS are provided in References 1 and 2.

Actuation of the PBDS Hi-Hi DR Alarm is not postulated to occur due to neutronic/thermal hydraulic instability outside the Restricted Region and the Monitored Region. Periodic perturbations can be introduced into the thermal hydraulic behavior of the reactor core from external sources such as recirculation system components and the pressure and feedwater control systems. These perturbations can potentially drive the neutron flux to oscillate within a frequency range expected for neutronic/thermal hydraulic instability. The presence of such oscillations would be recognized by the period based algorithm of the PBDS and potentially result in a Hi-Hi DR Alarm. Actuation of the PBDS Hi-Hi DR Alarm outside the Restricted Region and the Monitored Region would indicate the presence of a source external to the reactor core and are not indications of neutronic/thermal hydraulic instability.

APPLICABLE SAFETY ANALYSES

Analysis, as described in Section 4 of Reference 1, confirms that AOOs initiated from outside the Restricted Region without stability control and from within the Restricted Region with stability control are not expected to result in neutronic/thermal hydraulic instability. The stability control applied in the Restricted Region (refer to LCO 3.2.5, "Fraction of Core Boiling Boundary (FCBB)") is established to prevent neutronic/thermal hydraulic instability during operation in the Restricted Region. Operation in the Monitored Region is only susceptible to instability under hypothetical operating conditions beyond those analyzed in Reference 1. The types of transients specifically evaluated are loss of flow and coolant temperature decrease, which are limiting for the onset of instability.

The initial conditions assumed in the analysis are reasonability conservative and the immediate post-event reactor conditions are significantly stable. However, these assumed initial conditions do not bound each individual parameter which impacts stability performance (Ref.1). The PBDS instrumentation provides the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

operator with an indication that conditions consistent with a significant degradation in the stability performance of the reactor core has occured and the potential for imminent onset of neutronic/thermal hydraulic instability may exist. Such conditions are only postulated to result from events initiated from initial conditions beyond the conditions assumed in the safety analysis (refer to Section 4, Ref. 1).

The PBDS has no safety function and is not assumed to function during any FSAR design basis accident or transient analysis. However, the PBDS provides the only indication of the imminent onset of neutronic/thermal hydraulic instability during operation in regions of the operating domain potentially susceptible to instability. Therefore, the PBDS is included in the Technical Specifications.

LCO

One PBDS channel is required to be OPERABLE to monitor reactor neutron flux for indications of imminent onset of neutronic/thermal hydraulic instability. OPERABILITY requires the ability for the operator to be immediately alerted to a Hi-Hi DR Alarm. This is accomplished by the instrument channel control room alarm. The LCO also requires reactor operation be such that the Hi-Hi DR Alarm is not actuated by any OPERABLE PBDS instrumentation channel.

APPLICABILITY

At least one of two PBDS instrumentation channels is required to be OPERABLE during operation in either the Restricted Region or the Monitored Region specified in the COLR. Similarly, operation with the PBDS Hi-Hi DR Alarm of any OPERABLE PBDS instrumentation channel is not allowed in the Restricted Region or the Monitored Region. Operation in these regions is susceptible to instability (refer to the Bases for LCO 3.2.5 and Section 4 of Ref. 1). OPERABILITY of at least one PBDS instrumentation channel and operation with no indication of a PBDS Hi-Hi DR Alarm from any OPERABLE PBDS instrumentation channel is therefore required during operation in these regions.

The boundary of the Restricted Region in the Applicability of this LCO is analytically established in terms of thermal power and core

BASES

APPLICABILITY (continued)

flow. The Restricted Region is defined by the APRM Flow Biased Control Rod Block upscale setpoints, which are a function of reactor recirculation drive flow. The Restricted Region Entry Alarm (RREA) signal is generated by the Flow Control Trip Reference (FCTR) card using the APRM Flow Biased Control Rod Block upscale alarm setpoints. As a result, the RREA is coincident with the Restricted Region boundary when the setpoints are not "Setup," and provides indication of entry into the Restricted Region. However, APRM Flow Biased Control Rod Block upscale alarm signals provided by the FCTR card, that are not coincident with the Restricted Region boundary, do not generate a valid RREA. The Restricted Region boundary for this LCO Applicability is specified in the COLR.

When the APRM Flow Biased Control Rod Block upscale alarm setpoints are "Setup" the applicable setpoints used to generate the RREA are moved to the interior boundary of the Restricted Region to allow controlled operation within the Restricted Region. While the setpoints are "Setup" the Restricted Region boundary remains defined by the normal APRM Flow Control Rod Block upscale alarm setpoints.

Parameters such as reactor power and core flow available at the reactor controls, may be used to provide immediate confirmation that entry into the Restricted Region could reasonably have occurred.

The Monitored Region in the Applicability of this LCO is analytically established in terms of thermal power and core flow. However, unlike the Restricted Region boundary the Monitored Region boundary is not specifically monitored by plant instrumentation to provide automatic indication of region entry. Therefore, the Monitored Region boundary is defined in terms of thermal power and core flow. The Monitored Region boundary for this LCO Applicability is specified in the COLR.

Operation outside the Restricted Region and the Monitored Region is not susceptible to neutronic/thermal hydraulic instability even under extreme postulated conditions.

(continued)

BASES

ACTIONS

A.1

If at any time while in the Restricted Region or Monitored Region, an OPERABLE PBDS instrumentation channel indicates a valid Hi-Hi DR Alarm, the operator is required to initiate an immediate reactor scram. Verification that the Hi-Hi DR Alarm is valid may be performed without delay against another output from a PBDS card observable from the reactor controls in the control room prior to the manual reactor scram. This provides assurance that core conditions leading to neutronic/thermal hydraulic instability will be mitigated. This Required Action and associated Completion Time does not allow for evaluation of circumstances leading to the Hi-Hi DR Alarm prior to manual initiation of reactor scram.

B.1 and B.2

Operation with the APRM Flow Biased Simulated Thermal Power?High Function (refer to LCO 3.3.1.1, Table 3.3.1.1-1, Function 2.b.) "Setup" requires the stability control applied in the Restricted Region (refer to LCO 3.2.5) to be met. Requirements for operation with the stability control met are established to prevent reactor thermal hydraulic instability during operation in the Restricted Region. With the required PBDS channel inoperable, the ability to monitor conditions indicating the potential for imminent onset of neutronic/thermal hydraulic instability as a result of unexpected transients is lost. Therefore, action must be immediately initiated to exit the Restricted Region. While the APRM Flow Biased Control Rod Block upscale alarm setpoints are "Setup," operation in the Restricted Region may be confirmed by use of plant parameters such as reactor power and core flow available at the reactor controls.

Exit of the Restricted Region can be accomplished by control rod insertion and/or recirculation flow increases. Actions to restart an idle recirculation loop, withdraw control rods or reduce recirculation flow may result in unstable reactor conditions and are not allowed to be used to comply with this Required Action.

The time required to exit the Restricted Region will depend on existing plant conditions. Provided efforts are begun without delay and continued until the Restricted Region is exited, operation is acceptable based on the low probability of a transient which degrades stability performance occurring simultaneously with the required PBDS channel inoperable.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Required Action B.1 is modified by a Note that specifies that initiation of action to exit the Restricted Region only applies if the APRM Flow Biased Simulated Thermal Power-High Function is "Setup". Operation in the Restricted Region without the APRM Flow Biased Simulated Thermal Power-High Function "Setup" indicates uncontrolled entry into the Restricted Region. Uncontrolled entry is consistent with the occurrence of unexpected transients, which, in combination with the absence of stability controls being met may result in significant degradation of stability performance.

When the APRM Flow Biased Control Rod Block upscale alarm setpoints are not "Setup" uncontrolled entry into the Restricted Region is identified by receipt of a valid RREA. Immediate confirmation that the RREA is valid and indicates an actual entry into the Restricted Region may be performed without delay. Immediate confirmation constitutes observation that plant parameters immediately available at the reactor controls (e.g., reactor power and core flow) are reasonably consistent with entry into the Restricted Region. This immediate confirmation may also constitute recognition that plant parameters are rapidly changing during a transient (e.g., a recirculation pump trip) which could reasonably result in entry into the Restricted Region.

For uncontrolled entry into the Restricted Region with the required PBDS instrumentation channel inoperable, the ability to monitor conditions indicating the potential for imminent onset of neutronic/thermal hydraulic instability is lost and continued operation is not justified. Therefore, Required Action B.2 requires immediate reactor scram.

C.1

In the Monitored Region the PBDS Hi-Hi DR Alarm provides indication of degraded stability performance. Operation in the Monitored Region is susceptible to neutronic/thermal hydraulic instability under postulated conditions exceeding those previously assumed in the safety analysis. With the required PBDS channel inoperable, the ability to monitor conditions indicating the potential for imminent onset of neutronic/thermal hydraulic instability is lost. Therefore, action must be initiated to exit the Monitored Region.

(continued)

BASES

ACTIONS

C.1 (continued)

Actions to restart an idle recirculation loop, withdraw control rods or reduce recirculation flow may result in approaching unstable reactor conditions and are not allowed to be used to comply with this Required Action. Exit of the Monitored Region is accomplished by control rod insertion and/or recirculation flow increases. However, actions are allowed provided the Fraction of Core Boiling Boundary (FCBB) is recently (within 15 minutes) verified to be ≤ 1.0. Recent verification of FCBB being met provides assurance that with the PBDS inoperable, planned decreases in recirculation drive flow should not result in significant degradation of core stability performance.

The specified Completion Time of 15 minutes ensures timely operator action to exit the region consistent with the low probability that reactor conditions exceed the initial conditions assumed in the safety analysis. The time required to exit the Monitored Region will depend on existing plant conditions. Provided efforts are begun within 15 minutes and continued until the Monitored Region is exited, operation is acceptable based on the low probability of a transient which degrades stability performance occurring simultaneously with the required PBDS channel inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.3.1.3.1

During operation in the Restricted Region or the Monitored Region the PBDS Hi-Hi DR Alarm is relied upon to indicate conditions consistent with the imminent onset of neutronic/ thermal hydraulic instability. Verification every 12 hours provides assurance of the proper indication of the alarm during operation in the Restricted Region or the Monitored Region. The 12 hour Frequency supplements less formal, but more frequent, checks of alarm status during operation.

SR 3.3.1.3.2

Performance of the CHANNEL CHECK every 12 hours ensures that a gross failure of instrumentation has not occurred. This CHANNEL CHECK is normally a comparison of the PBDS indication to the state of the annunciator, as well as comparison to the same parameter on

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.3.2 (continued)

the other channel if it is available. It is based on the assumption that the instrument channel indication agrees with the immediate indication available to the operator, and that instrument channels monitoring the same parameter should read similarly. Deviations between the instrument channels could be an indication of instrument component failure. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL FUNCTIONAL TEST. Agreement criteria are determined by the plant staff based on a combination of the channel instrument uncertainties, including indication and readability.

The 12 hour Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.1.3.3

A CHANNEL FUNCTIONAL TEST is performed for the PBDS to ensure that the entire system will perform the intended function. The CHANNEL FUNCTIONAL TEST for the PBDS includes manual initiation of an internal test sequence and verification of appropriate alarm and inop conditions being reported.

Performance of a CHANNEL FUNCTIONAL TEST at a Frequency of 24 months verifies the performance of the PBDS and associated circuitry. The Frequency considers the plant conditions required to perform the test, the ease of performing the test, and the likelihood of a change in the system or component status. The alarm circuit is designed to operate for over 24 months with sufficient accuracy on signal amplitude and signal timing considering environment, initial calibration and accuracy drift (Ref. 2).

REFERENCES

1. NEDO 32339, Revision 1, "Reactor Stability Long Term Solution: Enhanced Option I-A," April 1998.

2. NEDO-32339P-A, Supplement 2, "Reactor Stability Long Term Solution: Enhanced Option I-A Solution Design," April 1998.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 Recirculation Loops Operating

BASES

BACKGROUND

The Reactor Coolant Recirculation System is designed to provide a forced coolant flow through the core to remove heat from the fuel. The forced coolant flow removes more heat from the fuel than would be possible with just natural circulation. The forced flow, therefore, allows operation at significantly higher power than would otherwise be possible. The recirculation system also controls reactivity over a wide span of reactor power by varying the recirculation flow rate to control the void content of the moderator. The Reactor Coolant Recirculation System consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps. Each external loop contains a two speed motor driven recirculation pump, a flow control valve, and associated piping, jet pumps, valves, and instrumentation. The recirculation loops are part of the reactor coolant pressure boundary and are located inside the drywell structure. The jet pumps are reactor vessel internals.

The recirculated coolant consists of saturated water from the steam separators and dryers that has been subcooled by incoming feedwater. This water passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold, from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the suction flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the driving flow. The drive flow and suction flow are mixed in the jet pump throat section. The total flow then passes through the jet pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core.

(continued)

BASES

BACKGROUND (continued)

The subcooled water enters the bottom of the fuel channels and contacts the fuel cladding, where heat is transferred to the coolant. As it rises, the coolant begins to boil, creating steam voids within the fuel channel that continue until the coolant exits the core. Because of reduced moderation, the steam voiding introduces negative reactivity that must be compensated for to maintain or to increase reactor power. The recirculation flow control allows operators to increase recirculation flow and sweep some of the voids from the fuel channel, overcoming the negative reactivity void effect. Thus, the reason for having variable recirculation flow is to compensate for reactivity effects of boiling over a wide range of power generation (i.e., 55 to 100% RTP) without having to move control rods and disturb desirable flux patterns.

Each recirculation loop is manually started from the control room. The recirculation flow control valves provide regulation of individual recirculation loop drive flows. The flow in each loop can be manually or automatically controlled.

APPLICABLE SAFETY ANALYSES

The operation of the Reactor Coolant Recirculation System is an initial condition assumed in the design basis loss of coolant accident (LOCA) (Ref. 1). During a LOCA caused by a recirculation loop pipe break, the intact loop is assumed to provide coolant flow during the first few seconds of the accident. The initial core flow decrease is rapid because the recirculation pump in the broken loop ceases to pump reactor coolant to the vessel almost immediately. The pump in the intact loop coasts down relatively slowly. This pump coastdown governs the core flow response for the next several seconds until the jet pump suction is uncovered (Ref. 1). The analyses assume that both loops are operating at the same flow prior to the accident. However, the LOCA analysis was reviewed for the case with a flow mismatch between the two loops, with the pipe break assumed to be in the loop with the higher flow. While the flow coastdown and core response are potentially more severe in this assumed case, (since the intact loop starts at a lower flow rate and the core response is the same as if both loops were operating at the lower flow rate) a small mismatch has been determined to be acceptable based on engineering judgement.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients (Ref. 2), which are analyzed in Chapter 15 of the USAR.

A plant specific LOCA analysis has been performed assuming only one operating recirculation loop. This analysis has demonstrated that, in the event of a LOCA caused by a pipe break in the operating recirculation loop, the Emergency Core Cooling System response will provide adequate core cooling, provided the APLHGR requirements are modified accordingly (Ref. 3).

The transient analyses of Chapter 15 of the USAR have also been performed for single recirculation loop operation (Ref. 3) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System average power range monitor (APRM) instrument setpoints is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR and MCPR limits for single loop operation are specified in the COLR. The APRM flow biased simulated thermal power setpoint is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation."

Recirculation loops operating satisfies Criterion 2 of the NRC Policy Statement.

LCO

Two recirculation loops are normally required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. Alternatively, with only one recirculation loop in operation, THERMAL POWER must be ≤ 83% RTP, the total core flow limitations identified above must be met, modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM

(continued)

BASES

LCO (continued)

CRITICAL POWER RATIO (MCPR)"), and APRM Flow Biased Simulated Thermal Power-High setpoint (LCO 3.3.1.1) must be applied to allow continued operation consistent with the assumptions of Reference 3.

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor Coolant Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur.

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

ACTIONS

A.1

With both recirculation loops operating but the flows not matched, the recirculation loops must be restored to operation with matched flows within 2 hours. If the flow mismatch cannot be restored to within limits within 2 hours, one recirculation loop must be shutdown.

Alternatively, if the single loop requirements of the LCO are applied to operating limits and RPS setpoints, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence.

The 2 hour Completion Time is based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

(continued)

BASES

ACTIONS (continued)

B.1

Should a LOCA or transient occur with THERMAL POWER > 83% RTP, during single loop operation the core response may not be bounded by the safety analyses. Therefore, only a limited time is allowed to reduce THERMAL POWER to ≤ 83% RTP.

The 1 hour Completion Time is based on the low probability of an accident occuring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing changes in THERMAL POWER to be quickly detected.

(continued)

BASES

ACTIONS (continued)

C.1

With the requirements of the LCO not met, the recirculation loops must be restored to operation with matched flows within 24 hours. A recirculation loop is considered not in operation when the pump in that loop is idle or when the mismatch between total jet pump flows of the two loops is greater than required limits. The loop with the lower flow must be considered not in operation. Should a LOCA occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status.

Alternatively, if the single loop requirements of the LCO are applied to operating limits and RPS setpoints, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence.

The 24 hour Completion Time is based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

This Required Action does not require tripping the recirculation pump in the lowest flow loop when the mismatch between total jet pump flows of the two loops is greater than the required limits. However, in cases where large flow mismatches occur, low flow or reverse flow can occur in the low flow loop jet pumps, causing vibration of the jet pumps. If zero or reverse flow is detected, the condition should be alleviated by changing flow control valve position to re-establish forward flow or by tripping the pump.

(continued)

BASES

ACTIONS (continued)

D.1

With no recirculation loops in operation, or the Required Action and associated Completion Time of Condition A not met, the unit is required to be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This SR ensures the recirculation loop flows are within the allowable limits for mismatch. At low core flow (i.e., < 70% of rated core flow), the MCPR requirements provide larger margins to the fuel cladding integrity Safety Limit such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can therefore be allowed when core flow is < 70% of rated core flow. The recirculation loop jet pump flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop.

The mismatch is measured in terms of percent of rated core flow. This SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

(continued)

BASES

REFERENCES

1. USAR, Section 6.3.3.
2. USAR, Section 5.4.1.4.
3. USAR, Section 15.0.6.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 106 TO FACILITY OPERATING LICENSE NO. NPF-47

ENTERGY OPERATIONS, INC.

RIVER BEND STATION, UNIT 1

DOCKET NO. 50-458

1.0 INTRODUCTION

By application dated October 8, 1998, as supplemented by letter dated April 15, 1999, Entergy Operations, Inc. (the licensee) requested changes to the Technical Specifications (TSs) for the River Bend Station, Unit 1 (River Bend). The proposed changes would implement the Boiling Water Reactor Owners Group (BWROG) Enhanced Option I-A for the reactor stability long-term solution to the neutronic and thermal hydraulic instability that is documented in NEDO-32339, Revision 1, "Reactor Stability Long-Term Solution, Enhanced Option I-A." The amendment would delete the limits on power and flow conditions in the specification on recirculating loops operating, add two new specifications to the TSs to establish limits for fraction of core boiling boundary and the period based detection system, modify the surveillance requirements for reactor protection instrumentation, and add the NEDO document to the core operating limits report. The associated Bases pages are also revised to reflect the TS changes. By letter dated April 15, 1999, the licensee provided additional information that did not change the initial proposed no significant hazards consideration determination originally published in the Federal Register on November 18, 1998 (63 FR 64112).

2.0 EVALUATION

The proposed revisions are described as follows:

2.1 TS 3.2.4, Fraction of Core Boiling Boundary (FCBB)

The proposed change is to add a new TS 3.2.4, Fraction of Core Boiling Boundary (FCBB), and its associated Bases to the section under TS 3.2, Power Distribution Limits. The proposed new TS and Bases are consistent with those described in the BWROG Reactor Stability Long-Term Solution: Enhanced Option I-A (Reference 2). However, modifications to the BWROG Bases in the proposed new Bases for River Bend include: (1) delete references to "Simulated Thermal Power High" on page B 3.2-14 under Applicability; (2) delete the wording "recirculation pump run back" on page B 3.2-15 under Actions B.1 and B.2, and on pages B 3.2-17 and B 3.2-18 under Surveillance Requirement (SR) 3.2.4; and (3) add the following wording to Actions B.1 and B.2 on page B 3.2-15, and to SR 3.2.4.1 on page B 3.2-17: "This action to exit the Restricted Region is required following unplanned events that occur while operating in the region and can result in significant loss of stability margin. During such unplanned events, adherence to the FCBB limit cannot be assured. Therefore, continued operation in the restricted region is not appropriate."

The FCBB is the ratio of the power generated in the lower fourth of the active reactor core to the power required to produce bulk saturated boiling of the coolant entering the fuel channels (Reference 2). The boiling boundary limit is established to ensure that the core remains stable during normal reactor operations in the Restricted Region of the power and flow operating domain, otherwise, the core may be susceptible to neutronic/thermal hydraulic instability.

The associated operating limit, FCBB, is required to be met during operation in the Restricted Region and meets Criterion 2 of Title 10 of the Code of Federal Regulations, Section 50.36(c)(2)(ii). Therefore, it is necessary to add a new specification (TS 3.2.4) to the Power Distribution Limits section of the TSs. In addition, the modifications to the BWROG generic EIA Bases are required for River Bend because: (1) River Bend will use the control rod block upscale alarm to define when it has entered the restricted region. This alarm is located on the control room front panel, in contrast to the simulated thermal power - high located at the back of the control room; (2) Recirculation pump run back is not applicable to River Bend; and (3) The action statement provides guidance to the operator to exit the restricted region following unplanned events that occur while operating in the region and can result in significant loss of stability margin. The staff has reviewed these proposed changes and finds them acceptable.

2.2 TS 3.3.1.1, Reactor Protection System Instrumentation

The proposed changes to the SRs are (1) to renumber SR 3.3.1.1.18 as SR 3.3.1.1.3; (2) to add Note 3 to SR 3.3.1.1.11 Function 2.b to exclude the digital components of the flow control trip reference (FCTR) card from the Channel Calibration; (3) to revise footnote (b) of Function 2.b in Table 3.3.1.1-1 to relocate the Allowable Value of the Average Power Range Monitor (APRM) flow biased scram function to the Core Operating Limits Report (COLR); and (4) to revise Bases B 3.3.1.1 including Applicable Safety Analyses, LCO [limiting condition for operation], and Applicability 2.b, SR 3.3.1.1.3, SR 3.3.1.1.11, SR 3.3.1.1.13, and SR 3.3.1.1.17.

The proposed changes are based on two approved methodologies - NEDO-32339-A, Supplement 4, Revision 1, April 1998 (Reference 3), and NEDC-32339P-A, Supplement 3, Revision 1, "Reactor Stability Long Term Solution EIA: Flow Mapping Methodology," dated April 1998 (Reference 4), which is used to complete initial flow alignment process to establish the initial relationship between core flow and drive flow, then only periodic adjustment to the digital component of the FCTR is required. Furthermore, the use of digital components in the EIA FCTR card also allows the incorporation of self-test features and more frequent internal checks, and the digital components used in the EIA FCTR card are highly reliable. The staff has reviewed the proposed change and finds it acceptable.

2.3 TS 3.3.1.3, Period Based Detection System (PBDS)

This proposed change adds a new TS 3.3.1.3, Period Based Detection System (PBDS), and its associated Bases B 3.3.1.3. The Bases includes the sections "Background," "Applicable Safety Analyses," "LCO," "Applicability," "Actions," "Surveillance Requirements," and "References." The licensee proposed removing the phrases "simulated thermal power - high" on pages B 3.3-43, -45, -46 and -47, and "which reduce recirculation flow" on page B 3.3-47.

The PBDS is a required feature of the EIA solution. The PBDS uses the neutron flux oscillation period confirmation process of the Period Based Algorithm (PBA) described in NEDO-31960-A and NEDO-31960-A, Supplement 1 (Reference 5). Verification that the High-High Decay Ratio (Hi-Hi DR) alarm is valid may be performed in the control room prior to the manual reactor scram using another output from the PBDS card, which generated the Hi-Hi DR alarm. The requirements of the EIA solution PBDS specification include immediate manual reactor scram without delay upon receipt of any valid PBDS channel Hi-Hi DR alarm while operating in regions of the power and flow operating domain potentially susceptible to neutronic/thermal hydraulic instability.

The PBA-EIA (Reference 6) has no safety function and is not credited during any design-basis accident or transient analysis. However, during operation in regions of the operating domain potentially susceptible to instability under any operating conditions, the PBDS provides an indication that conditions consistent with a significant degradation in the stability performance of the reactor has occurred and the potential for imminent onset of neutron/thermal hydraulic instability may exist.

The staff has reviewed the proposed change and found it acceptable because the addition of this TS and its associated Bases is based on an approved methodology stated in the BWROG reactor stability long-term solution EIA (Reference 3) and deletion of the phrases listed would meet the River Bend's plant-specific application and match the actual design features of the APRMs.

2.4 TS 3.4.1, Recirculation Loops Operating

These proposed changes include: (1) elimination of LCO 3.4.1.A.2, LCO 3.4.1.B.2 and 3, Actions C, D, E, F and G, SR 3.4.1.2 and their associated Bases; and (2) addition of NUREG-1434 Actions A and B as Actions C and D.

The staff has reviewed the proposed changes and found them acceptable. An approved methodology (Reference 3) and NUREG-1434 was used to implement the BWROG reactor stability EIA long term solution for RBS. Also, the elimination of the limits on power and flow conditions of the current TS 3.4.1 and the guidance of the BWROG Interim Corrective Actions (ICAs) is justified by the licensee based on several attributes of the EIA long term solution. Operation in the region of the power and flow operating domain most susceptible to neutronic/thermal hydraulic instability is automatically excluded from the licensed operating domain (Exclusion Region). Operation in the region of the power and flow operating domain potentially susceptible to neutronic/thermal hydraulic instability in the absence of stability control (Restricted Region) requires implementation of stability control prior to entry and verification that at least one PBDS card is operable. These regions are established using the NRC accepted EIA methodology and reflect the plant specific design of River Bend.

2.5 TS 5.6.5, Core Operating Limits Report (COLR)

These proposed changes include the addition of items 4, 5, and 6 to TS 5.6.5.a, and item 3 to TS 5.6.5.b.

The staff has reviewed the proposed changes and finds them acceptable. Proposed TSs LCO 3.2.4, Fraction of Core Boiling Boundary (FCBB), LCO 3.3.1.1, RPS [Reactor Protection System] Instrumentation, and LCO 3.3.1.3, Period Based Detection System (PBDS), are cycle-specific parameters supported by an approved methodology listed in TS 5.6.5.b (NEDO-32339P-A, "Reactor Stability Long-term Solution: Enhanced Option I-A," including Supplements 1 through 4 (April 1998)). Reference to this methodology was modified on TS page 5.0-19 to include all of its approved revisions.

On the basis of its evaluation, the staff concludes that the proposed changes to the TSs are acceptable for River Bend since the changes are analyzed based on the NRC-approved methodologies and the guidance stated in NUREG-1434. Additionally, the staff may elect to perform an on-site review of this plant-specific implementation of the BWROG reactor stability long-term EIA solution for River Bend.

3.0 STAFF CONCLUSION

The staff has reviewed the request by the licensee to revise the River Bend TSs to implement its BWROG EIA reactor stability long-term solution. On the basis of its review, the staff concludes that these revisions are acceptable since approved methodologies were used. In addition, the reference to the approved methodology listed in TS 5.6.5.b was modified to include its approved revisions.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (63 FR 64112, November 18, 1998). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

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Date: May 5, 1999

REFERENCES

1. Letter (RBG-44593) from R. K. Edington to NRC, "River Bend Station - Unit 1, Docket No. 50-458, License No. NPF-47, License Amendment Request (LAR) 98-02: "Stability," October 8, 1998.

2. NEDO-32339, Revision 1, Licensing Topical Report, "Reactor Stability Long-Term Solution: Enhanced Option I-A," GE Nuclear Energy, December 1996.

3. NEDO-32339-A, Supplement 4, Revision 1, "Reactor Stability Long Term Solution: Enhanced Option I-A Generic TSs," April 1998.

4. NEDC-32339P-A, Supplement 3, Revision 1, "Reactor Stability Long Term Solution EIA: Flow Mapping Methodology," GE Nuclear Energy, April 1998.

5. NEDO-31960-A, Supplement 1, "BWR Owners' Group Long Term Stability Solutions Licensing Methodology," November 1995.

6. NEDC-32339P-A, Supplement 2, Revision 1, "Reactor Stability Long-Term Solution: Enhanced Option I-A: Solution Design," April 1998.