Entergy

Entergy Operations, Inc.
River Bend Station
5485 U.S. Highway 61
P.O. Box 220
St. Francisville, LA 70775
Tel 225 381 4374
Fax 225 381 [illegible]

Randall K. Edington
Vice President Operations

August 24, 2000

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Subject: River Bend Station
Docket No. 50-458
License No. NPF-47
Submittal of Proposed Technical Specifications and Additional Information Related to License Amendment Request (LAR) 99-15, Changes to Technical Specifications for Power Uprate of River Bend Station

File Nos.: G9.5, G9.42

Reference:
1) Entergy Operations, Inc. (EOI) Letter to NRC, RBG-45077, dated July 30, 1999
2) Entergy Operations, Inc. (EOI) Letter to NRC, RBG-45293, dated April 3, 2000
3) Entergy Operations, Inc. (EOI) Letter to NRC, RBG-45337, dated May 9, 2000
4) Entergy Operations, Inc. (EOI) Letter to NRC, RBG-45428, dated July 18, 2000

RBEXEC-00-027
RBF1-00-0174
RBG-45471

Ladies and Gentlemen:

In the reference (1) letter, EOI requested a license amendment to NPF-47 and Appendix A – Technical Specifications, of the River Bend Station (RBS). This request is to extend operation of RBS from its current licensed power level of 2894 megawatts thermal (MWt) by five percent to an uprated power level of 3039 MWt. In support of this request, EOI is submitting the proposed technical specifications associated with this license amendment. In addition, this letter contains corrections to NEDC-32778P, *Safety Analysis Report for River Bend 5% Thermal Power Uprate,* which was included as Enclosure 7 to the reference (1) letter.

APO1

Enclosure 1 is an oath and affirmation executed in accordance with 10 CFR 50.30(b). Reference (2) and (4) letters provided additional information related to License Amendment Request (LAR) 99-15. Since an oath and affirmation was not provided for this additional information, it has also been included in the Enclosure 1 affirmation.

Enclosure 2 is the proposed technical specifications associated with LAR 99-15. These technical specifications were revised in accordance with the proposed changes contained in the reference (1) letter with the following exceptions:

a. Figure 3.4.11-1 (Minimum Temperature Required vs. RCS Pressure) will be replaced by the Minimum RPV Temperature vs. Reactor Vessel Pressure figure for 32 EFPY only (i.e., the 14 EFPY figure is not included). In addition, the marked-up technical specification pages submitted by the reference (1) letter indicated that this figure was GE proprietary information. Although the GE document (NEDC-32778P) that produced this figure contains proprietary information, the figure itself is not proprietary.

b. Technical Specification Surveillance Requirement 3.3.1.1.2, Surveillance Requirement 3.3.1.1.3, and Table 3.3.1.1-1 item 2.b have been revised to allow for implementation of LAR 99-15 while on-line. These temporary implementation changes are required to maintain compliance with technical specifications during the transition to the uprated power level. A detailed evaluation of these changes is contained in the reference (3) letter. Addition of these temporary implementation changes to page 3.3-3 of technical specifications caused information to shift on subsequent pages 3.3-4, 3.3-5, and 3.3-6. These pages are included in this amendment even though they do not contain any new information.

c. A note has been added to specification 3.3.1.1, Table 1 Item 3; Reactor Vessel Steam Dome Pressure – High, requiring the current pressure limit to be maintained until the implementation of the pressure portion of power uprate. This delay is consistent with justification provided in EOI submittal discussing the phased implementation of uprate, Reference 3.

Enclosure 3 contains corrections to NEDC-32778P, *Safety Analysis Report for River Bend 5% Thermal Power Uprate,* which was included as Enclosure 7 to the reference (1) letter. These corrections do not impact the report's conclusion that power uprate to 105% of original rated power does not involve a Significant Hazards Consideration. Note that this report along with the corresponding changes contain proprietary information that should be withheld from public disclosure in accordance with 10 CFR 2.790(a)(4).

An affidavit executed by GE supporting a request for proprietary treatment of this report was provided in Enclosure 6 to the reference (1) letter.

Enclosure 4 contains additional information developed since the submittal of Reference 4. This includes updates to responses provided to questions in previous RAI's and data presented at the August 15, 2000 meeting between EOI and NRC in Washington.

There are no new commitments in this letter. If you have any questions about this license amendment request, please contact Barry Burmeister at (225) 381-4148.

Sincerely,

Randall K. Edington

Enclosures
RKE/RJK/bmb

cc:

U. S. Nuclear Regulatory Commission
Region IV
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011

NRC Senior Resident Inspector
P. O. Box 1050
St. Francisville, LA 70775

Mr. David Jaffe
U.S. Nuclear Regulatory Commission
M/S OWFN 04D03
Washington, DC 20555

Mr. Prosanta Chowdhury
Program Manager - Surveillance Division
Louisiana Department of Environmental Quality
Office of Radiological Emergency Planning & Response
P. O. Box 82215
Baton Rouge, LA 70884-2215

Mr. Jefferey F. Harold
U.S. Nuclear Regulatory Commission
M/S OWFN 07D01
Washington, DC 20555

ENCLOSURE 1

BEFORE THE

UNITED STATES NUCLEAR REGULATORY COMMISSION

LICENSE NO. NPF-47

DOCKET NO. 50-458

IN THE MATTER OF

ENTERGY GULF STATES, INC.

ENTERGY OPERATIONS, INC.

AFFIRMATION

I, Randall K. Edington, state that I am Vice President – River Bend Station, Entergy Operations, Inc. (EOI), that on behalf of EOI, I am authorized to sign and file with the U. S. Nuclear Regulatory Commission, this River Bend Station License Amendment Request (LAR) 1999-15, consisting of proposed changes to the River Bend Station Technical Specifications, that I signed this letter as Vice President - River Bend Station, for Entergy Operations, Inc.; and that the statements made and the matters set forth herein are true and correct to the best of my knowledge, information, and belief. In addition, the statements made and the matters set forth in Entergy Operations, Inc. (EOI) letter to NRC, RBG-45293, dated April 3, 2000 and Entergy Operations, Inc. (EOI) letter to NRC, RBG-45428, dated July 18, 2000 which contained additional information related to LAR 1999-15 are true and correct to the best of my knowledge, information, and belief.

Randall K. Edington

Randall K. Edington

STATE OF LOUISISANA
PARISH OF WEST FELICIANA

SUBSCRIBED AND SWORN TO before me, a Notary Public, commissioned in the Parish and State above named, this 24th day of August, 2000.

(SEAL)

Claudia F. Hurst

Claudia F. Hurst
Notary Public

ENCLOSURE 2

ENTERGY OPERATIONS, INC.
RIVER BEND STATION (RBS)

PROPOSED TECHNICAL SPECIFICATIONS
ASSOCIATED WITH LAR 99-15

(See attached.)

(3) EOI, pursuant to the Act and 10 CFR Part 70, to receive, possess and to use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(4) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) EOI, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Comission now or hereafter in effect; and is subject to the additional conditions specified or incorpor ated below:

(1) Maximum Power Level

EOI is authorized to operate the facility at reactor core power levels not in excess of 3039 megawatts thermal (100% rated power) in accordance with the conditions specified herein. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 70 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.

1.1 Definitions (continued)

MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)	The MFLPD shall be the largest value of the fraction of limiting power density in the core. The fraction of limiting power density shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle type.
MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each class of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE — OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3039 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

(continued)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-1 (continued)

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of SR 3.0.4.

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after ≥ 23.8% RTP AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 23.8% RTP to ≥ 23.8% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the extension allowed by SR 3.0.2.

(continued)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-2 (continued)

"Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 23.8% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 23.8% RTP.

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
------------------NOTE------------------ Not required to be performed until 12 hours after ≥ 23.8% RTP. -- Perform channel adjustment.	7 days

The interval continues whether or not the unit operation is < 23.8% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 23.8% RTP, this Note allows 12 hours after power reaches ≥ 23.8% RTP to perform the Surveillance. The Surveillance is still considered to be within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day interval (plus the extension allowed by SR 3.0.2), but operation was < 23.8% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 23.8% RTP.

(continued)

1.4 Frequency

EXAMPLES

EXAMPLE 1.4-3 (continued)

Once the unit reaches 23.8% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
------------------NOTE------------------ Only required to be met in MODE 1. -- Verify leakage rates are within limits.	24 hours

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour (plus the extension allowed by SR 3.0.2) interval, but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23.8% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.12 for two recirculation loop operation or ≥ 1.13 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

2.2.2 Within 2 hours:

2.2.2.1 Restore compliance with all SLs; and

2.2.2.2 Insert all insertable control rods.

2.2.3 Within 24 hours, notify the plant manager and the corporate executive responsible for overall plant nuclear safety.

(continued)

Table 3.1.4-1
Control Rod Scram Times

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

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to notch position 13. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

NOTCH POSITION	SCRAM TIMES[a][b] (seconds)	
	REACTOR STEAM DOME PRESSURE[c] 950 psig	REACTOR STEAM DOME PRESSURE[c] 1059 psig
43	0.30	0.31
29	0.78	0.84
13	1.40	1.53

(a) Maximum scram time from fully withdrawn position, based on de-energization of scram pilot valve solenoids as time zero.

(b) Scram times as a function of reactor steam dome pressure when < 950 psig are within established limits.

(c) For intermediate reactor steam dome pressures, the scram time criteria are determined by linear interpolation.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Two or more control rod scram accumulators inoperable with reactor steam dome pressure ≥ 600 psig.	B.1 Restore charging water header pressure to ≥ 1540 psig.	20 minutes from discovery of Condition B concurrent with charging water header pressure < 1540 psig
	<u>AND</u>	
	B.2.1 --------NOTE--------- Only applicable if the associated control rod scram time was within the limits of Table 3.1.4-1 during the last scram time Surveillance. --------------------- Declare the associated control rod scram time "slow."	1 hour
	<u>OR</u>	
	B.2.2 Declare the associated control rod inoperable.	1 hour
C. One or more control rod scram accumulators inoperable with reactor steam dome pressure < 600 psig.	C.1 Verify all control rods associated with inoperable accumulators are fully inserted.	Immediately upon discovery of charging water header pressure < 1540 psig
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.2 Declare the associated control rod inoperable.	1 hour
D. Required Action and associated Completion Time of Required Action B.1 or C.1 not met.	D.1 --------NOTE--------- Not applicable if all inoperable control rod scram accumulators are associated with fully inserted control rods. --------------------- Place the reactor mode switch in the shutdown position.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.5.1 Verify each control rod scram accumulator pressure is ≥ 1540 psig.	7 days

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Standby Liquid Control (SLC) System

LCO 3.1.7 Two SLC subsystems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (C)(E) < 570.	A.1 Restore (C)(E) ≥ 570.	72 hours AND 10 days from discovery of failure to meet the LCO
B. One SLC subsystem inoperable for reasons other than Condition A.	B.1 Restore SLC subsystem to OPERABLE status.	7 days AND 10 days from discovery of failure to meet the LCO
C. Two SLC subsystems inoperable for reasons other than Condition A.	C.1 Restore one SLC subsystem to OPERABLE status.	8 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	-------------------NOTE------------------- The minimum required available solution volume is determined by the performance of SR 3.1.7.5. -- Verify available volume of sodium pentaborate solution is greater than or equal to the minimum required available solution volume.	24 hours
SR 3.1.7.2	Verify temperature of sodium pentaborate solution is $\geq$ 45°F.	24 hours
SR 3.1.7.3	-------------------NOTE------------------- Sodium Pentaborate Concentration (C), in weight percent, is determined by the performance of SR 3.1.7.5. Boron-10 enrichment (E), in atom percent, is determined by the performance of SR 3.1.7.9. -- Verify that the SLC System satisfies the following equation: $(C)(E) \geq 570$	31 days
SR 3.1.7.4	Verify continuity of explosive charge.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.5	Verify the available weight of Boron-10 is ≥ 143 lbs. and the percent weight concentration of sodium pentaborate in solution is ≤ 9.5% by weight, and determine the minimum required available solution volume.	31 days <u>AND</u> Once within 24 hours after water or boron is added to solution <u>AND</u> Once within 24 hours after solution temperature is restored to ≥ 45°F
SR 3.1.7.6	Verify each SLC subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is in the correct position, or can be aligned to the correct position.	31 days
SR 3.1.7.7	Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1250 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	18 months on a STAGGERED TEST BASIS

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 23.8% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any APLHGR not within limits.	A.1 Restore APLHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 23.8% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify all APLHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 23.8% RTP AND 24 hours thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 23.8% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any MCPR not within limits.	A.1 Restore MCPR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 23.8% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.2.1 Verify all MCPRs are greater than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 23.8% RTP __AND__ 24 hours thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 23.8% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any LHGR not within limits.	A.1 Restore LHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 23.8% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.3.1 Verify all LHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 23.8% RTP __AND__ 24 hours thereafter

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1 Reduce THERMAL POWER to < 40% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1 Reduce THERMAL POWER to < 23.8% RTP.	4 hours
G. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1 Be in MODE 2.	6 hours
H. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	H.1 Be in MODE 3.	12 hours
I. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	I.1 Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately

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 ≥ 23.8% RTP. -- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power ≤ 2% RTP[a] .	 7 days
SR 3.3.1.1.3	Adjust the flow control trip reference card to conform to reactor flow[b] .	Once within 7 days after reaching equilibrium conditions following refueling outage.

(a) For a period of 30 days beginning with uprate COLR implementation and corresponding plant monitoring computer data bank changes the difference between the average power range monitor (APRM) channels and the calculated power must be within -2% RTP to +7% RTP.
(b) Within 30 days of uprate COLR implementation and corresponding plant monitoring computer data bank changes the flow control trip reference card will be verified to conform to reactor flow in accordance with the uprated COLR.

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
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
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.	2000 MWD/T average core exposure
SR 3.3.1.1.9	Perform CHANNEL FUNCTIONAL TEST.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.10	Calibrate the trip units.	92 days
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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.1.15	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months
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)(c)

(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.

(C) Within 30 days of uprate COLR implementation and corresponding plant monitoring computer data bank changes the flow control trip reference card will be verified to conform to reactor flow in accordance with the uprated COLR.

Table 3.3.1.1-1 (page 2 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
2. Average Power Range Monitors (continued)					
c. Fixed Neutron Flux — High	1	3	G	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.15 SR 3.3.1.1.18	≤ 120% RTP
d. Inop	1,2	3	H	SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.15	NA
3. Reactor Vessel Steam Dome Pressure — High	1,2	2	H	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.18	≤ 1109.7 psig[a]
4. Reactor Vessel Water Level — Low, Level 3	1,2	2	H	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.18	≥ 8.7 inches
5. Reactor Vessel Water Level — High, Level 8	≥ 23.8% RTP	2	F	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.18	≤ 52.1 inches
6. Main Steam Isolation Valve — Closure	1	8	G	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.18	≤ 12% closed
7. Drywell Pressure — High	1,2	2	H	SR 3.3.1.1.1 SR 3.3.1.1.9 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 1.88 psid

(continued)

(a) ALLOWABLE VALUE to remain as ≤ 1079.7 psi until pressure increase portion of Power Uprate.

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.4.2.3	Calibrate the trip units.	92 days
SR 3.3.4.2.4	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. Reactor Vessel Water Level — Low Low, Level 2: ≥ -47 inches; and b. Reactor Steam Dome Pressure — High: ≤ 1165 psig.	18 months
SR 3.3.4.2.5	Perform LOGIC SYSTEM FUNCTIONAL TEST, including breaker actuation.	18 months

Table 3.3.6.1-1 (page 1 of 5)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level — Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ -147 inches
b. Main Steam Line Pressure — Low	1	2	E	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 837 psig
c. Main Steam Line Flow — High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 190 psid, Line A ≤ 194 psid, Line B ≤ 194 psid, Line C ≤ 194 psid, Line D
d. Condenser Vacuum — Low	1,2[(a)], 3[(a)]	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 7.6 inches Hg vacuum
e. Main Steam Tunnel Temperature — High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 148.5°F
f. Main Steam Tunnel Area Temperature — High (El. 95ft)	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 145.3°F
g. Main Steam Tunnel Area Temperature — High (El. 114ft)	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 145.3°F
h. Main Steam Line Turbine Shield Wall Temperature-High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 111.3°F

(continued)

(a) With any turbine stop valve not closed.

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTE-----------------------------------
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 LLS or relief initiation capability, as applicable.

SURVEILLANCE		FREQUENCY
SR 3.3.6.4.1	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.6.4.2	Calibrate the trip unit.	92 days
SR 3.3.6.4.3	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. Relief Function Low: 1133 ± 15 psig Medium: 1143 ± 15 psig High: 1153 ± 15 psig b. LLS Function Low open: 1063 ± 15 psig close: 956 ± 15 psig Medium open: 1103 ± 15 psig close: 966 ± 15 psig High open: 1143 ± 15 psig close: 976 ± 15 psig	18 months
SR 3.3.6.4.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 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 ≤ 79% 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 > 79% RTP during single loop operation.	B.1 Reduce THERMAL POWER to ≤ 79% RTP.	1 hour

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	-------------------NOTES------------------- 1. Not required to be performed until 4 hours after associated recirculation loop is in operation. 2. Not required to be performed until 24 hours after $>$ 23.8% RTP. --- Verify at least two of the following criteria (a, b, and c) are satisfied for each operating recirculation loop: a. Recirculation loop drive flow versus flow control valve position differs by $\leq$ 10% from established patterns. b. Recirculation loop drive flow versus total core flow differs by $\leq$ 10% from established patterns. c. Each jet pump diffuser to lower plenum differential pressure differs by $\leq$ 20% from established patterns, or each jet pump flow differs by $\leq$ 10% from established patterns.	24 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 Safety/Relief Valves (S/RVs)

LCO 3.4.4 The safety function of five S/RVs shall be OPERABLE.

AND

The relief function of four additional S/RVs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required S/RVs inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.4.1 Verify the safety function lift setpoints of the required S/RVs are as follows: Number of S/RVs — Setpoint (psig) 7 — 1195 +/- 36 5 — 1205 +/- 36 4 — 1210 +/- 36	In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	-------------------NOTE------------------- Only required to be performed in MODES 1 and 2. -- Verify equivalent leakage of each RCS PIV is $\leq$ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at an RCS pressure $\geq$ 1040 psig and $\leq$ 1070 psig.	In accordance with Inservice Testing Program

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ---------NOTE--------- Required Action C.2 shall be completed if this Condition is entered. --------------------- Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1 Initiate action to restore parameter(s) to within limits. <u>AND</u> C.2 Determine RCS is acceptable for operation.	Immediately Prior to entering MODE 2 or 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	------------------NOTE------------------- Only required to be performed during RCS heatup and cooldown operations, and RCS inservice leak and hydrostatic testing. --- Verify: a. RCS pressure and RCS temperature are within the limits of Figure 3.4.11-1, and b. RCS heatup and cooldown rates are ≤ 100°F in any one hour period for core not critical and core critical limits. c. RCS heatup and cooldown rates are <_20°F in any one hour period for inservice leak and hydrostatic testing limits	30 minutes

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.4.11.2	-------------------NOTE------------------- Only required to be met during control rod withdrawal for the purpose of achieving criticality. -- Verify RCS pressure and RCS temperature are within the core critical limits specified in Figure 3.4.11-1.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
SR 3.4.11.3	-------------------NOTE------------------- Only required to be met in MODES 1, 2, 3, and 4 with reactor steam dome pressure ≥ 25 psig during recirculation pump start. -- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ≤ 100°F.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.11.4	-------------------NOTE------------------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump start. -- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is ≤ 50°F.	Once within 15 minutes prior to each startup of a recirculation pump

(continued)

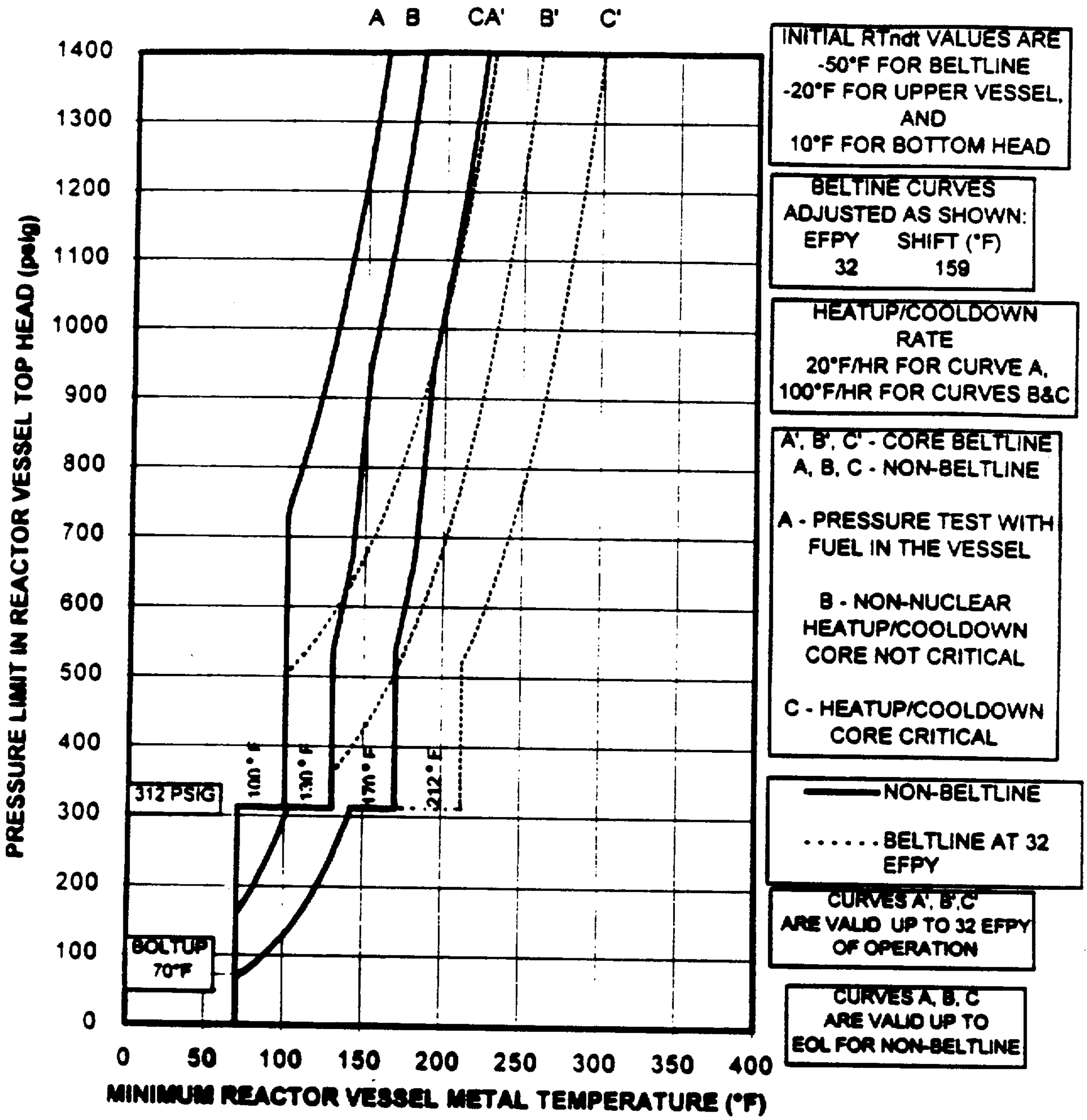

Figure 3.4.11-1 (page 1 of 1)
Minimum Temperature Required vs. RCS Pressure

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Reactor Steam Dome Pressure

LCO 3.4.12 The reactor steam dome pressure shall be ≤ 1075 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1 Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify reactor steam dome pressure is ≤ 1075 psig.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.3.2	Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.3.3	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with RCIC steam supply pressure ≤ 1075 psig and ≥ 920 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	92 days
SR 3.5.3.4	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify, with RCIC steam supply pressure ≤ 165 psig and ≥ 150 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	18 months

(continued)

3.7 PLANT SYSTEMS

3.7.5 Main Turbine Bypass System

LCO 3.7.5 The Main Turbine Bypass System shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 23.8 RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Main Turbine Bypass System inoperable.	A.1 Restore Main Turbine Bypass System to OPERABLE status.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 23.8% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify one complete cycle of each main turbine bypass valve.	31 days
SR 3.7.5.2	Perform a system functional test.	18 months
SR 3.7.5.3	Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	18 months

3.9 REFUELING OPERATIONS

3.9.5 Control Rod OPERABILITY — Refueling

LCO 3.9.5 Each withdrawn control rod shall be OPERABLE.

APPLICABILITY: MODE 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more withdrawn control rods inoperable.	A.1 Initiate action to fully insert inoperable withdrawn control rods.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	-------------------NOTE------------------- Not required to be performed until 7 days after the control rod is withdrawn. -- Insert each withdrawn control rod at least one notch.	7 days
SR 3.9.5.2	Verify each withdrawn control rod scram accumulator pressure is ≥ 1540 psig.	7 days

3.10 SPECIAL OPERATIONS

3.10.8 SHUTDOWN MARGIN (SDM) Test — Refueling

LCO 3.10.8 The reactor mode switch position specified in Table 1.1-1 for MODE 5 may be changed to include the startup/hot standby position, and operation considered not to be in MODE 2, to allow SDM testing, provided the following requirements are met:

a. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," MODE 2 requirements for Function 2.a and 2.d of Table 3.3.1.1-1;

b. 1. LCO 3.3.2.1, "Control Rod Block Instrumentation," MODE 2 requirements for Function 1.b of Table 3.3.2.1-1,

<u>OR</u>

2. Conformance to the approved control rod sequence for the SDM test is verified by a second licensed operator or other qualified member of the technical staff;

c. Each withdrawn control rod shall be coupled to the associated CRD;

d. All control rod withdrawals during out of sequence control rod moves shall be made in single notch withdrawal mode;

e. No other CORE ALTERATIONS are in progress; and

f. CRD charging water header pressure ≥ 1540 psig.

APPLICABILITY: MODE 5 with the reactor mode switch in startup/hot standby position.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.10.8.5	Verify each withdrawn control rod does not go to the withdrawn overtravel position.	Each time the control rod is withdrawn to "full out" position <u>AND</u> Prior to satisfying LCO 3.10.8.c requirement after work on control rod or CRD System that could affect coupling
SR 3.10.8.6	Verify CRD charging water header pressure ≥ 1540 psig.	7 days

ENCLOSURE 4

ENTERGY OPERATIONS, INC.
RIVER BEND STATION (RBS)

ADDITIONAL INFORMATION

(See attached.)

Simulator Observations

Included in Reference 4, Question 7 (page 25 of 40) an initial response to an NRC question concerning Simulator Observations provided the following:

Question 7: Provide examples of operator actions that are particularly sensitive to the proposed increase in power level and discuss how the power uprate will effect operator reliability or performance. Identify all operator actions that will have their response times changed because of the power uprate. Specify the expected response times before the power uprate and the new (reduced/increased) response times. Discuss why any reduced operator response times are needed. Discuss whether any reduction in time available for operator actions, due to the power uprate, will significantly affect the operator's ability to compete the required manual actions in the times allowed. Discuss results of simulator observations regarding operator response times for operator actions that are potentially sensitive to power uprate.

(Initial) Response 7:

Operations Simulator Observations

Initial simulator observations comparing simulator response at current 100% power to the simulator response at 105% (Uprated Power) indicate no appreciable time frame or parameter differences. The observations were based on simulator response with no operator actions. Confirmation of this information is expected by August 11, 2000.

(Completed) Response:

Operations Simulator Observations

Two simulator observations were performed: An MSIV closure with an ATWS at 100% reactor power (2894 MWt) and an MSIV closure with an ATWS at 105% reactor power (3039 MWt). This event was selected because it is expected to be the most challenging event associated with operator response/actions and plant/containment impact.

The Operations Training staff performed simulator observations for the purpose of comparing simulator response at current 100% power to the expected simulator response at 105% (Uprated Power). The observations were performed for a MSIV closure with an ATWS at equilibrium xenon conditions and no operator actions. Parameters monitored for comparison during the events included reactor power, RPV pressure, suppression pool temperature, and suppression pool level.

Based on the observed response of the simulator, no appreciable time frame or parameter differences were noted. The observations were strictly based on simulator response with no operator actions.

In addition to the modifications to the plant the simulator model changes to support the power uprate conditions will be incorporated early in September, 2000. Operator training under uprate conditions is included after the simulator is upgraded. In addition to simulator training, operators receive classroom training, and briefings regarding uprate operations prior to performing MCR duties under uprate conditions.

Operational Radiation Levels

During discussions with the NRC staff additional information was requested concerning the expected normal radiation levels and effects on operation.

Response

The discussions in Section 8.4 and 8.5 of the GE report (NEDC-32778P), describe the expected changes to plant radiation levels during normal operation. The RWCU system, discussed in Section 3.10 of the GE report, will continue to perform its function at the uprated power level with a slightly higher feedwater iron input. The radiation levels for most areas in the plant are expected to increase by no more than the percentage of the power increase, which is within original design margins.

For areas affected by activated corrosion products the increase is expected to be proportional to the square of the power level increase. These are systems connected to the reactor vessel which are inside the drywell (reactor recirculation system), which is inaccessible during power operations, or within locked high radiation areas (RWCU system).

Accident Dose Information

During a meeting on August 15, 2000, the NRC and EOI discussed radiological results for events under the uprated conditions. At this meeting the NRC staff requested further information concerning the doses at various locations and conditions in the plant. The information below is in response to this request.

Response

Table 1 - LOCA Input Assumptions – Constant

Parameter	Value
Primary Containment Leakage Rate	0.26%/day
Leakage bypassing secondary containment	170,000 cc/hr
ESF Leakage	60 gph=1 gpm
Drywell Bypass $A/\sqrt{K}$	1.0 ft^2
Suppression Pool Iodine Decontamination Factor • Elemental • Organic • Particulate	 10 1 1
Containment Volume Drywell Volume	1,191,590 ft^3 2.36×10^5 ft^3
Annulus free volume	357,400 ft^3
Auxiliary Building Free Volume	1.16×10^6 ft^3
Fuel Building Free Volume	7.42×10^5 ft^3
Annulus recirculation system parameters • Flow rate • Mixing efficiency • Exhaust flow	 47,250 cfm (min.) 50% 2,500 cfm
Fuel Building Design Exhaust Flow Rate	10,000 cfm
Building initial vacuum Annulus Auxiliary Building	 ≤3.0 in W.G. ≤0.0 in W.G.
SGT Building Design Exhaust Flow Aux. Building Exhaust Flow	12,500 cfm 10,000 cfm
Minimum SGTS exhaust flow	1,500 cfm
Adsorption and filtration efficiencies (%) • Organic iodine • Elemental iodine • Particulate iodine	 99 99 99
Control Room (CR) Volume	240,702 ft^3
CR Ventilation Parameters • Ingress/egress • Intake (filtered) • Discharge • Recirculation (filtered)	 10 cfm 1947.6 cfm 1947.6+10=1957.6 cfm 1947.6 cfm
CR filter efficiency (Intake/Recirc.)	99 %
CR filter actuation time	66 sec

Additional Information

Control Room χ/Q (Local Intake) – Containment Release 0-8 hrs. 8-24 hr. 24-96 hr. 96-720 hr.	 1.62×10^{-3} sec/m^3 1.20×10^{-3} sec/m^3 4.05×10^{-4} sec/m^3 6.48×10^{-5} sec/m^3
Limiting Control Room χ/Q values for secondary containment bypass ($^{sec}/_{m^3}$) 0-8 hr 8-24 hr 1-4 days 4-30 days	 4.04×10^{-3} 3.03×10^{-3} 9.29×10^{-4} 1.62×10^{-4}
Offsite Dispersion Factors (χ/Q) – Containment Release EAB 0-2 hr. LPZ 0-8 hr. 8-24 hr. 24-96 hr. 96-720 hr.	 8.58×10^{-4} sec/m^3 1.13×10^{-4} sec/m^3 7.89×10^{-5} sec/m^3 3.65×10^{-5} sec/m^3 1.21×10^{-5} sec/m^3
Limiting 0-2 hr EAB χ/Q value for secondary containment bypass ($^{sec}/_{m^3}$)	9.01×10^{-4}
Limiting LPZ χ/Q values for secondary containment bypass ($^{sec}/_{m^3}$) 0-8 hr 8-24 hr 1-4 days 4-30 days	 1.14×10^{-4} 8.00×10^{-5} 3.71×10^{-5} 1.23×10^{-5}
Breathing Rate (offsite) 0-8 hrs. 8-24 hrs. 24-720 hrs.	 3.47×10^{-4} m^3/sec 1.75×10^{-4} m^3/sec 2.32×10^{-4} m^3/sec
Breathing Rate (Control Room)	3.47×10^{-4} m^3/sec
Dose Conversion Factors	ICRP 30
Control Room Occupancy Factor • 0 – 24 hours • 1 – 4 days • 4 – 30 days	 1.0 0.6 0.4
Suppression Pool Peak Temperature	< 185F
Information Notice 91-56 Term • Flow Rate • Start Time • Duration	 50 gpm 24 hrs. 30 min.
Suppression Pool Volume (Calculated Minimum/Assumed in Calculation)	123,180/120,000 ft^3
Iodine Chemical Fractions • Elemental • Organic • Particulate	 91% 4% 5%
Airborne Fractions • Noble Gases • Halogens	 100% 25%
Annulus bypass leakage • To Fuel Building • To Auxiliary Building Total annulus bypass leakage	 6,750 cc/hr 6,750 cc/hr 13500 cc/hr

Table 2 - LOCA Input Assumptions – Variable

Parameter	**Amendment 111 LOCA Dose USQ**	**LAR 99-15 Power Uprate**
Power level	3039 MWt	3100 MWt
Positive Pressure Period	195.5 sec.	700 sec.

LOCA Computer Files' Descriptions

- **CONTAIN** - This file determines the dose consequences of air leakage from the primary and secondary containment buildings.
- **PVLCS** – This file determines the dose consequences due to the secondary containment bypass leakage term. This leakage is assumed to be released directly to the environment.
- **LIQUID** – This file determines the dose consequences of liquid leakage of ESF systems into the auxiliary building.
- **IN91-56** – This file models the gross failure of a passive component outside of secondary containment. This file is not impacted by secondary containment assumptions. Note that the power level previously assumed for this term was 3100 MWt so the impact of Power Uprate was already considered in the Amendment 111 submittal.

Table 3 – Off-site LOCA Dose Results

Location	**Dose**	**Contributor**	**Amendment 111 LOCA Dose USQ**	**LAR 99-15 Power Uprate**
EAB	Whole Body	CONTAIN	4.127E+00	4.848E+00
		PVLCS	4.797E-01	4.893E-01
		LIQUID	2.398E-02	8.546E-02
		IN91-56	0.000E+00	0.000E+00
		Total	***4.63***	***5.42***
	Thyroid	CONTAIN	1.264E+01	4.014E+01
		PVLCS	1.918E+01	1.956E+01
		LIQUID	6.016E+00	2.092E+01
		IN91-56	0.000E+00	0.000E+00
		Total	***37.84***	***80.62***
LPZ	Whole Body	CONTAIN	2.581E+00	2.708E+00
		PVLCS	1.834E-01	1.871E-01
		LIQUID	5.913E-03	1.406E-02
		IN91-56	4.260E-02	4.260E-02
		Total	***2.81***	***2.95***
	Thyroid	CONTAIN	8.999E+00	1.276E+01
		PVLCS	3.730E+01	3.804E+01
		LIQUID	4.826E+00	6.867E+00
		IN91-56	6.394E+01	6.394E+01
		Total	***115.1***	***121.6***

Table 4 – Main Control Room LOCA Dose Results

Dose	Contributor	Amendment 111 LOCA Dose USQ	LAR 99-15 Power Uprate
Whole Body	CONTAIN	3.595E-01	4.003E-01
	PVLCS	5.719E-02	5.834E-02
	LIQUID	6.302E-05	9.845E-05
	IN91-56	1.038E-05	1.038E-05
	Total	***0.42***	***0.46***
Skin	CONTAIN	7.697E+00	8.380E+00
	PVLCS	1.124E+00	1.147E+00
	LIQUID	4.703E-04	7.302E-04
	IN91-56	1.309E-04	1.309E-04
	Total	***8.82***	***9.53***
Thyroid	CONTAIN	2.453E+00	2.872E+00
	PVLCS	3.006E+00	3.067E+00
	LIQUID	4.548E-01	6.645E-01
	IN91-56	4.168E-01	4.168E-01
	Total	***6.33***	***7.02***

Control Rod Drop Accident

Table 5 – CRDA Input Assumptions

Parameter	Current USAR Analysis	Power Uprate Analysis
Power Level	3039 MWt	3100 MWt
Rods Damaged (GE 8x8)	770	850
Radial Peaking Factor	1.5	1.65
Release Fractions • Noble Gases • Iodines	 1.00 0.50	 1.00 0.50
Coolant to Steam Dome Fractions • Noble Gases • Iodines	 1.00 0.10	 1.00 0.10
Plateout in Condenser/Turbine • Noble Gases • Iodines	 1.00 0.10	 1.00 0.10
Leakage Rate	1% per day	1% per day
Leakage Duration	24 hours	24 hours

Additional Information

Control Room (CR) Ventilation • Ingress/Egress • Filtered Intake • Discharge • Filtered Recirculation	 0 cfm 4,000 cfm 4,000 cfm 0 cfm	 10 cfm 1947.6 cfm 10 + 1947.6 = 1957.6 cfm 1947.6 cfm
CR Filter Start Time	66 sec.	66 sec.
EAB X/Q • 0-2 hours	 8.36×10^{-4} sec./m^3	 8.36×10^{-4} sec./m^3
LPZ X/Q • 0-8 hours • 8-24 hours • 1-4 days • 4-30 days	 1.12×10^{-4} sec./m^3 7.82×10^{-5} sec./m^3 3.61×10^{-5} sec./m^3 1.19×10^{-5} sec./m^3	 1.12×10^{-4} sec./m^3 7.82×10^{-5} sec./m^3 3.61×10^{-5} sec./m^3 1.19×10^{-5} sec./m^3
Main Control Room X/Q • 0-20 minutes • 20 minutes - 8 hours • 8-24 hours • 1-4 days • 4-30 days	 4.04E-03 sec/m^3 4.04E-03 sec/m^3 3.03E-03 sec/m^3 9.29E-04 sec/m^3 1.62E-04 sec/m^3	 4.04E-03 sec/m^3 9.65E-04 sec/m^3 7.56E-04 sec/m^3 2.32E-04 sec/m^3 4.05E-05 sec/m^3
Dose Conversion Factors	ICRP 2	ICRP 30

The following are the changes in assumptions between the current analysis documented in the USAR and the analysis prepared for Power Uprate:

- **Power Level:** A power level of 3100 MWt was assumed which is 102% of the Power Uprate power level.
- **Damaged Rods:** The number of damaged rods assumed in the initial analysis was based on a GE document. GESTAR II methodology dictates that an additional 10% be added to the number of damaged rods (770 x 1.1 = 847 ≈ 850). Use of GE 8 (8x8) fuel is slightly more conservative than use of GE11 (9x9 array) fuel.
- **Radial Peaking Factor:** The RPF assumed was conservatively increased from the 1.5 recommended in regulatory guidance to 1.65. This is consistent with the FHA analyses submitted to the NRC (Amendment 110).
- **Control Room Ventilation Model:** The revised analysis uses the current MCR ventilation model (~2000 cfm filtered intake & 2000 cfm filtered recirc.). The previous analysis used the previous model (4000 cfm filtered intake, no filtered recirculation).
- **Control Room X/Q:** The main control room X/Q were changed in order to credit SRP 6.4. Specifically, that document allows a factor of 4 reduction in MCR X/Q for plants with manual dual air intakes. This credit is also taken in the LOCA (Amendment 111) and FHA (Amendment 110) analyses.
- **Dose Conversion Factors:** The previous analysis used ICRP2 DCF. The Power Uprate analysis used ICRP30 DCFs.

Table 6 – CRDA Results

Dose (REM)	Power Uprate Analysis	Current USAR Value
EAB		
• Whole Body	7.036E-01	9.900E-01
• Thyroid	5.447E+00	7.800E+00
LPZ		
• Whole Body	2.163E-01	4.300E-01
• Thyroid	4.506E+00	5.900E+00
Control Room		
• Whole Body	7.637E-02	4.900E-01
• Skin	1.339E+00	7.900E+00
• Thyroid	5.505E-01	3.300E+00

Fuel Handling Accident

Note: The results of these analyses were submitted independently to the NRC via LAR 99-29 which was approved by the NRC via Technical Specification Amendment 110.

Table 7 - FHA Input Assumptions

Parameter	Case I	Case II (Amendment 35)	Case III (Amendment 85)
Building	Fuel Building	Containment	Containment
Core Power Level[(1)]	3100 MWt	3100 MWt	3100 MWt
Number of Pins per Bundle	74	74	74
Number of Bundles in Core	624	624	624
Decay Time	24 hr.	80 hr.	11 days
Number of Damaged Pins	150 GE9x9	150 GE9x9	150 GE9x9
Release Rate[(2)]	Puff Release	87.4 $^{vol. \%}/_{day}$	6000 $^{vol. \%}/_{day}$
Building Filter Efficiency	99%	0%	0%
Pool Decontamination Factor			
Halogens	100	100	100
Noble Gases	1	1	1
Off-Site Atmospheric Dispersion Factors[χ/Q] ($^s/_{m^3}$)			
EAB			
0 – 2 hours	8.58E-04	8.58E-04	8.58E-04
LPZ			
0 – 8 hours	1.13E-04	1.13E-04	1.13E-04
8 – 24 hours	7.89E-05	7.89E-05	7.89E-05
1 – 4 days	3.65E-05	3.65E-05	3.65E-05
4 – 30 days	1.21E-05	1.21E-05	1.21E-05

Control Room χ/Q (s/m^3)[3]			
0 – 8 hours	1.62E-03	1.62E-03	1.62E-03
8 – 24 hours	1.20E-03	1.20E-03	1.20E-03
1 – 4 days	4.05E-04	4.05E-04	4.05E-04
4 – 30 days	6.48E-05	6.48E-05	6.48E-05
Gap Fractions			
Kr-85	0.30	0.30	0.30
All Other Noble Gases	0.10	0.10	0.10
I-131	0.12	0.12	0.12
All Other Halogens	0.10	0.10	0.10

Note 1: The assumed power level corresponds to a Power Uprated core thermal power level of 3039 MWt. An instrument uncertainty of 2% is assumed in accordance with Regulatory Guide 1.78 for a total core power level of 1.02x3039=3100 MWt.

Note 2: Case 1 represents the most conservative assumption in that all activity is instantaneously released to the environment. The Case II leakage rate corresponds to L_a (0.26 vol % per day) + 70.2 cfm and accounting for only 10% mixing (per the Amendment 35 SER). The Case III release rate used ensures that the Regulatory Guide 1.25 two hour release duration is met.

Note 3: The FHA analyses assumes an "operator action" at 20 minutes for Control Room personnel to manually select the most favorable air intake. Therefore, the χ/Q values presented in the Table are divided by four, beginning 20 minutes into the event, as allowed per SRP Section 6.4.

Table 2 - Power Uprate Analyses - Summary of Results

Case I

Dose (REM)	Pre-Uprate	Power Uprate Analysis	Regulatory Limits
EAB			
• Whole Body	0.3	0.5	6
• Thyroid	1.1	1.9	75
LPZ			
• Whole Body	0.1	0.1	6
• Thyroid	0.2	0.3	75
Main Control Room			
• Whole Body	0.1	0.1	5
• Skin	1.0	1.7	30
• Thyroid	2.0	3.3	30

Additional Information

Case II

Dose (REM)	**Pre-Uprate**	**Power Uprate Analysis**	**Regulatory Limits**
EAB			
• Whole Body	0.1	0.1	6
• Thyroid	7.8	9.3	75
LPZ			
• Whole Body	0.1	0.1	6
• Thyroid	6.3	7.5	75
Main Control Room			
• Whole Body	0.1	0.1	5
• Skin	0.2	0.2	30
• Thyroid	0.3	0.4	30

Case III

Dose (REM)	**Pre-Uprate**	**Power Uprate Analysis**	**Regulatory Limits**
EAB			
• Whole Body	0.1	0.2	6
• Thyroid	36*	67	75
LPZ			
• Whole Body	0.1	0.1	6
• Thyroid	7.2	8.8	75
Main Control Room			
• Whole Body	0.1	0.1	5
• Skin	0.2	0.4	30
• Thyroid	0.8	3.6	30