October 6, 2000

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: INCREASE IN MAXIMUM ALLOWABLE THERMAL POWER TO 3039 MEGAWATTS THERMAL (TAC NO. MA6185)

Dear Mr. Edington:

The Commission has issued the enclosed Amendment No. 114 to Facility Operating License (FOL) No. NPF-47 for the River Bend Station, Unit 1. The amendment consists of changes to the Technical Specifications (TSs) and the FOL in response to your application dated July 30, 1999, as supplemented by letters dated April 3, May 9, July 18, August 24, and October 2, 2000.

The amendment changes the FOL and the TSs to allow an increase in the maximum, allowable thermal power from 2894 megawatts thermal (MWt) to 3039 MWt.

A copy of our related Safety Evaluation is enclosed. The Notice of Issuance will be published in the *Federal Register*.

Sincerely,

/RA/

Jefferey F. Harold, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FRN: ML003761606

Docket No. 50-458

Enclosures: 1. Amendment No. 114 to NPF-47
2. Safety Evaluation
3. Notice of Issuance

cc w/encls: See next page
* see previous concurrence

DISTRIBUTION: See attached list
ACCESSION NO: ML00

OFFICE	PDIV-1/PM	PDIV-1/LA	PDIV-1/PM	EMCB *	SRXB *	IOLB *	IQMB *	EEIB *	SPLB *
NAME	DJaffe	DJohnson for JHA	JHarold	WBateman	JWermiel	GTracy	TQuay	JCalvo	JHannon
DATE	10/05/00	10/5/00	10/5/00	9/15/00	9/11/00	9/18/00	9/11/00	9/11/00	9/13/00

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NAME	RHoefling	RGramm for	SDembek for SRichards	JZwolinski	PKleene	BSheron	SCollins for SJC
DATE	10/4/00	10/6/00	10/4/00		9/19/00	10/6/00	10/6/00

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River Bend Station

cc:

Winston & Strawn
1400 L Street, N.W.
Washington, DC 20005-3502

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

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

President of West Feliciana
Police Jury
P. O. Box 1921
St. Francisville, LA 70775

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

Ms. H. Anne Plettinger
3456 Villa Rose Drive
Baton Rouge, LA 70806

Administrator
Louisiana Radiation Protection Division
P. O. Box 82135
Baton Rouge, LA 70884-2135

Wise, Carter, Child & Caraway
P. O. Box 651
Jackson, MS 39205

Executive Vice President and
Chief Operating Officer
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286

General Manager - Plant Operations
Entergy Operations, Inc.
River Bend Station
P. O. Box 220
St. Francisville, LA 70775

Director - Nuclear Safety
Entergy Operations, Inc.
River Bend Station
P. O. Box 220
St. Francisville, LA 70775

Vice President - Operations Support
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-1995

Attorney General
State of Louisiana
P. O. Box 94095
Baton Rouge, LA 70804-9095

DATED: October 6, 2000

AMENDMENT NO. 114 TO FACILITY OPERATING LICENSE NO. NPF-47 - RIVER BEND STATION, UNIT 1

Hard Copy:
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PDIV-1 R/F

E-Mail:
RidsNrrOd (SCollins/RZimmerman)
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ENTERGY GULF STATES, INC.

AND

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

RIVER BEND STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 114
License No. NPF-47

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

   A. The application for amendment by Entergy Operations, Inc.* (the licensee) dated July 30, 1999, as supplemented by letters dated April 3, May 9, July 18, August 24, and October 2, 2000, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

* Entergy Operations, Inc. 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.

2. Accordingly, the license is amended by changes to the Operating License and the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of FOL No. NPF-47 is hereby amended to read as follows:

   (2) <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 114 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. Entergy Operations, Inc. 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 no later than the start-up following the next refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Brian W. Sheron for

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachments: Changes to the Facility Operating License and Technical Specifications

Date of Issuance: October 6, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 114

FACILITY OPERATING LICENSE NO. NPF-47

DOCKET NO. 50-458

Replace the following pages of Facility Operating License No. NPF-47 and 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.

FACILITY OPERATING LICENSE

Remove	Insert
3	3
6	6

TECHNICAL SPECIFICATIONS

Remove	Insert
1.0-5	1.0-5
1.0-26	1.0-26
1.0-27	1.0-27
1.0-28	1.0-28
2.0-1	2.0-1
3.1-14	3.1-14
3.1-16	3.1-16
3.1-17	3.1-17
3.1-20	3.1-20
3.1-21	3.1-21
3.1-22	3.1-22
3.2-1	3.2-1
3.2-2	3.2-2
3.2-3	3.2-3
3.3-2	3.3-2
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
3.3-8	3.3-8
3.3-31	3.3-31
3.3-53	3.3-53
3.3-67	3.3-67
3.4-1	3.4-1
3.4-9	3.4-9
3.4-10	3.4-10

*New information added to page 3.3-3 caused current information to shift to the next page. No new changes were made.

Remove	Insert
3.4-16	3.4-16
3.4-28	3.4-28
3.4-29	3.4-29
3.4-32	3.4-32
3.4-33	3.4-33
3.5-11	3.5-11
3.7-14	3.7-14
3.9-7	3.9-7
3.10-19	3.10-19
3.10-22	3.10-22

(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 Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated 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.

(13) Partial Feedwater Heating (Section 15.1. SER)

During power operation, the facility shall not be operated with a feedwater heating capacity which would result in a rated thermal power feedwater temperature less than 326 °F.

(14) Emergency Response Capabilities (Generic Letter 82-33. Supplement 1 to NUREG-0737. Section 7.5.2.4. SER and SSER 3. Section 18. SER. SSER 2 and SSER 3)

E0I shall complete the requirements of NUREG-0737 Supplement #1 as specified in Attachment 5. Attachment 5 is hereby incorporated into this license.

(15) Salem ATWS Events. Generic Letter 83-28 (Section 7.2.2.5. SSER 3)

E0I shall submit responses to and implement the requirements of Generic Letter 83-28 on a schedule which is consistent with that given in its letters dated August 3. 1984 and May 30. 1985.

(16) Merger Related Reports

Entergy Gulf States. Inc. shall inform the Director. NRR:

a. Sixty days prior to a transfer (excluding grants of security interests or liens) from Entergy Gulf States. Inc. to Entergy or any other entity of facilities for the production. transmission or distribution of electric energy having a depreciated book value exceeding one percent (1%) of Entergy Gulf States. Inc.'s consolidated net utility plant. as recorded on Entergy Gulf States. Inc.'s books of account.

b. Of an award of damages in litigation initiated against Entergy Gulf States. Inc. by Cajun Electric Power Cooperative regarding River Bend within 30 days of the award.

(17) Primary containment air lock doors may be open during CORE ALTERATIONS, except when moving recently irradiated fuel. (i.e.. fuel that has occupied part of a critical reactor core within the previous 11 days). provided the following conditions exist:

1) One door in each air lock is capable of being closed.

2) Hoses and cables running through the air lock employ a means to allow safe. quick disconnect and are tagged at both ends with specific instructions to expedite removal.

3) There is a minimum of 23 feet of water over the core.

4) The air lock doors are not blocked open to allow expeditious closure.

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 ≥ 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 <u>AND</u> 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 ≤ 10% from established patterns. b. Recirculation loop drive flow versus total core flow differs by ≤ 10% from established patterns. c. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns. or each jet pump flow differs by ≤ 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.	12 hours
	AND	
	A.2 Be in MODE 4.	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.	Immediately
	<u>AND</u>	
	C.2 Determine RCS is acceptable for operation.	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:	30 minutes
	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	

(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)

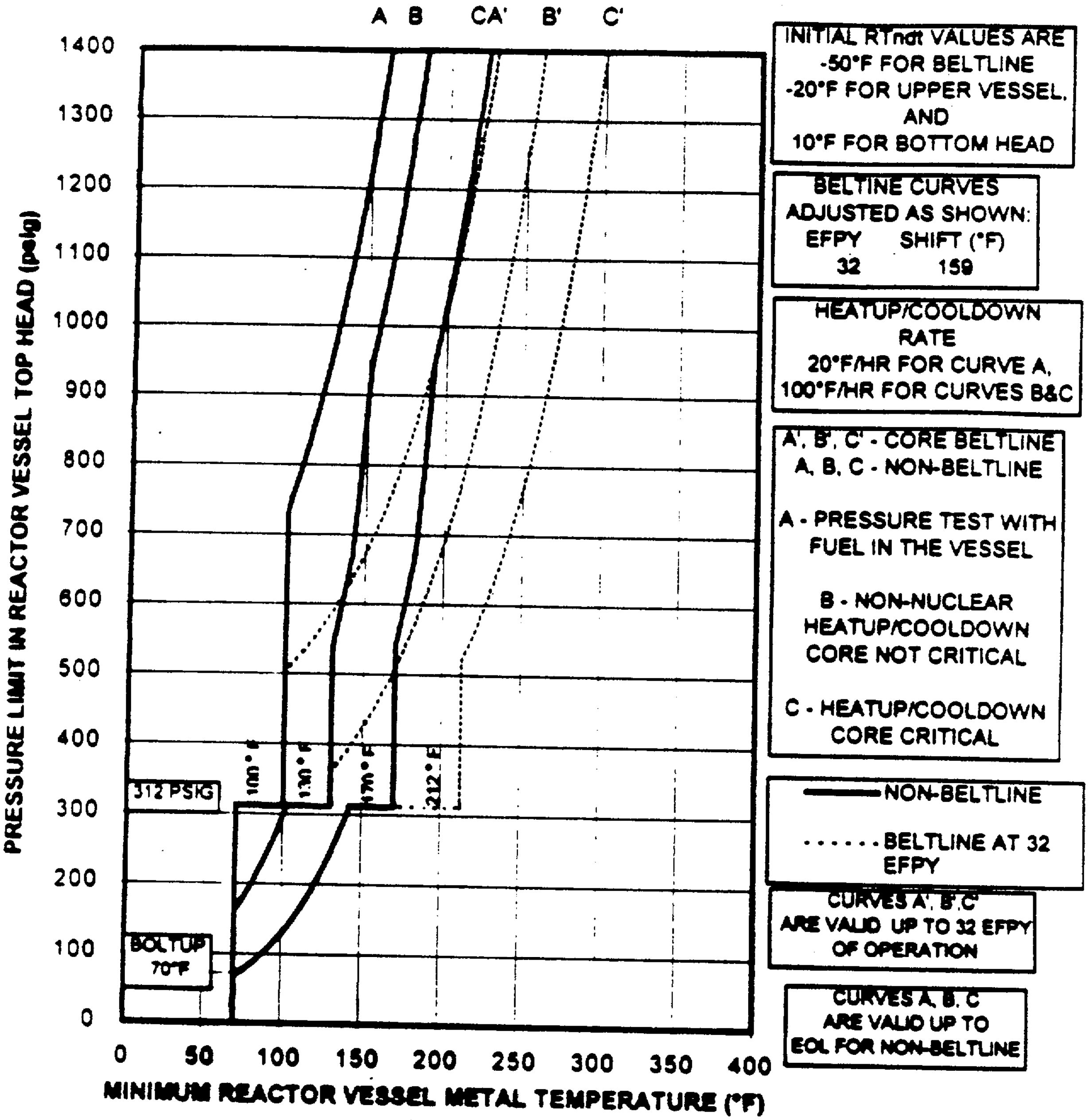

Figure 3.4.11-1 (page 1 of 1)
Minimum Temperature Required vs. RCS Pressure

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 $\geq$ 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 $\geq$ 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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 114

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 July 30, 1999 (Reference 1), as supplemented by letters dated April 3, (Reference 2), May 9 (Reference 3), July 18 (Reference 4), August 24 (Reference 5), and October 2, 2000 (Reference 40), Entergy Operations, Inc. (EOI, the licensee) requested changes to Facility Operating License No. (FOL) NPF-47 and the Technical Specifications (TSs), Appendix A to the FOL for the River Bend Station, Unit 1 (RBS). The proposed changes would allow an increase in the maximum allowable thermal power from 2894 megawatts thermal (MWt) to 3039 MWt.

2.0 BACKGROUND

RBS is a boiling water reactor (BWR), Model Six with a Mark III containment and is described in the RBS updated safety analysis report (USAR). At the present time, RBS is restricted to operation at 2894 MWt by FOL NPF-47 and the TSs. The proposed changes to the FOL and TSs would allow the licensee to increase the maximum allowable thermal power from 2894 MWt to 3039 MWt, a 5 percent power increase. The licensee plans to implement the 5 percent power increase at RBS in two phases; a steam flow/feedwater flow increase in power (flow-only, Phase One) to be implemented with the plant in operation, and a flow increase/reactor pressure increase phase (Phase Two) to be completed after the next refueling outage or an outage of sufficient duration to prepare for this phase. The high-pressure main turbine steam flow path was modified during Refueling Outages 8 and 9 to accommodate the increase in reactor thermal power output. These changes reduce the pressure drop through the high pressure-turbine.

Following issuance of the proposed changes to the FOL and the TSs, the licensee plans to increase reactor power without an intervening shutdown (SD). The planned approach to achieve the higher power level for the flow-only phase involves (1) an increase in the core thermal power (with a more uniform and flattened power distribution) to create increased steam flow, (2) a corresponding increase in feedwater system flow, (3) no increase in maximum core flow, (4) reactor operation primarily along an extension of the current rod/flow control lines, (5) and a small increase in reactor operating pressure. Following startup from the next refueling outage, the full-power increase phase would be implemented with an increase in reactor pressure and steam/feedwater flow. This approach is consistent with the BWR generic power uprate guidelines presented in General Electric (GE) report NEDC-31897P-A (Reference 6). The plant-unique evaluations which follow the guidelines are based on a review of plant design and operating data to confirm excess design capabilities and, if necessary, identify any areas that may require modifications associated with the power uprate. For some items, bounding analyses and evaluations in NEDC-31984P (Reference 7) demonstrate plant operability and safety.

3.0 EVALUATION

Reference 1 contained GE Nuclear Energy Licensing Topical Report NEDC-32778P (Reference 8) as an enclosure, which provided a safety analysis of the proposed 5 percent uprate for RBS. The U.S. Nuclear Regulatory Commission (NRC or the Commission) staff review of the licensee's application and supporting information generally follows the format of Reference 8.

3.1 Reactor Core and Fuel Performance

3.1.1 Fuel Design and Operation

All fuel in the current RBS core is supplied by GE. RBS is currently scheduled to transition to Siemens Power Corporation (SPC) fuel in Fuel Cycle 11. RBS will make the required submittal to the NRC for transition to SPC to reflect the use of SPC methodology. This evaluation of the fuel is only applicable to GE fuel.

The power uprate will increase the plant's average power density; however, this power density will remain within the current operating power density range of operating BWRs. The power uprate will have minor effects on operating flexibility, reactivity characteristics, and energy requirements. The power distribution of the core will be changed to achieve increased core power while limiting the absolute power in any individual fuel bundle.

At uprated conditions, all fuel and core design limits will continue to be met by planned deployment of fuel enrichment and burnable poison and adjustments of core management control rod patterns or core flow. Revised loading patterns, larger batch sizes, and new fuel designs may be used to provide additional operating flexibility and maintain fuel cycle length. Core configurations will be evaluated on a cycle-specific basis in accordance with the RBS TSs.

The reactor core design power distribution usually represents the most limiting thermal operating state at design conditions. It includes allowances for the combined effects on the fuel heat flux and temperature of the gross and local power density distributions, control rod pattern, and reactor power level adjustments during plant operation. Core design methods are not changed for the power uprate. Parametric studies for the RBS demonstrate that the uprate can be accommodated. Thermal-hydraulic design and operating limits (OLs) ensure an acceptably low probability of boiling transition in the core at any time, even for the most severe postulated operational transients. Limits are also placed on the fuel average planar linear heat generation rates in order to meet peak cladding temperature limits for the limiting loss-of-coolant accident (LOCA) and fuel mechanical design bases.

The reloaded core designs for operation at the uprated power take into account the applicable limits to assure acceptable margins between the licensing limits and their corresponding operating values. The power uprate may result in an increase in fuel burnup relative to the current level of burnup, but NRC-approved limits on the fuel designs to be utilized will not be exceeded.

Management of fuel performance will continue to be governed by the core operating limit report (COLR) prepared for uprated power, as defined in RBS TS 5.6.5. Any fuel degradation identified in the future will continue to be managed by the station's existing program for monitoring fuel integrity and action program for failed fuel.

The impact of higher power operation on radiation sources and design basis accident (DBA) doses are discussed in Reference 8. The power uprate will have minor effects on operating flexibility, reactivity characteristics, and energy requirements. These items are discussed below.

3.1.2 Thermal Limits Assessment

OLs are established to ensure that regulatory and/or safety limits (SLs) are not exceeded for a range of postulated events such as transients and accidents. This section addresses the effect of power uprate on thermal limits. A representative cycle core (RBS Cycle 7) was used for the uprate evaluation. Cycle-specific core configurations will be evaluated for each reload to confirm the power uprate capability and to establish or confirm cycle-specific limits, in accordance with the currently required practice.

3.1.2.1 Minimum Critical Power Ratio (MCPR) OL

The OL MCPR is determined on a cycle-specific basis from the results of a reload analysis, as described in Reference 7, which does not change for the power uprate.

3.1.2.2 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Maximum Linear Heat Generation Rate (LHGR) OLs

The MAPLHGR and LHGR limits will also be maintained as described in Reference 7. The plant-specific safety evaluation (SE) for RBS is contained in Reference 8 and GE Nuclear Energy Licensing Topical Report NEDC-32778P (Reference 9).

3.1.3 Reactivity Characteristics

All minimum SD margins apply to cold SD conditions, and will be maintained without change. Operation at higher power could reduce the excess reactivity during the fuel cycle. The potential loss of reactivity will not significantly degrade the ability to manage the power distribution through the cycle to achieve the target power level. Through fuel cycle design, sufficient excess reactivity can be obtained to match the desired cycle length. The increase in hot reactivity may result in less hot-to-cold reactivity difference and, therefore smaller cold SD margins; however, this loss in the margin can be accommodated through core design. If needed, a fuel bundle design with improved SD margin characteristics can be used to preserve the flexibility between hot and cold reactivity requirements for future cycles.

3.1.3.1 Power/Flow Operating Map

The uprated power/flow operating map includes operating domain changes for the uprated power level, including the plant performance improvement features identified in Section 1.3.2 of Reference 8. The uprate-related changes to the power/flow operating map are consistent with the previous NRC-approved generic descriptions in Reference 7. The maximum thermal operating power and maximum core flow shown on Figure 2-1 in Reference 8 correspond to the uprated power and the previously analyzed core flow range, but are rescaled so that uprated power is 100 percent rated. The changes to the power/flow operating map are consistent with the previous NRC-approved generic descriptions in Reference 7.

The recirculation pump cavitation line in the power flow map is affected by the flow-only increase operation. The reduction in reactor pressure during the flow-only increase operation results in a slight reduction in subcooling; however, when the power map was created, no credit was taken for the improvement in subcooling.

3.1.4 Stability

The RBS has implemented long-term stability solution Enhanced Option 1-A (E1A). The NRC approval of E1A is documented in an NRC letter to EOI dated May 5, 1999 (Reference 10). The solution consists of exclusion, restricted, and monitored regions on the power/flow map, as well as a defense-in-depth instability detection system. The exclusion and restricted regions are enforced by flow-biased scram and control rod blocks, which are implemented by hardware changes to the flow control trip reference (FCTR) card. The monitored region is administratively controlled. The E1A regions are affected by rated core power operating conditions and other plant/fuel cycle changes associated with power uprate; therefore, the E1A regions cannot be recalculated for power uprate until the fuel cycle (actual core reload) conditions are defined. Corresponding revised scram and rod block set points will be implemented on the FCTR card. The instability detection system is not affected by power uprate.

3.1.5 Reactivity Control

3.1.5.1 Control Rod Drives (CRDs) and CRD Hydraulic System

The CRD system controls gross changes in core reactivity by positioning neutron absorbing control rods within the reactor. It is also required to scram the reactor by rapidly inserting withdrawn control rods into the core. The CRD system was evaluated for Phase One and Phase Two operation.

Operation of the CRD system is not impacted when reactor pressure is not increased and, thus, there are no changes during Phase One operation.

Regarding Phase Two operation, the CRD system scram performance was evaluated for a bounding reactor dome pressure of 1059 psig at 102 percent of uprated power and an additional 35 psi for the vessel bottom head; rod insertion is slowed slightly due to the increased pressure. The licensee predicts that the scram times for power uprate will increase no more than 9 milliseconds than at 1050 psig (1065 psia) reactor dome pressure; therefore, the higher dome pressure due to power uprate will have little effect on the scram protection, and the performance of a nominal CRD during power uprate will be able to meet current TSs. The TS control rod surveillance scram time requirements are specified at two reactor dome pressures (950 and 1050 psig before power uprate), and are used to determine the intermediate scram time requirements by linear interpolation. To accommodate the higher nominal operating pressure condition of 1055 psig, the dome pressure of 1050 psig is revised for power uprate to 1059 psig, while the corresponding scram times remain unchanged. Because surveillance scram time testing is normally performed at pressures less than the rated power nominal dome pressure, 1059 psig is judged to be a sufficiently high pressure for bounding the expected scram time testing pressure condition.

The TS allowable scram accumulator and charging water header minimum pressures would be increased from 1520 psig to 1540 psig to maintain pre-power-uprate margins relative to the TS surveillance scram time limits. Based on the results of these evaluations, the NRC staff concludes that the CRD system will continue to perform all its functions at the uprated power level.

The licensee indicated that control rod drive mechanisms (CRDMs) have been designed in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (the Code), Section III, 1974 edition with addenda to and including winter 1975, which is the Code of record. The components of the CRDM which form part of the primary pressure boundary have been designed for a dome pressure of 1250 psig, which is higher than the maximum operating pressure of 1160 psig (the updated vessel dome pressure scram analytical limits of 1125 psig plus 35 psi for the reactor bottom head).

In Reference 2, the licensee indicated that the maximum stresses in the CRDMs remain within the allowable stress limits since they are caused by the maximum pump discharge pressure, which is not affected by the power uprate. The analysis of cyclic operation of the CRDMs resulted in a maximum cumulative usage factor (CUF) of 0.15 for the limiting CRD main flange for the power uprate. This is less than the Code-allowable CUF limit of 1.0.

On the basis of its review, the NRC staff concludes that the CRDMs will continue to meet their design basis and maintain their structural and pressure integrity at the uprated power conditions.

3.2 Reactor Coolant System (RCS) and Connected Systems

3.2.1 Nuclear System Pressure Relief - Safety/Relief Valve Setpoint Tolerance

The nuclear boiler pressure relief system prevents overpressurization of the nuclear system during abnormal operating transients. The safety/relief valves (SRVs) provide this protection.

Power uprate operation without a pressure increase and without the modification to the SRV set points retains the pressure relief capacity and margins in the current design. The SRVs are not affected by Phase One operation.

The set points for the SRVs would be increased for Phase Two operation. The operating steam dome pressure would be increased to achieve good control characteristics for the turbine control valves (TCVs) at the higher steam flow condition corresponding to Phase Two operation. The appropriate increase in the SRV set points also ensures that adequate differences between operating pressure and set points are maintained (i.e., the "simmer margin"), and that the increase in steam dome pressure does not result in an increase in the number of unnecessary SRV actuations.

The SRVs have three main protection functions: (1) overpressure relief operation (power relief mode), in which the valves open automatically to limit reactor pressure, (2) overpressure safety operation (spring safety mode) to prevent reactor overpressurization of the reactor pressure vessel (RPV), and (3) automatic depressurization system (ADS) operation, wherein designated SRVs automatically actuate as part of the emergency core cooling system (ECCS) for events involving small breaks in the reactor pressure boundary. The proposed power uprate does not alter the SRV lift set point test frequency or the number of SRVs required to be operable. Also, the proposed power uprate requires the as-left safety valve function settings to be within plus or minus 1 percent of the specified nominal lift set points before the valves were installed and tested. The licensee has proposed a change in as-found SRV set point tolerance from minus 2 percent to 0 percent (-2/+0%) to plus or minus 3 percent (+/-3%) as one of the performance improvement features in the power uprate. The NRC staff has previously granted approval to individual BWRs to increase the as-found SRV tolerance to 3 percent. The basis for the approval is described in the NRC staff SE for NEDC-31753P (Reference 11) of the set point tolerance increase. The NRC staff's SE included six conditions which must be addressed on a plant-specific basis for licensees applying for the increased SRV set point tolerance:

1. Transient analysis of all abnormal operational occurrences, as described in NEDC-31753P, should be performed utilizing a [plus or minus 3 percent] tolerance for the safety mode of SSVs [steam safety valves] and SRVs. In addition, the standard reload methodology (or other method approved by the staff) should be used for this analysis.

   RBS has performed an evaluation to determine if the proposed set point tolerance would affect any of the previously analyzed abnormal operational occurrences. Each of these abnormal occurrences were analyzed using the safety function lift set points at the proposed plus or minus 3 percent tolerance. The analyses conducted at the uprated conditions verified that the SL MCPR (SLMCPR) is not violated. The NRC staff concludes that this condition is acceptable.

2. Analysis of the design basis overpressurization event using the [3 percent] tolerance limit for the SRV setpoint is required to confirm that the vessel pressure does not exceed the ASME pressure vessel code upset limit.

   The overpressurization analyses credits five SRVs in safety mode and four SRVs in relief mode. The licensee has reevaluated the limiting design basis pressurization transient using the 3 percent tolerance limit to confirm that the vessel pressure does not exceed the Code upset limit. Refer to Section 3.2.2 of this SE for the code overpressure protection analyses. The NRC staff concludes that this condition is acceptable.

3. The plant-specific analysis described in Items 1 and 2 should assure that the number of SSVs, SRVs, and relief valves (RVs) included in the analyses correspond to the number of valves required to be operable in the [TSs].

The number of SRVs assumed in the analyses required in Items 1 and 2 above is consistent with RBS TS 3.4.4, crediting operation of only five safety mode SRVs and four relief mode of SRVs, which the NRC staff concludes is acceptable.

4. The performance of high-pressure systems (pump capacity, discharge pressure, etc.), motor-operated valves (MOVs), and vessel instrumentation and associated piping must be evaluated, considering the [3 percent] tolerance limit.

   The high-pressure systems are as follows:

   (1) Reactor Core Isolation Cooling (RCIC)

   The RCIC system power uprate analysis included the increased SRV setpoint tolerance. Refer to Section 3.2.8 of this SE.

   (2) High Pressure Core Spray (HPCS)

   The HPCS system power uprate analysis included the increased SRV set point tolerance. Refer to Section 3.3.2.1 of this SE.

   (3) Standby Liquid Control System (SLCS)

   The SLCS system operation is not affected by the SRV set point tolerance increase.

5. The effect of the [plus or minus 3 percent] tolerance on any plant-specific alternate operating modes (e.g., increased core flow, extended operating domain, etc.) should be evaluated.

The licensee's analyses included evaluations for the currently approved operating domains: maximum extended load line limit analysis, single loop-operation (SLO), increased core flow, and feedwater temperature reduction. The analyses for the power uprate included the increased SRV set point tolerance. The results of these analyses were acceptable, and adequate margin is maintained for the alternate modes of operation stated above.

6. The effect of the [3 percent] tolerance limit on the containment response during [LOCAs] and the effect of hydrodynamic loads on the SRV discharge lines and containment should be evaluated.

The SRV air-clearing loads include discharge line (SRVDL) loads, suppression pool boundary pressure loads, and drag loads on submerged structures. These loads are influenced by the SRV opening setpoint pressure, the initial water leg height in the SRVDL, SRVDL geometry, and suppression pool geometry. Of these parameters, only the SRV setpoint pressure is affected by power uprate and can impact the SRV loads.

The licensee indicated that the SRV opening setpoint, which is the basis for the SRV loads on the suppression pool boundary, and submerged structure is 1190 psig. The power uprate results in an increase in as-found SRV opening setpoint pressure of 3 percent. The licensee stated that an evaluation performed for the combined effect of power uprate and 3 percent tolerance results in an increase in the SRV load of less than 2 percent and is well within the conservatism in the SRV loads defined for RBS.

Based upon the above evaluation, the NRC staff concludes that the plant operation at uprated power will not impact the SRV containment loads. The NRC staff also concludes that the licensee adequately addressed the six conditions identified in Reference 11. Accordingly, the NRC staff concludes that the proposed increase in the as-found SRV set point tolerance is acceptable.

3.2.2 Reactor Overpressure Protection

The results of the overpressure protection analysis are contained in each cycle-specific reload safety analysis. The design pressure of the RPV remains at 1250 psig. The Code-allowable peak pressure for the RPV is 1375 psig (110 percent of the design value), which is the acceptance limit for pressurization events. The limiting pressurization event is a main steam isolation valve (MSIV) closure with a failure of the valve position scram. This transient was analyzed by the licensee with the NRC staff-approved model ODYN (Reference 12) and the assumptions listed in Reference 8. For the power uprate, the analysis assumes the event initiates at a reactor dome pressure of 1078 psig, which is higher than the nominal uprated dome pressure. The seven lowest setpoint SRVs (out of a total of 16 SRVs) are assumed to be out-of-service in the overpressurization analysis. Consistent with the RBS TS, a total of nine SRVs (five in the safety mode and four in the relief mode) were assumed operable in the overpressure analysis. The SRV opening pressures were positive 3 percent above the nominal setpoint for the valves as shown in Table 5-1 of Reference 8. At uprated conditions, a higher peak RPV pressure of 1347 psig occurs at the bottom of the RPV (compared to 1305 psig for the cycle 8 pre-uprate analysis results), but the pressure remains below the 1375 psig Code limit. The corresponding calculated dome pressure is 1322 psig. The peak calculated RPV pressure remains below the 1375 psig ASME Code limit, and the maximum dome pressure remains below the RBS TS 2.1.2 SL of 1325 psig; therefore, there is minimal decrease in the margin of safety.

When this event is reanalyzed for Phase One operation, it would be initiated from a lower pressure (30 psi lower). The result is that for Phase One operation without the pressure increase, the resultant peak vessel pressure is less than the 1322 psig calculated for the case with the pressure increase. The peak calculated RPV pressure remains below the 1375 psig Code limit, and the maximum dome pressure remains below the 1325 psig SL. The NRC staff-approved ODYN methodology was used for this analysis.

The NRC staff concludes that the overpressure protection is acceptable for the power uprate.

3.2.3 RPV and Internals

The licensee evaluated the RPV and internal components in accordance with the current licensing basis. Load combinations include reactor internal pressure difference (RIPD), LOCA, SRV discharge, seismic, and fuel lift loads.

The RIPDs for the proposed power uprate were recalculated as shown in Tables 3-5, 3-6, and 3-7 of Reference 8, for normal, upset, and faulted conditions, respectively. The seismic loads are unaffected by the power uprate. The existing design loads are unchanged since they are bounding even with the increase in dynamic loads from the LOCA and with SRV discharge for the power uprate conditions (Section 4.1.2 of Reference 8); however, in Reference 2, the licensee indicated that the power uprate at RBS will incorporate the use of GE 11 fuel, which affects the structural dynamic characteristics and the dynamic responses of the reactor and internals. The RPV and internals, therefore, were evaluated for the effects of the increased RIPDs and the increased seismic, SRV discharge, LOCA, and acoustic loads. In addition, in Reference 2, the licensee indicated that the LOCA loads such as the asymmetric pressurization and line break thrust loads were considered in appropriate load combinations for the evaluation of reactor internal components for power uprate and that the governing load combinations were used for detailed component evaluations. The load combinations for normal, upset, and faulted conditions were considered in the evaluation in accordance with the RBS USAR.

The stresses and CUFs for the reactor internal components and the core support structure were evaluated by the licensee in accordance with the Code of record at RBS, the ASME Code, Section III, 1974 Edition with summer 1976 addenda for RBS. The licensee evaluated the RPV components, nozzles, and supports in compliance with the Code of record, the 1971 Edition with addenda to and including summer 1973; however, for components that underwent design modifications, the governing Code for a particular component is the Code used in the stress analysis of that component. For instance, the recirculation inlet nozzle safe end was evaluated using the ASME Code, 1974 Edition with the addenda through summer 1976, consistent with the Code used in the analysis associated with the modification of that component. The NRC staff concludes that the methodology used by the licensee is consistent with the NRC-approved methodology in Appendix I of Reference 6, and is therefore acceptable.

The maximum stresses for critical components of the reactor internals, listed in Table 3-3 of Reference 8, indicate that the design criteria remain satisfied for the power uprate conditions. The calculated stresses are less than the Code-allowable limits shown in the table. The licensee provided the calculated stresses and CUFs in Table 3-2 of Reference 8 for critical components such as the main closure flange and studs, reactor vessel support skirt, refueling bellows, stabilizer brackets, and feedwater nozzles. The NRC staff concludes that the calculated CUFs and stresses provided by the licensee are within the Code-allowable limits and are therefore acceptable.

The licensee assessed the potential for flow-induced vibration based on the GE prototype plant vibration data for the reactor internal components recorded during startup testing and on operating experience from similar plants. The vibration levels were calculated by extrapolating the recorded vibration data to the power uprate conditions and compared to the plant-allowable limits for acceptance. The licensee found that the maximum flow-induced vibration occurs at the jet pump riser braces, which were also found to be within the acceptance limit for the RBS proposed power uprate conditions.

The NRC staff has reviewed the licensee's evaluations regarding the effect of the power uprate on core shroud and core spray piping and concludes that the licensee has bounded the effects of power uprate on the existing flaws. The NRC staff concludes that the proposed power uprate will not affect the operation of core shroud, core spray header, or any other RPV internals. With regard to the RPV piping, the proposed power uprate will slightly increase the licensee's susceptibility to erosion/corrosion (E/C), but the power uprate should not cause an adverse increase in E/C since EOI has reexamined its E/C inspection programs in light of plant-specific uprate concerns (i.e., increased flow-induced E/C in systems associated with the turbine cycle). The NRC staff concludes that the licensee's evaluation of E/C is acceptable.

Based on its review of the information provided by the licensee, the NRC staff concludes that the maximum stresses and CUFs are within the Code-allowable limits, and concludes that the RPV and internal components will continue to maintain their structural integrity for the power uprate conditions.

The licensee provided an assessment of (1) the impact of the power uprate on the adjusted reference temperature of the limiting RPV material, (2) the need to revise the RBS pressure-temperature (P-T) limit curves, (3) the changes in the predicted upper shelf energy (USE) drop for the RPV materials, and (4) whether changes in the RPV surveillance program (as required by 10 CFR Part 50, Appendix H) are necessary.

In analyzing the RPV, EOI examined the effect on the RPV fluence of operating RBS at a power of 3039 MWt until end-of-license (EOL). The license's analysis, therefore, attempted to address the expected RPV material embrittlement through EOL since it is directly related to the RPV neutron fluence, which is in turn related to the reactor operating power. In References 1 and 2, EOI provided the information on its current fluence projections, the new fluence projections calculated considering power uprate operation, and its fluence calculation methodology. In addition, information on the fluence calculation methodology for RBS was referenced in a letter dated May 8, 2000 (Reference 14), regarding the deferral of the withdrawal of the first RBS surveillance capsule from 10.4 effective full-power years (EFPY) to 13.4 EFPY.

The RBS fluence methodology is based on the use of a two-dimensional, discrete ordinate transport code. The code uses a distributed source term determined from core physics calculations and establishes a calculated RPV and surveillance capsule fluence distribution based on the transport model. The cross-sections for the transport code are prepared with 1/E flux weighted, first-order Legendre polynomial (P_1) expansion matrices for anisotropic scattering but do not include resonance self-shielding factors. In addition, simplifications, such as ignoring the presence of the jet pumps in the region between the core shroud and the vessel, are incorporated into the modeling. The results from first cycle dosimeter wire tests are then used to "scale" the calculated fluence distribution. The first-cycle dosimeter wire results establish a "measured" flux at the dosimeter capsule and the calculated fluence distribution provides the lead factor, defined as the ratio of the capsule flux divided by the peak vessel flux, between the dosimetry location and the peak RPV location. With these two quantities, and the operating time, the peak RPV fluence is determined. The licensee's analysis conservatively assumes the peak fluence value to be applicable to all RPV materials.

Previously, EOI had determined the first-cycle dosimetry to show a flux at the dosimetry capsule location of 4.4×10^9 neutrons per centimeter squared second ($n/(cm^2 \cdot s)$) at energy greater than 1 Mega electron volt (MeV) ($E > 1$ MeV) and a lead factor from the transport calculation of 0.67. EOI also determined that power density at the core periphery during a pre-power-uprate equilibrium cycle could be as much as 18 percent higher than the first cycle power density. For a post-power-uprate equilibrium cycle, this increase relative to the first cycle power density could be as much as 20 percent. As a result of these changes, the licensee's peak clad-to-base metal interface EOL (32 EFPY) RPV fluence estimate increased from 6.6×10^{18} n/cm^2 ($E > 1.0$ MeV) prior to the power uprate to 7.95×10^{18} n/cm^2 ($E > 1.0$ MeV) after the effects of the power uprate were considered. Fluence estimates for intermediate operating times would show an increase of lesser magnitude. The result of this fluence increase was observed to have a significant effect on the RBS pressure and temperature (P-T) limit curves and new curves were submitted as part of the power uprate. These P-T curves were intended to address operation up to 14 EFPY and 32 EFPY. These curves were submitted as Figures 3-2a and 3-2b in Reference 8 and would be incorporated as Figures 3.4.11-1 and 3.4.11-2, respectively, in the RBS TSs. Subsequently, in Reference 5, the 14 EFPY P-T curves were withdrawn from the final proposed TSs.

Regarding the RPV assessment, the NRC staff has reviewed the information provided by EOI. Given the bases presented for the EOI analyses, the NRC staff generally agrees with the conclusions reached by the licensee regarding the topic addressed in Section 3.3 of Reference 8; however, as addressed below, the NRC staff has expressed concerns regarding the RBS RPV fluence analysis. Hence, in addition to the TS changes submitted in the RBS power uprate amendment, the NRC staff requested that the licensee commit to use the "32 EFPY" P-T limit curves for RPV heatup, cooldown, criticality, and hydrostatic/leak rate testing until EOI can complete updated RPV fluence analyses for RBS. The licensee agreed to make this commitment and provided written confirmation of the commitment to the NRC in Reference 14.

The NRC staff's primary concern, with regard to the RBS RPV fluence calculation, is that EOI does not use a methodological approach that agrees with currently accepted industry standards. For example, the use of anything less than a third-order Legendre polynomial expansion (P_3) of the scattering cross-sections has long been known to inadequately address anisotropic neutron scattering and potentially lead to an underprediction of the RPV fluence. Likewise, the use of inverse energy (1/E) flux weighting and the failure to use resonance self-shielding factors may lead to an underprediction of the RPV fluence compared to state-of-the-art methodologies. Finally, the NRC staff has not accepted the use of dosimetry data to directly scale the results of neutron transport calculations (particularly in the case of first-cycle dosimetry data, which, as discussed above, is expected to be non-representative of equilibrium conditions) without a rigorous evaluation of the consistency of the dosimetry data.

To date, it has not been evident to the NRC staff that a sufficiently rigorous evaluation of the available RBS data has been performed to warrant the modifications incorporated in the RBS methodology. Hence, the NRC staff questioned use of the RBS fluence calculations while reviewing the RBS proposal to defer the withdrawal of the first RBS RPV surveillance capsule and questioned their use in the power uprate submittal to assess RPV integrity. A proposed interim solution to address the RBS fluence issues in the context of the power uprate submittal was developed. Since RBS has currently been operating for approximately 11 EFPY, the licensee had submitted power uprated P-T limit curves for both near-term application (based on a projected 14 EFPY fluence value) and for long-term application (based on a projected 32 EFPY fluence value). The NRC staff has concluded that a commitment by the licensee to use the power uprated P-T limit curves based on their current estimate of the 32 EFPY fluence value (7.95×10^{18} n/cm^2 (E > 1.0 MeV)) is sufficient to ensure that, for near-term operation, the RBS RPV will continue to be operated in a manner which will not challenge RPV integrity. The NRC staff reached this conclusion based on the fact that, even when the questions raised by the NRC staff are considered, the current RBS fluence methodology is not expected to be nonconservative by more than a factor of two. Hence, operation to the "32 EFPY" P-T limits proposed by the licensee is expected to be adequate to at least 16 EFPY if the fluence were calculated using a fluence methodology consistent with current industry practice.

The NRC staff approves the use of the P-T limit curves (designated as "32 EFPY" curves) through 16 EFPY of operation. For continued operation beyond 16 EFPY, additional information to address staff concerns regarding the RBS fluence calculations must be provided or an amendment to the P-T limit curves submitted. By letter dated May 8, 2000, EOI committed to use the "32 EFPY" curves in the power uprate request until the test results from the first RBS surveillance capsule were acquired and EOI received NRC approval to implement revised P-T limit curves based on this information. In accordance with the current RBS RPV surveillance capsule program, this information will be submitted to the NRC staff, as required by the reporting requirements of 10 CFR Part 50, Appendix H, with sufficient time for NRC staff review and action before the RBS RPV exceeds 16 EFPY of operation. Should EOI request NRC staff approval to modify the RBS RPV surveillance program, EOI should address how these modifications will affect their ability to meet the commitment in their May 8, 2000, letter and address how continued operation of the RBS RPV will be demonstrated in light of the surveillance program modifications.

In addition to the effect on the RBS P-T limit curves, EOI evaluated the effect of the proposed power uprate on the issue of RPV material USE drop and on the RBS RPV surveillance program. The licensee concluded that, with regard to these topics, appropriate analyses demonstrated continued compliance with the original design and licensing criteria of the reactor vessel; therefore, EOI determined that no changes in the licensing basis were required to address these issues. On the subject of USE, EOI stated that, in the post-power-uprate condition, the projected minimum USE for the RBS beltline materials at EOL is foot-pound (ft-lb), therefore, all materials will be above the 50 ft-lb limit required by 10 CFR Part 50, Appendix G. The licensee also concluded that no changes to the RBS RPV surveillance program were required to ensure continued compliance with 10 CFR Part 50, Appendix H.

Based upon the licensee's evaluations, the NRC staff concludes that the USE drop for the RPV materials, and the RBS Appendix H program, are acceptable for power uprate conditions. Regarding the previously mentioned issue of fluence calculational uncertainty, the staff assessment has demonstrated that, even assuming that the licensee-calculated fluence value at 32 EFPY was nonconservative by a factor of two, no RBS beltline material would be projected to fall below the acceptable upper shelf energy level of 50 ft-lb before EOL.

Based on the information presented above, the NRC staff has concluded that RPV integrity, RPV internals, and RCS erosion/corrosion issues have been adequately addressed in the EOI submittal. As stated above, this conclusion is predicated on the licensee's commitment in Reference 14 to operate using the submitted 32 EFPY RPV P-T limit curves until an updated, acceptable fluence analysis is completed for RBS. In this regard, withdrawal of the 14 EFPY curves, in Reference 5, satisfies the NRC staff's concerns. This fluence reanalysis will be completed prior to 16 EFPY of operation of the RBS RPV.

3.2.4 Reactor Recirculation System

The power uprate will be accomplished by operating along extensions of the rod lines on the power/flow map with no increase in maximum core flow (currently 107 percent of the original rated flow). The core reload analyses are performed with the most conservative allowable core flow. The evaluation of the reactor recirculation system performance at the uprated power level determined that adequate core flow can be maintained. For operation at the uprated power, the evaluated core flow is 107 percent.

Achieving the 107 percent core flow, under the flow-only increase operation, will require slightly higher recirculation pump motor horsepower. There is no significant impact on the recirculation system if reactor pressure is not increased.

The cavitation protection interlock will remain the same in terms of absolute flow rates. These interlocks are based on inlet subcooling in the external loop and thus are a function of absolute thermal power. With the full power uprate, slightly more subcooling occurs in the external loop due to the higher RPV dome pressure. Thus, it is possible to lower the cavitation interlock setpoint slightly. However, this change would be small, and is not necessary or recommended by GE.

The evaluation of recirculation pump net positive suction head (NPSH) found that, at full power, the power uprate alone does not increase the NPSH required and that the secondary effects of the 30 psi increase in RPV pressure increase the available NPSH; therefore, the power uprate alone increases the NPSH margin. The licensee concluded that the power uprate is, therefore, within the capability of the recirculation system.

During SLO, thermal power is currently limited to less than or equal to 83 percent of rated thermal power. To maintain the same power limit with respect to absolute power, this percent of rated thermal power value will be decreased to 79 percent, a decrease determined by the ratio of full power to the uprate power (100/105). This condition is addressed in proposed TS 3.4.1 (see Section 3.10, herein).

Recirculation pump vibration is not expected to be a problem because the RBS is equipped with flow control valves (FCVs) and there is no change in maximum core flow for the RBS power uprate. To maintain the same core flow with the increased core pressure drop (due to the increased steam production), recirculation flow (drive flow) increases slightly (less than 1 percent). Since the RBS is equipped with FCVs, there is no change to the recirculation pump speed due to power uprate; only a slight change in FCV position to achieve the increase in recirculation drive flow. Therefore, there should be no change in vibration for RBS since there is no change from the pre-power-uprate recirculation pump operating speeds. The more open FCV position results in less fluid turbulence, which results in recirculation pump vibration levels that remain constant or decrease with the increase in recirculation drive flow.

Based upon the above, the NRC staff concludes that the recirculation system is acceptable for operation under uprated power conditions.

3.2.5 Reactor Coolant Pressure Boundary (RCPB) Piping

The RCPB piping systems evaluated include the main steam piping, reactor recirculation piping, feedwater piping, RPV bottom head drain line, reactor water cleanup (RWCU), RCIC, core spray piping, high-pressure coolant injection (HPCI) piping, residual heat removal (RHR), SRV discharge piping, and CRD piping. The evaluation included appropriate components, connections, and supports. The licensee's evaluation of the RCBP piping systems consisted of comparing the increase in pressure, temperature, and flow rate against the same parameters in the original design basis analyses. The NRC staff concludes that the methodology used by the licensee is consistent with the NRC-approved methodology in Appendix K of Reference 6, and is therefore acceptable.

As summarized in Reference 8, a majority of the RCPB piping systems were originally designed to maximum temperatures and pressures that bound the increased operating temperature and pressure due to the power uprate. For those systems whose design temperature and pressure did not envelop the uprate power conditions, the licensee performed stress analyses in accordance with the requirements of the Code and the Code addenda of record for the power uprate conditions. The licensee found that the original design analyses have a sufficient margin between calculated stresses and ASME allowable limits to justify operation at the higher operating flow, pressure, and temperature for the proposed power uprate. The maximum stress ratios (ratio of the maximum calculated stresses to the allowable stresses), the maximum CUFs, and the maximum support loads for the most critical RCPB piping systems, such as main steam and recirculation piping, are provided in tables on pages 2 and 3 of Enclosure 2 of Reference 2. The licensee concluded that the evaluation indicated compliance with all appropriate Code requirements for the piping systems evaluated and that the power uprate condition will not have an adverse effect on the reactor coolant piping system design. The NRC staff has reviewed the results of the licensee's analysis and concludes that it is acceptable.

3.2.6 Main Steamline Flow Restrictors

The licensee stated that the power uprate will have no impact on the structural integrity of the main steam flow restrictors. In Section 3.2 of Reference 8, the licensee indicated that a higher peak RPV transient pressure of 1347 psig results from the RBS plant operation at an uprated power of 3039 MWt, but this value remains below the ASME Code limit of 1375 psig. Therefore, the NRC staff concludes that the main steam line flow restrictors will maintain their structural integrity following the power uprate, since the restrictors were designed for a differential pressure of 1375 psig, which envelops the evaluated power uprate conditions.

3.2.7 MSIVs

The MSIVs are part of the RCPB and perform the steamline isolation safety function. The MSIVs must be able to close within the specified design limits at all design and operating conditions upon receipt of a closure signal and are designed to satisfy leakage limits set forth in the RBS TSs. The licensee indicated that the changes in the operating conditions associated with power uprate are small when compared to the original normal operating conditions of pressure in the reactor dome and coolant temperature. The MSIVs are designed to accommodate such small operating changes. The MSIVs have been evaluated for the effects of the changes to the structural capability of the MSIV to meet pressure boundary requirements and the effects of the changes to the safety functions of the MSIV, and were determined to remain acceptable at uprated power.

Based on the review of the licensee's evaluation, the NRC staff concludes that the plant operations at the proposed uprated power level will not affect the ability of the MSIVs to perform their isolation function.

3.2.8 RCIC

The RCIC provides core cooling when the RPV is isolated from the main condenser and the RPV pressure is greater than the maximum allowable for initiation of a low-pressure core cooling system. The RCIC system has been evaluated by the licensee, and its operation is consistent with the bases and conclusions of Reference 7. The pre-power-uprate RCIC system design rated flow of 600 gpm was found to satisfy the core cooling assumptions of the transient analysis under uprated power conditions.

The maximum injection pressure for the RCIC system has been conservatively based on the upper analytical set point for the lowest available group of SRVs in the spring safety mode. For the flow-only increase power uprate, there is no change to the system.

For the power uprate (Phase Two), the reactor dome pressure and the SRV setpoints increase by 30 psi. Consequently, there is a small change to the RCIC high-pressure injection process parameters.

Upon reevaluation, the RCIC system was still found to have the capability to deliver its design flow rate at the increased reactor pressure resulting from the increase in the SRV setpoint pressure and an assumed allowable, as-found, SRV setpoint tolerance of 3 percent. The increase in reactor pressure resulting from these changes increases the maximum required pump operating discharge pressure head from 2980 ft to 3045 ft. In order for the RCIC system to deliver its design flow rate at the higher pump discharge head requirements associated with the power uprate, the maximum specified turbine pump and turbine rated speed was increased from 4550 rpm to 4600 rpm.

The recommendations of GE's Service Information Letter No. 377 (Reference 15) have been implemented at RBS. The licensee used an alternate approach to the control system modification described in Reference 15 for minimizing the effect of reactor pressure on the startup transient response. The RCIC system tests will be conducted during power ascension for power uprate to the new system operating pressure, which is acceptable to the NRC staff. Periodic surveillance testing at slightly higher pressures, combined with infrequent demands for the system to operate under the new high-pressure conditions, should result in an insignificant change in component reliability rates. The reliability of the system will be monitored in accordance with the requirements of the maintenance rule (10 CFR 50.65). The RCIC has been evaluated for loss of feedwater transient events and is consistent with the bases and conclusions of the generic evaluation in Reference 7.

Based upon the above, the NRC staff concludes that the RCIC is acceptable for operation under the conditions associated with the power uprate.

3.2.9 RHR System

The RHR system is designed to restore and maintain the coolant inventory in the reactor vessel and to provide primary system decay heat removal following reactor SDs for both normal and post-accident conditions. The RHR system is designed to operate in the low-pressure coolant injection (LPCI) mode, SD cooling mode, suppression pool cooling mode, containment spray cooling mode, and fuel pool cooling assist mode. The effects of the power uprate on these operating modes are discussed in the following sections.

3.2.9.1 SD Cooling Mode

The operational objective for a normal SD is to reduce the bulk reactor temperature to 125 °F in approximately 20 hours, using two RHR loops. At the uprated power level, the decay heat is increased proportionally, thus slightly increasing the time required to reach the SD temperature. The NRC staff concludes that the additional time to achieve cool down of the reactor is insignificant and will not affect the ability of the RHR system to cool down the reactor.

3.2.9.2 Suppression Pool Cooling Mode (SPCM)

The SPCM of the RHR system is designed to remove heat discharged into the suppression pool to maintain pool temperature below the RBS TS limit during normal plant operation and below the suppression pool design temperature limit of 185 °F after an accident. The power uprate increases the reactor decay heat, which increases the heat input to the suppression pool during a LOCA, which results in a slightly higher peak suppression pool temperature. The power uprate effect on suppression pool cooling after a design basis LOCA remains acceptable as described in Section 3.3.1.1.1.

The functional design basis for SPCM stated in the RBS USAR is to ensure that the pool temperature does not exceed its maximum temperature limit after a blowdown. The NRC staff concludes that this objective is met for the power uprate, since the peak suppression pool temperature analysis by the licensee confirms that the pool temperature will stay below its design limit at uprated conditions.

Based on the review of the licensee's evaluation, the NRC staff concludes that plant operations at the proposed uprated power level will have an insignificant impact on the SPCM.

3.2.9.3 Fuel Pool Cooling Assist Mode

The power uprate has no impact on the fuel pool cooling assist mode since this mode, using the RHR heat removal capacity, provides supplemental fuel pool cooling in the event that the fuel pool heat load exceeds the heat removal capability of the fuel pool cooling and cleanup (FPCC) system due to the offloading of the entire core. This mode is designed to operate along with the FPCC system to maintain spent fuel pool (SFP) temperature within acceptable limits during a reactor cold SD.

In the event that the SFP heat load exceeds the heat removal capability of the SFP cooling system (i.e., during full-core offload events), the RHR system provides supplemental cooling. Heat loads on the RHR system SFP cooling assist mode will increase proportionally to the increase in reactor operating power level. The SFP temperature management evaluation is contained in Section 3.5.3 herein. The licensee performed evaluations and stated that the proposed power uprate has no impact on this mode of RHR system operations.

Based on the review of licensee's evaluation, and the experience gained from NRC staff review of power uprate applications for similar BWR plants, the NRC staff concludes that plant operations at the proposed uprated power level will have an insignificant impact on the RHR system SFP cooling assist mode.

3.2.10 RWCU System

The RWCU system is designed to remove solid and dissolved impurities from the recirculated reactor coolant, thereby reducing the concentration of radioactive and corrosive species in the RCS. System temperature and pressure during operation are not changed at the uprated power level.

The licensee reviewed the RWCU system functional capability. Based on the licensee's experience, the feedwater iron input to the reactor is expected to increase very slightly as a result of the increased feedwater flow. This input increases the reactor water iron concentration; however, this change is not considered significant and does not affect the RWCU system operation.

A slight reduction in the proportion of the RWCU system flow to feedwater flow results in a slightly higher reactor water conductivity because of the increase in feedwater flow without a change in RWCU system flow. The present reactor water conductivity limits are unchanged with the power uprate.

The integrity of the system piping and components was reviewed by the licensee and found to meet all safety and design objectives, including maintaining structural integrity during normal, upset, emergency, and faulted conditions. The NRC staff concludes that the licensee's evaluation is acceptable, and concludes that the RWCU system is capable of performing its function at the uprated power level.

3.2.11 Balance-of-Plant Evaluation (BOP)

The licensee evaluated the stress levels for BOP piping and appropriate components, connections, and supports by evaluating the effect of increasing temperature and pressure from the design basis analysis input. The evaluated BOP systems include lines which are affected by the power uprate but not evaluated in Section 3.5 of Reference 8, such as feedwater heater piping, main steam bypass lines, and portions of the main steam, recirculation, feedwater, RCIC, HPCI, and RHR systems outside the primary containment. The percentage bounding stress increases associated with the increase in pressure, temperature, and flow for affected limiting BOP piping systems were identified in Table 3-7 of the power uprate safety analysis of Reference 2. The limiting stress ratios of the maximum calculated stresses to the allowable stresses, resulting from evaluations of the most critical BOP piping systems for the power uprate, are shown in tables on pages 7 through 11 of Enclosure 2 of Reference 2. The licensee concluded that all piping stresses are below the Code-allowable limits. The NRC staff concludes that the stress ratios calculated by the licensee are within the Code-allowable limits and are, therefore, acceptable.

The licensee evaluated pipe supports such as snubbers, hangers, struts, anchorages, equipment nozzles, guides, and penetrations by evaluating the piping interface loads due to the increases in pressure, temperature, and flow for affected limiting piping systems. The licensee indicated that there is an adequate margin between the original design stresses and Code limits of the supports to accommodate the load increase and, therefore, all evaluated pipe supports were within the Code-allowable limits. The licensee reviewed the original postulated pipe break analysis and concluded that the existing pipe break locations were not affected by the power uprate, and no new pipe break locations were identified. The NRC staff concludes that the licensee's evaluation is acceptable.

3.3 Engineered Safety Features

3.3.1 Containment System Performance

The RBS USAR provides the results of analyses of the containment response to various postulated accidents that constitute the design basis for the containment. Operation with 5 percent power uprate from 2894 MWt to 3039 MWt would change some of the conditions and assumptions of the containment analyses. Section 5.10.2 of Reference 6 requires the power uprate applicant to show the acceptability of the effect of the uprated power on containment capability. These evaluations will include containment pressures and temperatures, LOCA containment dynamic loads, and SRV containment dynamic loads. Appendix G of Reference 6 prescribes the generic approach for this evaluation and outlines the methods and scope of plant-specific containment analyses to be done in support of power uprate. Appendix G of Reference 6 also states that the applicant will analyze short-term containment pressure and temperature response using the GE M3CPT code (current analyses References 16, 17, and 18). These analyses will cover the response through the time of peak drywell pressure throughout the range of power/flow operating conditions with power uprate. A more detailed computer model of the nuclear steam supply system (NSSS) (LAMB, Reference 18) may be used to determine more realistic RPV break flow rates for input to the M3CPT code. The use of the LAMB code has been reviewed and approved by the NRC staff for application to LOCA analysis in accordance with 10 CFR Part 50, Appendix K. The results from these analyses will also be used for input to the LOCA dynamic loads evaluation.

Appendix G of Reference 6 also requires the applicant to perform long-term containment heatup (suppression pool temperature) analyses for the limiting USAR events to show that pool temperatures will remain within limits for suppression pool design temperature, ECCS NPSH, and equipment qualification temperatures. These analyses can be performed using the GE computer code SHEX. The SHEX computer code is partially based on M3CPT and is used to analyze the period from when the break begins until after peak pool heatup (i.e., the long-term response). The SHEX computer code has been used by GE on all BWR power uprates and has been shown to be acceptable based on confirmatory calculations for validation of the results.

3.3.1.1 Containment Pressure and Temperature Response

Short-term and long-term containment analysis results following a large break inside the drywell are documented in the RBS USAR. The short-term analysis was performed to determine the peak drywell and wetwell pressure response during the initial blowdown of the reactor vessel inventory into the containment following a DBA, a large-break LOCA, inside the drywell (DBA LOCA), while the long-term analysis was performed to determine the peak suppression pool temperature response considering decay heat addition.

The licensee indicated that the containment analyses were performed in accordance with Regulatory Guide 1.49 (Reference 19) and Reference 6 using GE codes and models. The M3CPT code was used to model the short-term containment pressure and temperature response. The more detailed RPV model (LAMB) was used for determining the vessel break flow for input to the M3CPT code in the containment analyses to evaluate hydrodynamic loads for power uprate. The use of LAMB model was approved by the NRC staff in Reference 18.

The licensee also indicated that the SHEX code was used to model the long-term containment pressure and temperature response. Based on the NRC staff review of the licensee's evaluation and experience gained from NRC staff review of power uprates for similar BWR plants, the NRC staff concludes that the use of this code is acceptable for the RBS power uprate.

3.3.1.1.1 Long-Term Suppression Pool Temperature Response

The licensee indicated that the long-term bulk suppression pool temperature response was evaluated for the DBA LOCA including the main steam line break (MSLB) and recirculation suction line break (RSLB) LOCA. The bounding analysis was performed at 102 percent of the uprate power using the SHEX code and contains a more realistic decay heat model, using American Nuclear Society/American National Standards Institute (ANS/ANSI) Standard 5.1 (Reference 20), plus two sigma uncertainty, than used in the current RBS USAR analysis. The NRC staff has determined the use of the Reference 20 decay heat model with an added uncertainty of two sigma is acceptable.

The revised long-term containment response analyses were performed at 102 percent of the uprated power level and at 102 percent of the original power level using current methods and decay heat models to show the difference in containment pressure and temperature due to the uprated power. These analyses calculated the peak suppression pool temperature of 170.7 °F at the uprated power level and 168.8 °F at the current power level for the DBA MSLB. The present RBS USAR value for the above case was 167.5 °F with previous methods and decay heat models. The peak calculated suppression pool temperature of 170.7 °F at the uprate power remains below the suppression pool design temperature of 185 °F.

The long-term bulk pool temperature response was also evaluated for the limiting event identified in the RBS USAR, which assumes a transient event with a stuck-open RV with only one RHR heat exchanger available. This event calculated a peak suppression pool temperature of 181.1 °F for the uprated power. The bulk pool temperature for the alternate SD cooling event was also analyzed for the power uprate. This event calculated a peak suppression pool temperature of 183.1 °F. These temperatures remain within the design value of 185 °F.

The licensee indicated that the NPSH for the ECCS (RHR and core spray) pumps are conservatively based on 0 psig containment pressure and a peak post-LOCA suppression pool temperature of 185 °F. Because the peak post-accident suppression pool temperature does not exceed 185 °F, the power uprate does not affect compliance with the ECCS pump NPSH requirements.

Based on the results of these analyses, the NRC staff concludes that the peak bulk suppression pool temperature response remains acceptable from both NPSH and temperature design standpoints for the power uprate.

The local suppression pool temperature limit for SRV discharge is specified in NUREG-0783 (Reference 21) because of concerns resulting from unstable condensation observed at high suppression pool temperatures in plants without quenchers. Elimination of this limit for plants with quenchers on the SRV discharge lines is addressed in Reference 22. In a Safety Evaluation Report dated August 29, 1994, Reference 23, the NRC staff eliminated the maximum local suppression pool temperature limit for plants with quenchers on the SRV discharge lines, provided the ECCS suction strainers are below the quencher elevation. The licensee indicated that the RBS has the ECCS suction strainers below the quenchers, so no evaluation of this limit is necessary. Additionally, the local suppression pool temperature has been evaluated for power uprate and found acceptable.

Based on the review of the licensee's evaluation and the experience gained from NRC staff review of power uprate applications for similar plants, the NRC staff concludes that RBS operation at the uprated power will have no impact on the local pool temperature with SRV discharge.

3.3.1.1.2 Containment Gas Temperature Response

The licensee indicated that the limiting DBA with respect to peak drywell and containment airspace temperatures is the MSLB. The results of the analyses show that the power uprate did not produce significant changes in the peak drywell and containment gas temperatures. The power uprate increases the calculated peak drywell gas temperatures by 0.5 °F for the MSLB and 0.3 °F for the RSLB. The analyses calculated the peak drywell temperature of 332.8 °F at the uprated power level. The peak calculated drywell temperature exceeds the drywell design temperature of 330 °F by 3 °F for less than 1 second and has no adverse effect on the drywell structure during this short duration. The licensee also indicated that the computer program used to calculate the peak temperature does not include the drywell or containment structural passive heat sinks. Based on engineering judgment, if the heat sinks are considered, the peak drywell temperature at the uprated power will remain below the drywell design temperature of 330 °F.

The analyses also calculated the peak long-term containment temperature of 123.8 °F at the uprated power. The peak calculated long-term containment temperature of 123.8 °F at the uprated power remains below the containment design value of 185 °F; therefore, the containment gas temperature response for the power uprate has no adverse effect on the containment structure.

Based on the review of the licensee's evaluation, and the experience gained from NRC staff review of power uprate applications for similar plants, the NRC staff concludes that the drywell and containment air temperature response will remain acceptable after the power uprate.

3.3.1.1.3 Short-Term Containment Pressure Response

The licensee indicated that the short-term containment response analyses were performed for the limiting DBA LOCA, which assumes a double-ended guillotine break of a recirculation suction line or a double-ended guillotine break of a main steam line, to demonstrate that operation at the proposed uprated power level does not result in exceeding the drywell and containment design pressure limits. The short-term analysis covers the blowdown period during which the maximum drywell pressure and maximum differential pressure between the drywell and containment occur. These analyses were performed at 102 percent of the uprated power level using methods reviewed and accepted by the NRC staff. The results of these analyses and sensitivity studies calculated a peak drywell to containment pressure difference of 20.7 psid, a peak suppression pool pressure of 9.3 psig, and a peak containment pressure of 3.6 psig for the power uprate. These pressures remain below the drywell to containment design pressure of 25.0 psid, the suppression pool design pressure of 15.0 psig, and the containment design pressure of 15 psig for RBS. The current value of calculated peak containment pressure, P_a, used for containment testing is 7.6 psig and bounds the peak containment pressure calculated for the power uprate.

Based on the review of the licensee's evaluation and the experience gained from the NRC staff review of power uprate applications for similar BWR plants, the NRC staff concludes that the containment pressure response following a postulated LOCA will remain acceptable under the power uprate conditions.

3.3.1.2 Containment Dynamic Loads

3.3.1.2.1 LOCA Containment Dynamic Loads

The licensee indicated that the LOCA containment dynamic loads for the power uprate are based primarily on the short-term MSLB and RSLB LOCA analyses. Break flows for the RSLB were also calculated using a more detailed RPV model (LAMB). These analyses provide calculated values for the controlling parameters for the dynamic loads throughout the blowdown. The key parameters are the drywell and containment pressures, vent flow rates, and suppression pool temperature. The LOCA dynamic loads which are considered in the power uprate evaluations include pool swell, condensation oscillation (CO), and chugging.

The licensee stated that the short-term containment response conditions with power uprate are within the range of test conditions used to define the suppression pool swell and CO loads for RBS. The long-term response conditions in which chugging would occur with the power uprate are within the conditions used to define the chugging loads; therefore, the LOCA dynamic loads for RBS are not affected by the power uprate.

Based on the review of the licensee's evaluation, and the experience gained from NRC staff review of power uprate applications for similar BWR plants, the NRC staff concludes that the LOCA containment dynamic loads will remain acceptable for the power uprate.

3.3.1.2.2 SRV Loads

The SRVDL loads, suppression pool boundary pressure loads, and drag loads on submerged structures are influenced by the SRV opening setpoint pressure, the initial water leg height in the SRVDL, SRVDL geometry, and suppression pool geometry. Of these parameters, only the SRV setpoint pressure is affected by the power uprate.

The licensee indicated that the SRV opening setpoint which is the basis for the SRV loads on the suppression pool boundary and submerged structure is 1190 psig. The power uprate results in a 3 percent increase in SRV opening setpoint pressure. The licensee stated that an evaluation performed for the combined effect of power uprate and 3 percent tolerance (see Section 3.2.1, herein) results in an increase in the SRV load of less than 2 percent and is well within the conservatism in the SRV loads defined for RBS.

Based on the review of the licensee's evaluation, and the experience gained from NRC staff review of power uprate applications for similar BWR plants, the NRC staff concludes that the plant operation at uprated power will not impact the SRV containment loads.

3.3.1.2.3 Subcompartment Pressurization

The licensee indicated that the actual asymmetrical loads on the RPV, attached piping, and biological shield wall due to a postulated pipe break in the annulus between the RPV and biological shield wall increase slightly due to operation at the higher reactor pressure associated with the power uprate. The biological shield wall and component design remain adequate because the original analyzed loads were based on mass and energy releases which bound the uprated power conditions.

Based on the review of the licensee's evaluation, and the experience gained from NRC staff review of power uprate applications for similar plants, the NRC staff concludes that plant operation at the proposed uprated power level will have an insignificant impact on the subcompartment pressurization.

3.3.1.3 Generic Letter (GL) 89-10 Program

In Reference 8, the licensee indicated that all MOVs used as containment or high-energy line break isolation valves were reviewed and documented in the RBS GL 89-10 (Reference 24) and MOV Program Power Uprate Evaluation Report (Reference 25). In Reference 2, the licensee indicated that 22 out of 182 GL 89-10 Program MOVs are affected by the RBS power uprate. Of these, 12 require revision to their calculation. Ten MOVs will need modifications (four motor replacements, four actuator upgrades, and two gear changes). The licensee provided a list of the affected valves and the basis for the MOV changes required for the uprated condition in Appendix A to Reference 2. The licensee provided followup information in Reference 4. In Reference 4, the licensee committed (1) to completing the modifications discussed in its April 3, 2000, response before the Phase Two power uprate, and (2) to revising MOV calculations to include consideration of the updated guidance in Limitorque Technical Update 98-01 (Reference 26) for predicting MOV motor actuator output. The NRC staff concludes that the licensee's evaluation is adequate to ensure satisfactory performance of its MOVs for the power uprate at RBS.

3.3.1.4 GL 96-06

The licensee reviewed the plant-specific information on RBS systems and components for the power uprate to determine its potential effect on the performance of mechanical components as addressed in GL 96-06 (Reference 27). The licensee concluded that the operability of all mechanical components such as heat exchangers, pumps, and valves was confirmed at the power uprate condition. The licensee also concluded that the proposed power uprate conditions are bounded by the current containment design temperature and pressure and thus, have no impact on the evaluation in response to GL 96-06 (Reference 28) on potential over-pressurization of isolated piping segments for RBS. The NRC staff concludes that the licensee's evaluation is acceptable.

3.3.2 ECCS

3.3.2.1 HPCS

The HPCS system was evaluated by the licensee and its operation is consistent with the bases and conclusions contained in Reference 7. The maximum injection pressure for the HPCS has been conservatively based on the lowest available group of SRVs.

The system was found to have the capability to deliver its design rated flow at the lower reactor pressure expected for the Phase Two operation. The pump flow rate is verified at a pump differential pressure that is sufficient to overcome the RPV pressure expected during a LOCA. Consequently there is no change in the HPCS pump surveillance test pressure for the power uprate. Since there is no increase of reactor operating pressure and no increase in the SRV set points, there is no impact on the HPCS during Phase One operation.

The system was found to have the capability to deliver its design rated flow at the increased reactor pressure resulting from the increase in the SRV setpoint pressure for Phase Two operation and the allowable, as-found SRV setpoint tolerance of 3 percent. The increase in reactor pressure resulting from these changes increases the maximum system operating head from 2886 feet to 3012 feet. The HPCS pump has the capability to deliver its design flow at the higher reactor pressure associated with power uprate. The pump flow rate is verified at a pump differential pressure that is sufficient to overcome the RPV pressure expected during a LOCA. Consequently there is no change in the HPCS pump surveillance test pressure for the power uprate, which the NRC staff concludes is acceptable.

3.3.2.2 LPCI mode of RHR

The low-pressure portions of the RHR system are not affected by power uprate. The upper limit of the low-pressure ECCS injection set-points will not be changed for the power uprate, and, therefore, the low-pressure portions of these systems will not experience any higher pressures. The design basis flow rates of the low-pressure ECCS will not be increased. In addition, the RHR system SD cooling mode flow rates and operating pressures will not be increased; therefore, since the system does not experience different operating conditions due to power uprate, there is no impact due to power uprate. This is consistent with the bases and conclusions of Reference 7, which the NRC concludes is acceptable.

3.3.2.3 Low-pressure Core Spray (LPCS) System

The LPCS system is not affected by the power uprate. The upper limit of the LPCS injection set points will not be changed for the power uprate; therefore the low-pressure portions of this system will not experience any higher pressures. The design basis flow rates of the low-pressure ECCS will not be increased; therefore, this system does not experience different operating conditions due to the power uprate. In addition, the impact of the power uprate on the long term response to a LOCA will continue to be bounded by the short-term response. Operation of the LPCS is bounded by the generic evaluation (Reference 7), which the NRC staff concludes is acceptable.

3.3.2.4 ADS

The ADS uses SRVs to reduce reactor pressure following a small break LOCA with HPCS failure. This function allows LPCI and LPCS to inject into the RPV. The ADS initiation logic and ADS valve controls are adequate for the power uprate. ADS initiates on Low Water Level 1 and a signal that at least one LPCI or LPCS pump is running with permissive from Low Water Level 3. If these conditions are met ADS is activated following a maximum time delay of 120 seconds after the initiating signals. The NRC staff concludes that the ability of the ADS to perform its safety function is not affected by the power uprate.

3.3.3 ECCS Performance Evaluation

The ECCSs are designed to provide protection against hypothetical LOCAs caused by ruptures in the primary system piping. The ECCS performance under all LOCA conditions and their analysis models satisfy the requirements of 10 CFR 50.46 and 10 CFR Part 50, Appendix K. The GE fuel used in RBS was analyzed by the licensee (Reference 29) with the NRC-approved methods. The results of the ECCS-LOCA analysis using NRC-approved methods and input parameters that bound the power uprate conditions are provided in Reference 29 . The SAFER/GESTAR-LOCA methodology is the current analysis of record for the RBS. Cycle-specific analyses will be done at each reload and will be a part of the COLR developed by the licensee, per RBS TS 5.6.6, which the NRC staff concludes is acceptable.

3.3.4 Main Control Room Atmosphere Control System

The licensee indicated that this system is not significantly affected by the power uprate. The impact of the power uprate is insignificant with regard to the radiological loading on the main control room filters. For the power uprate, the control room doses resulting from a LOCA would be less than the 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19 limits (see Section 3.8.2, herein.)

Based on the review of the licensee's evaluation and the experience gained from NRC staff review of power uprate applications for similar plants, the NRC staff concludes that plant operations at the proposed uprated power will have an insignificant impact on the main control room atmosphere control system.

3.3.5 Standby Gas Treatment System (SGTS)

The SGTS is designed to minimize offsite dose rates during venting and purging of both the primary and secondary containment atmosphere under accident or abnormal conditions, while air-borne particulates and halogens might be present. The licensee indicated that the capacity of the SGTS is adequate to maintain the reactor building at a slightly negative pressure; this capability is not affected by the power uprate. The power uprate has an insignificant effect on the charcoal filter beds. The post-LOCA total iodine loading at uprated power conditions remains well below the original design capability of the filter.

Based on the review of the licensee's evaluation and the experience gained from NRC staff review of power uprate applications for similar BWR plants, the NRC staff concludes that the plant operations at the proposed uprated power level will have an insignificant impact on the SGTS.

3.3.6 Main Steam Positive Leakage Control System (MS-PLCS)

The licensee indicated that the MS-PLCS will not be affected by the power uprate since the peak post-LOCA containment pressure does not increase beyond the design basis.

The NRC staff concludes that the MS-PLCS safety function will not be affected by the power uprate.

3.3.7 Post-LOCA Combustible Gas Control System

The combustible gas control system is designed to maintain the drywell and containment atmospheres as a noncombustible mixture after a LOCA. The combustibility of the post-LOCA atmosphere is controlled by the concentration of hydrogen. The licensee indicated that the post-LOCA production of hydrogen and oxygen from radiolysis will increase in proportion to the power level. Sufficient capacity exists in the combustible gas control system to accommodate the increased oxygen production. The increase in hydrogen generation due to the power uprate has a minor impact on the time available to start the system before reaching procedurally controlled limits, but does not impact the ability of the system to maintain hydrogen below the lower flammability limit, of 4 percent volume, following a LOCA. The predicted start time of the recombiners decreases from 14 days at preuprate power to 12.5 days at uprated power. The above timing change does not affect the ability of the operator to take action. Power uprate has no impact on the recombiner maximum operating temperature, which is dependent only on the containment hydrogen concentration when the recombiners are started.

Based on the review of the licensee's evaluation, and the experience gained from NRC staff review of power uprate applications for similar BWR plants, the NRC staff concludes that plant operations at the proposed uprated power level will have an insignificant impact on the post-LOCA combustible gas control system and the system will remain acceptable.

3.4 Instrumentation and Control

In Reference 8, GE stated that most BWR plants, as originally licensed, have an assigned equipment and system capability to accommodate steam flow rates at least 5 percent above the original rating. In addition, improvements in analytical techniques, plant performance feedback, fuel, and core designs have resulted in a significant increase in the difference between the calculated safety analysis results and the licensing limits. GE also stated that most GE BWR plants have the capability and margins for an uprating of 5 percent to 20 percent without major NSSS hardware modifications. GE further stated that the Reference 8 analyses are based on the guidelines and evaluations provided in References 6 and 7.

In addition, GE used GE Nuclear Energy Topical Report NEDE-31336-P (Reference 30) with vendor-supplied accuracy values, site measured drift values, and site-specific design and environmental data to generate the allowable values and trip setpoints. Each setpoint has been selected with sufficient margin between the setpoint and the analytical limit to preclude inadvertent operation of the protective system while assuring adequate allowances for instrument accuracy, calibration, and drift.

GE also stated in Reference 8 that prior to operation at the uprated power level, the RBS will be subjected to power uprate testing to Section 5.11.9 and Appendix L, Section L.2, of Reference 6. The power uprate testing will include surveillance testing of all instrumentation that requires recalibration, evaluation of steady-state data from 90 percent to previous rated thermal power and steady-state data for power increase beyond the previous rating at increments of approximately less than or equal to 3 percent power. These tests will be specifically conducted for Intermediate range monitors, average power range monitors, pressure regulator system, feedwater control system, recirculation flow control, recirculation flow, and radiation measurements (see Section 3.9.4 herein).

Based on NRC staff review of the licensee's evaluation and the NRC staff review of power uprate applications for similar BWR plants, the NRC staff concludes that the proposed power uprate will have no significant impact on the instrumentation and control systems.

3.5 Electrical Power and Auxiliary Systems

3.5.1 Station Auxiliary Electric Power Distribution System

In Reference 2, the licensee stated that the onsite power distribution system loads were reviewed under both normal and emergency operating scenarios. In both cases loads are computed based on equipment nameplate data or brake horsepower (BHP). These loads are used as inputs for the computation of load, voltage drop, and short circuit current values. Operations at the uprate power level is achieved in both normal and emergency conditions by operating equipment at or below the nameplate rating running kW or BHP; therefore, the NRC staff concludes that there are no changes to the load, voltage drop or short circuit current values. Also, the amount of power required to perform safety-related functions will not be increased with the power uprate, and the current emergency power systems remain adequate.

The dc power distribution system loads were reviewed like the onsite power distribution system. The licensee states that there are no changes to the load, voltage drop, or short circuit current values. Operation at the uprated power level will not increase any loads beyond nameplate rating or revise any control logic; therefore, the NRC staff concludes that the dc power distribution system is adequate.

3.5.2 Grid Stability Analysis

In Reference 2, the licensee stated that a grid stability analysis has been performed, considering the increase in electrical output, to demonstrate conformance to 10 CFR Part 50, Appendix A, GDC 17, for the RBS. Both steady-state and transient analyses were performed.

The steady-state analysis determined the effect of the RBS power uprate on the Fancy Point 230 kilovolts (kV) bus voltage. The steady-state electrical analysis of the power uprate conditions was performed at 1130 megawatts electrical (MWe). The 1130 MWe is a bounding condition for the analysis as it exceeds the actual uprated main generator electrical output of 1043.1 MWe. The analysis examined both load conditions (the 2000 summer peak was the maximum loading condition for load growth and loading level 100 percent, while the 1999 spring light case was the minimum loading condition for load growth and loading level 40 percent) for the pre-upgrade RBS power output (990 MWe) and post-uprate RBS power output (1130 MWe). Normal system conditions, as well as contingency conditions, were evaluated.

The contingency list was based on the contingencies listed in the RBS USAR, and revised to incorporate the circuit breaker arrangements at the Fancy Point 230/500 kV and adjacent substations. The steady-state contingency list includes six generator trips, six single line trips, three multiple element trips, one load trip, and one LOCA contingency. The LOCA contingency consists of loss of the RBS generator and switching out loads at busses SWG4A and SWG4B, respectively, and switching in motor loads at safety-related busses SWG*1A, SWG*1B, and E22*S004. In order to simulate the worst-case LOCA scenario, the largest motors SWG*1A and SWG*1B were conservatively modeled in the locked-rotor condition. The results show that the RBS power uprate has a negligible effect on post-contingency steady-state voltage at the Fancy Point 230 kV bus. In all simulations, including the LOCA contingency, the Fancy Point 230 kV bus voltage remained at approximately 1.02 per unit (pu), bus voltage at RBS emergency busses remained above 0.90 pu (above the relay settings for loss of voltage and degraded voltage relays). The steady-state analysis shows that the upgrade of the RBS plant from 990 MWe to 1130 MWe has a little impact on Fancy Point 230 kV bus voltage.

For the transient analysis the same contingencies were used. The transient analysis results show stable performance with RBS power output level of 990 MWe for all contingencies under both 1999 spring light load and 2002 spring load conditions. At an RBS level of 1100 MWe, all contingencies under both 1999 spring light load and 2002 summer peak load conditions demonstrate stable performance.

Based upon the above evaluations, the NRC staff concludes that the proposed power uprate at RBS will have minimal impact on the grid stability.

3.5.3 Fuel Pool Cooling

The SFP cooling and cleanup (SFPCC) system is designed to remove the decay heat from the spent fuel assemblies stored in the SFP, and to clarify and purify the water in the SFP. The SFP cooling portion of the SFPCC system consists of two 100 percent capacity cooling trains, each primarily equipped with one pump, one heat exchanger, and its associated valves, piping, instrumentation, and controls. Heat is removed from the SFP heat exchanger by the service water system (SWS). In addition, the RHR system, which has a higher heat removal capacity and serves as a backup system to the SFPCC system, provides supplemental cooling to maintain the SFP temperature within acceptable limits in the event that the SFP heat load exceeds the heat removal capability of the SFPCC system.

As a result of plant operations at the proposed uprated power level, the decay heat load for any specific fuel discharge scenario will increase slightly. The licensee performed evaluations and showed that the maximum heat load in the SFP for power uprate increases, but is still below the preuprate design basis heat load for the SFP. The combination of the SFPCC system heat exchangers and the availability of the RHR system is sufficient to remove the decay heat during a planned[1] refueling outage or an unplanned full core offload event. Therefore, the licensee concluded that plant operations at the proposed uprated power level will not have any negative effect on the capability of the SFPCC system or the fuel pool cooling assist mode of the RHR system.

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1 In Reference 2, the licensee stated that at the RBS a full core offload is not a normal practice during refueling outages.

Based on the review of licensee's evaluations, and the experience gained from NRC staff review of power uprate applications for similar BWR plants, the NRC staff concludes that plant operations at the proposed uprate power level will have no significant impact on the SFP cooling system and the RHR system in the fuel pool cooling assist mode.

3.5.4 Water Systems

3.5.4.1 Service Water Systems

The service water systems are designed to provide cooling water to various systems (both safety-related and nonsafety-related).

3.5.4.1.1 Safety-Related Loads (Safety-Related Standby Water (SSW) System)

The SSW system provides a reliable supply of cooling water during and following a DBA for the following components and systems: RHR heat exchangers, RHR pump seal coolers, emergency diesel generator heat exchangers, HPCS diesel generator heat exchangers, control room air conditioned (AC) water chillers, containment unit coolers, penetration valve leakage control compressors, auxiliary unit coolers, and SFP heat exchangers (if needed). The safety-related performance of the SWS to provide cooling water for these components and systems during and following the DBA is not significantly dependent on the reactor rated power. The licensee performed evaluations and stated that plant operations at the proposed uprated power level will have an insignificant impact on the SSW system and that the SSW system has sufficient capacity to remove the increased heat loads due to power uprate; therefore, the licensee concluded that plant operation at the proposed uprated power level does not require the modification of the SWS for the safety-related loads.

Based on the review of the licensee's evaluation and the experience gained from NRC staff review of power uprate applications for similar BWR plants, the NRC staff concludes that plant operations at the proposed power uprate level have an insignificant impact on the SSW system regarding the safety-related loads.

3.5.4.1.2 Non-Safety-Related Loads (Non-Safety Related Service Water/Cooling Systems)

The non-safety-related service water/cooling systems are designed to provide cooling water to various plant equipment during normal plant operation and SD periods. The increase in heat loads on these systems due to uprated operation is approximately proportional to the power uprate. The licensee performed evaluations and demonstrated that the increase in heat loads on these systems is insignificant and that these systems have sufficient cooling capacities for plant operations at the proposed power uprate level.

Based on the review of the licensee's evaluation and the experience gained from NRC staff review of power uprate applications for similar BWR plants, the NRC staff concludes that plant operations at the proposed uprated power level have an insignificant impact on the non-safety-related service water/cooling systems.

3.5.4.2 Main Condenser, Circulating Water, and Normal Heat Sink Performance

The main condenser, circulating water, and normal heat sink (cooling tower) systems are designed to provide the main condenser with a continuous supply of cooling water for removing heat rejected to the condenser by turbine exhaust, turbine bypass steam, and other exhausts over the full range of operating loads, thereby maintaining low condenser pressure as recommended by the turbine vendor. The licensee stated that the performance of the main condenser, circulating water, and normal heat sink systems was evaluated and found adequate for plant operations at the proposed uprated power level.

Since the main condenser, circulating water, and normal heat sink systems do not perform or support any safety-related function, the impact of the proposed uprated power operations on the designs and performances of these systems was not reviewed by the NRC staff.

3.5.4.2.1 Discharge Limits

The licensee compared the State discharge limits to current discharges and bounding analysis discharges for power update. The comparison demonstrates that the plant will remain within the State discharge limits during operations at the uprated power level.

Based on the review of the licensee's comparison of the State discharge limits to current discharges and bounding analysis discharges for power uprate, and the experience gained from NRC staff review of power uprate applications for similar BWR plants, the NRC staff concludes that the plant will remain within the State discharge limits during operations at uprated power level.

3.5.4.3 Reactor Plant Component Cooling Water System (RPCCW) System

The RPCCW system is designed to remove heat from various auxiliary plant equipment housed in the reactor building. The licensee performed evaluations and stated that the increase in heat loads on this system due to uprated power operations is insignificant and that the RPCCW heat exchangers were conservatively designed for heat loads which bound those anticipated for both normal and accident conditions at the uprated power level.

Based on the review of the licensee's evaluation and the experience gained from NRC staff review of power uprate applications for similar BWR plants, the NRC staff concludes that plant operations at the proposed uprated power level will have an insignificant impact on the RPCCW.

3.5.4.4 Turbine Plant Component Cooling Water (TPCCW) System

The TPCCW system supplies cooling water to auxiliary plant equipment in the turbine building. The licensee stated that the TPCCW system heat load increases due to power uprate are those related to the operation of the turbine-generator, and that the TPCCW system has adequate heat removal capability for plant operations at the proposed uprated power level.

Since the TPCCW system does not perform or support any safety-related function, the impact of the proposed uprated power operations on the designs and performances of this system was not reviewed by the NRC staff.

3.5.4.5 Ultimate Heat Sink (UHS)

The UHS for the RBS is the standby service water cooling tower, which functions as both the supply and return for the SSW system. The licensee performed an evaluation and stated that the post-LOCA UHS water temperature does not change as a result of operation at the uprated power level.

Based on the review of the licensee's evaluation and the experience gained from NRC staff review of power uprate applications for similar BWR plants, the NRC staff concludes that plant operations at the proposed uprated power level will have an insignificant impact on the UHS.

3.5.5 SLCS

The SLCS is designed to shut down the reactor from rated power to cold SD, assuming that all of the control rods cannot be fully inserted. It is a manually operated system that injects sodium pentaborate solution into the RPV in order to provide neutron absorption and achieve a subcritical reactor condition. The SLCS SD capability is reevaluated for each reload core. The SLCS is designed for injection at a reactor pressure equal to the upper analytical setpoint for the lowest available group of SRVs operating in the relief mode.

Since there is no increase in reactor operating pressure for Phase One operation, no increase in injection pressure is required for the system; therefore, system operation is unaffected.

For Phase Two operation, the reactor dome pressure and the SRV set points will increase by 30 psi. Consequently, the maximum required pump discharge pressure will increase from 1217 psig to 1247 psig. The licensee determined that the system has the capability to deliver its design rated flow at this increased operating pressure. The SLCS pumps are positive displacement pumps, and small pressure changes in the SRV set points have little effect on the rated flow to the reactor. Even with the resulting increase in operating pressure, there is adequate pressure margin for the SLCS pump RV; therefore, the SLCS ability to provide its backup SD function is not affected by the power uprate.

The SLCS surveillance test pressure is based on the maximum SLCS injection pressure required for injection at the SRV setpoint pressure, including allowances for system test inaccuracies. Because the SRV set points have been increased by 30 psi, the SLCS surveillance test pressure is increased by 30 psi from the pre-power-uprate value of 1220 psig to 1250 psig.

The licensee has increased the RBS boron concentration and enrichment in the SLCS to meet the anticipated transients without scram (ATWS) rule (10 CFR 50.62). The SLCS ATWS performance is evaluated in Section 3.8.3.1 herein for a representative core design at uprated power.

Based on the review of the licensee's evaluation and the experience gained from NRC staff review of power uprate applications for similar BWR plants, the NRC staff concludes that plant operations at the proposed uprated power level will have an insignificant impact on the SLCS.

3.5.6 Power-Dependent Heating, Ventilation and Air Conditioning (HVAC)

The licensee indicated that HVAC systems affected by power uprate are those systems that service the turbine building, auxiliary building, fuel building, reactor building, steam tunnel, and the containment drywell. The power uprate is expected to result in slightly higher process temperatures and a small increase in the heat load due to higher electrical currents in some motors and cables. All steam cycle process temperatures increase less than 4 °F from current plant operation due to the power uprate. This includes main steam, feedwater, condensate, extraction steam, and heater drains. There is sufficient capacity in the existing HVAC systems to accommodate the increased heat gain in these associated buildings and therefore, the area temperatures do not increase. Other areas are unaffected by the power uprate because the process temperatures remain relatively constant. Heat gain from electrical loads was conservatively accounted for in the original design by assuming that all cables in an area were energized and that all electrical equipment operating at rated/nameplate load; therefore, design of the HVAC is not adversely affected by the power uprate.

Based on the review of licensee's evaluation and the experience gained from NRC staff review of power uprate applications for similar BWR plants, the NRC staff concludes that plant operations at the proposed uprated power level will have an insignificant impact on the HVAC systems.

3.5.7 Fire Protection

Fire detection and suppression systems are not expected to be impacted by plant operations at the proposed uprated power level since there are no physical plant configurations or combustible load changes resulting from the uprated power operations. In addition, the safe SD systems and equipment used to achieve and maintain cold SD conditions do not change for the uprated conditions, and the operator actions required to mitigate the consequences of a fire are not affected. The licensee concluded that plant operation at the proposed uprated power level does not affect the ability of the Appendix R systems to perform their safe SD function.

Based on NRC staff review of the licensee's evaluation and the experience gained from NRC staff review of power uprate applications for similar BWR plants, the NRC staff concludes that the fire detection and suppression systems and post-fire safe SD capability will not be affected by plant operations at the proposed uprated power level.

3.5.8 Systems Not Impacted or Insignificantly Impacted by Power Uprate

The licensee identified other systems which are not affected or insignificantly affected by plant operations at the proposed uprated power level. The NRC has review those systems (i.e., auxiliary steam, compressed air, service air, miscellaneous HVAC, diesel generator and its associated supporting systems) and agrees with the licensee that plant operations at the proposed uprated power level has no impact or insignificant impact on these systems.

3.6 Power Conversion Systems

3.6.1 Turbine-Generator

The licensee performed evaluations for turbine operations with respect to design acceptance criteria to verify the mechanical integrity under the conditions imposed by the power uprate. Results of the evaluations showed that there would be no increase in the probability of turbine overspeed and of turbine missile production due to plant operations at the proposed uprated power level; therefore, the licensee concluded that no change in the turbine overspeed trip setting is required.

The isophase bus rating, the main transformer ratings, the unit auxiliary transformer rating, and the switchyard components are adequate for the uprated power; therefore, the turbine-generator and major electrical components remain adequate at the uprated power level.

Based on the review of the licensee's evaluation, the NRC staff concludes that the turbine-generator will continue to operate safely at the proposed uprated power levels and that operation of the turbine-generator at the proposed uprated power level is acceptable.

3.6.2 Miscellaneous Power Conversion Systems

The licensee evaluated the miscellaneous steam and power conversion systems and their associated components (including the condenser air removal and steam jet air ejectors, turbine steam bypass, and feedwater and condensate systems) for plant operations at the proposed uprated power level. The licensee stated that the existing equipment for these systems are acceptable for plant operations at the proposed uprated power level.

Since these systems do not perform or support any safety-related function, the impact of plant operations at the proposed uprated power level on the design and performance of these systems was not reviewed by the NRC staff.

3.7 Radioactive Waste (Radwaste) Systems and Radiation Sources

3.7.1 Liquid Waste Management

The single largest source of liquid waste is from the backwash of the condensate demineralizers. The licensee stated that with power uprate, the average time between backwash/precoat will be reduced slightly. This reduction does not affect plant safety. The licensee further stated that the activated corrosion products in liquid wastes are expected to increase proportionally to the power uprate; however, the total volume of processed waste is not expected to increase appreciably, since the only significant increase in processed waste is due to the more frequent backwash of the condensate demineralizers. The licensee performed evaluations of plant operations and effluent reports, and concluded that the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I, will continue to be satisfied with regard to liquid waste processing.

Based on the review of the licensee's evaluation and the experience gained from NRC staff review of power uprate applications for similar BWR plants, the NRC staff concludes that the liquid radwaste system is acceptable for operation under power uprate conditions.

3.7.2 Gaseous Waste Management

Gaseous wastes generated during normal and abnormal operation are collected, controlled, processed, stored, and released utilizing the gaseous waste processing treatment systems. These systems, which are designed to meet the requirements of 10 CFR Part 20 and 10 CFR Part 50, Appendix I, include the offgas system and SGTS, as well as other building ventilation systems. With regards to the offgas system, core radiolysis increases linearly with core thermal power, thus increasing the heat load on the recombiner and related components. The licensee performed an evaluation and stated that the operational increase in hydrogen flow rate due to power uprate remains well within the design capacity of the system. The system radiological release is administratively controlled and is not changed with operating power; therefore, the licensee concluded that power uprate does not affect the offgas system design or operation. Various devices and processes, such as radiation monitors, filters, isolation dampers, and fans are used to control airborne radioactive gases. The results of the licensee's analyses demonstrate that airborne effluent activity released through building vents is not expected to increase significantly due to plant operations at the proposed uprated power level.

Based on the review of the licensee's evaluation and the experience gained from the NRC staff review of power uprate applications for similar BWR plants, the NRC staff concludes that plant operations at the proposed uprated power level will have an insignificant impact on the gaseous waste management systems.

3.7.2.1 Offgas System

Core radiolysis increases linearly with core thermal power, thus increasing the heat load on the recombiner and related components. The licensee performed an evaluation and stated that the operational increase in hydrogen flow rate due to power uprate remains well within the design capacity of the system. The system radiological release is administratively controlled and is not changed with operating power. Therefore, the licensee concluded that power uprate does not affect the offgas system design or operation.

Based on the review of the licensee's rationale and evaluation, and the experience gained from review of power uprate applications for similar BWR plants, we agree with the licensee's conclusion that plant operations at the proposed uprated power level will have an insignificant impact on the offgas system.

3.7.3 Radiation Sources and Plant Dose Assessment

The NRC staff has reviewed the proposed RBS power uprate with respect to its effect on the facility radiation levels and on the radiation sources in the core and coolant. The radiation sources in the core include radiation from the fission process, accumulated fission products, and neutron reactions as a secondary result of reactor power. The radiation sources in the core are expected to increase in proportion to the increase in power. This increase, however, is bounded by the built-in safety margins of the design basis sources. Since the reactor vessel is inaccessible during operation, a 5 percent increase in the radiation sources in the reactor core will have no effect on personnel doses.

During operations, the reactor coolant passing through the reactor core region becomes radioactive as a result of nuclear reactions. The activation products in the steam will remain nearly constant following the power uprate, since the increase in activation production in the steam passing through the core will be balanced by the increase in steam flow through the core. The activation products in the reactor water, however, will increase in approximate proportion to the increase in thermal power. The shielding at RBS was conservatively designed so that this small percent increase in activation products in the reactor coolant resulting from the proposed power uprate will not affect radiation zoning in the plant.

Activated corrosion products, which are the result of the activation of metallic wear materials in the reactor coolant, could increase by as much as 10 percent (proportional to the square of the power level increase) as a result of the proposed 5 percent power uprate. The equilibrium level of activated corrosion products in the reactor coolant is expected to increase in proportion to both the increase in feedwater flow rate and the increase in neutron flux in the reactor. As indicated in Reference 5, most of the areas that would be affected by this increase in activated corrosion products (e.g., recirculation system and the RWCU system) are located in locked areas or areas, such as the containment drywell, that are inaccessible during plant operation. Since these areas are usually high dose rate areas, personnel access to these areas will be restricted during plant operations.

Fission products in the reactor coolant result from the escape of minute fractions of the fission products which are contained in the fuel rods. Since the current fuel thermal limits (which affect the rate of release of fission products from the fuel rods) will be maintained for the proposed power uprate, there will be no change in the amounts of fission products released to the reactor coolant from the fuel. Therefore, the fission product activity levels in the steam and reactor water are expected to be approximately equal to current measured data. The current levels of fission product activity in the reactor coolant are fractions of the design basis data; therefore, the design basis data are unchanged and are used for power uprate.

Radiation sources in the coolant contribute to the plant radiation levels. As discussed above, the proposed 5 percent power uprate will result in a proportional increase in certain radiation sources in the reactor coolant. This increase in reactor coolant activity will result in a similar percentage increase in plant radiation levels in most areas of the plant. This increase in plant radiation levels may be higher in certain areas of the plant (e.g., inside the containment drywell and near the RWCU system) due to the presence of activated corrosion products. Some post-operational radiation levels may also be higher in those areas of the plant where accumulation of corrosion product crud is expected (i.e., near the SFP cooling system piping and the reactor water piping, as well as near some liquid radwaste equipment). Many of these areas are normally locked and require infrequent access.

The licensee has stated that many aspects of the plant were originally designed for higher-than-expected radiation sources; therefore, the small potential increase in radiation levels resulting from the proposed power uprate will not affect radiation zoning or shielding in the various areas of the plant that may experience higher radiation levels. The purpose of the licensee's "as low as is reasonably achievable" (ALARA) radiation program is to ensure that doses to individual workers will be maintained within acceptable limits by controlling access to radiation areas. The licensee will use procedural controls to compensate for any increased radiation levels.

On the basis of the NRC staff review of proposed RBS power uprate, the NRC staff concludes that the 5 percent power uprate will have an insignificant effect on personnel doses, and that the doses to workers will be maintained ALARA in accordance with the requirements of 10 CFR 20.1101.

3.8 Reactor Safety Performance Evaluations

3.8.1 Reactor Transients

Reload licensing analyses evaluate the limiting plant transients. Disturbances of the plant caused by a malfunction, a single failure of equipment, or personnel error are investigated according to the type of initiating event. The licensee utilizes its NRC-approved licensing analysis methodology to calculate the effects of the limiting reactor transients. The limiting events for RBS were identified. These are the same as those in Reference 7. The generic guidelines (Reference 6) also identified the analytical methods, the operating conditions that are to be assumed, and the criteria that are to be applied. Representative changes in core critical power ratios were analyzed; however, specific core OLs will be supplied for each specific fuel cycle. The power uprate analyses were presented for a representative core using the GEMINI transient analysis methodology (Reference 31). The operating conditions that apply most directly to the transient analysis are summarized in Table 9-1 of Reference 8. They are compared to the conditions used for the USAR and the most recent reload fuel cycle (Cycle 7) analyses. The Cycle 7 core was used as the representative fuel cycle for power uprate. Most of the transients were analyzed at the full uprated power and maximum allowed core flow operating point on the power/flow map. Direct or statistical allowance for power uncertainty is included in the analysis. The effect of the power uprate on the SLMCPR will be confirmed for each operating fuel cycle, at the time of the reload analysis, using the NRC-approved methodology. The effect of the power uprate on the SLMCPR is generically evaluated in Reference 7.

The limiting transients for each category were analyzed to determine their sensitivity to core flow, feedwater temperature, and cycle exposure. The licensing basis for transient analyses at the uprated power level were developed from these results. The limiting transient results were presented in the licensee submittal (Reference 8) in Table 9-2, and Figures 9-1 through 9-4. These were the applicable transients as specified in Reference 6. No changes to the basic characteristics of any of the limiting events are caused by power uprate. Cycle-specific analyses will be done at each reload and will be a part of the COLR developed by the licensee, which is acceptable to the NRC staff.

3.8.2 DBAs

In Section 9.2 of Attachment 7 to Reference 8 and in Reference 5, the licensee evaluated the radiological consequences of three postulated DBAs. The analyzed DBAs are (1) LOCA, (2) fuel-handling accident (FHA), and (3) control rod drop accident. The licensee stated in the amendment request that the MSLB accident outside containment need not be reanalyzed because the mass release rates from the postulated MSLB are bounded by those in the current analysis, which NRC staff concludes is acceptable. The licensee concluded in the amendment request that the radiological consequences of a DBA, subsequent to implementation of the RBS power uprate, remain well below the dose criteria specified in 10 CFR Part 100 with regard to offsite doses and GDC 19 of 10 CFR Part 50, Appendix A, with regard to control room personnel doses.

The licensee analyzed DBA radiological consequences in the current RBS updated final safety analysis report (UFSAR) for a maximum power level of 3039 MWt. This power level corresponds to 105 percent of the currently licensed power level of 2894 MWt. For the RBS power uprate, the NRC staff performed confirmatory analyses and assumed a reactor core power level of 3100 MWt for the radiological consequence analyses, which is equal to 1.02 times the proposed, uprated, reactor power level of 3039 MWt. This assumption allows for possible instrument errors in determining the reactor power level and accounts for the margin in the turbine-generator design above rated capacity, as described in Reference 19. The NRC staff's fission product release and removal models predict that the radiological consequences of an accident subsequent to a power uprate will be increased in approximate proportion to the increase in reactor power level.

The licensee submitted revised radiological consequence analyses for the exclusion area boundary (EAB), low-population zone (LPZ), and the control room for three postulated DBAs, stated above, at a reactor power level of 3100 MWt. The analyses showed that the resulting radiological doses at the proposed uprated reactor core power level are still well within the relevant dose criteria provided in 10 CFR Part 100 and GDC 19. To verify the licensee's analyses, the NRC staff performed an independent radiological consequence calculation and concluded that the licensee's calculated doses are consistent with those calculated by the NRC staff. With the same parameters and assumptions used for the radiological dose calculations, the NRC staff's resulting doses increased by approximately 2 percent with the proposed power uprate. The radiological doses calculated by the NRC staff at a reactor core power level of 3100 MWt are given in Table 1, and the major parameters and assumptions used for the dose calculations are given in Tables 2 and 3.

The NRC staff's analyses also showed that the resulting radiological doses at the proposed uprated reactor core power level are still well within the relevant dose criteria provided in 10 CFR Part 100 and GDC 19; therefore, the NRC staff concludes that the proposed increase of reactor core power level to be acceptable with regard to these post-accident doses.

Table 1

Radiological Consequences of DBAs at the Uprated Reactor Core Power Level (3100 MWt) (rem)

	EAB[1]		LPZ[2]		Control Room	
Postulated Accidents	Thyroid	WB[3]	Thyroid	WB	Thyroid	WB
Loss-of-Coolant	85	4	118	2	3	<1
Dose Acceptance Criteria[4]	300	25	300	25	30	5
FHA	65	0.1	8.5	0.1	1.0	<0.1
Dose Acceptance Criteria[5]	75	6	75	6	30	5
Control Rod Drop Accident[5]	4.5	0.6	3.7	0.2	0.5	<0.1
Dose Acceptance Criteria	75	6	75	6	30	5

[1] Exclusion area boundary
[2] Low-population zone
[3] Whole body
[4] 10 CFR Part 100
[5] Standard Review Plan

Table 2

Assumptions Used to Evaluate Radiological Consequence

LOCA

Parameter	Value
Reactor Power MWt	3100
Fraction of core inventory released, fractions	
Noble gases	1.0
Iodine	0.5
Iodine chemical forms, fractions	
Organic	0.04
Elemental	0.91
Particulate	0.05
Primary containment leakage, percent/day	0.26
Secondary containment bypass leakage, cc/hour	1.35E+4
Penetration valve leakage control system leakage, cc/hour	1.70E+5
Primary containment free volume, ft^3	1.2E+6
Drywell volume, ft^3	2.4E+5
Drywell leakage (suppression pool bypass), percent	3.0
Effective suppression pool decontamination factors	
Noble gas	1
Organic iodine	1
Elemental iodine	7.87
Particulate iodine	7.87
Suppression pool water volume, ft^3	1.23E+5
ECCS leak rate, gpm	1.0
ECCS leak iodine partition factor	10
SGTS flow rate, cfm	2.5E+3
SGTS filter efficiencies, percent	
Organic	99
Elemental	99
Particulate	99
Positive pressure period, seconds	700
Atmospheric dispersion values (sec/m^3)	
0-02 hour EAB	8.58E-4
0-08 hour LPZ	1.13E-5
8-24 hour LPZ	7.89E-5
1-04 day LPZ	3.65E-5
4-30 day LPZ	1.21E-5

Table 2

Assumptions Used to Evaluate Radiological Consequence (Continued)

Control room volume, ft^3	2.4E+5
Control room unfiltered inleakage, cfm	10
Control room filtered air intake, cfm	1.948E+3
Control room air recirculation rate, cfm	1.948E+3
Control room filter efficiency, %	99
Control room atmospheric dispersion values (sec/m^3)	
0-08 hour	1.62E-3
8-24 hour	1.20E-3
1-04 day	4.05E-4
4-30 day	6.48E-4

Table 3

Assumptions Used to Evaluate Radiological Consequence

FHA

Parameters	Values
Reactor power, MWt	3100
Number of rods per bundle	74
Number of bundle in core	624
Decay time, day	11
Number of damaged rod	150
Release rate, %/day	6000
Pool decontamination factors	
Noble gas	1
Iodine	100
Gap fractions released	
Kr-85	0.30
All other noble gas	0.10
Iodine-131	0.12
All other iodines	0.10
Atmospheric dispersion values (sec/m^3)	
0-02 hour EAB	8.58E-4
0-08 hour LPZ	1.13E-5
8-24 hour LPZ	7.89E-5
1-04 day LPZ	3.65E-5
4-30 day LPZ	1.21E-5

Control Rod Drop Accident

Parameters	Values
Reactor power, MWt	3100
Number of damaged rod	850
Radial peaking factor	1.65
Gap fractions released	
Noble gas	1
Iodine	0.5
Fractions reached condenser	
Noble gas	1
Iodine	0.1
Plate out in condenser	
Noble gas	1
Iodine	0.5
Condenser leak rate, percent/day	1
Leakage duration, hours	24

3.8.3 Special Events

3.8.3.1 ATWS

The RBS meets the ATWS mitigation requirements defined in 10 CFR 50.62 using the following systems:

1. Installation of an alternate rod insertion system
2. Boron injection equivalent to 86 gpm
3. Installation of an automatic recirculation pump trip (RPT) logic

In addition, a plant specific ATWS analysis was performed to ensure that the following ATWS acceptance criteria were met:

1. Peak vessel bottom pressure less than ASME Service Level C limit of 1500 psig
2. Peak clad temperature within the 10 CFR 50.46 limit of 2200 °F
3. Peak clad oxidation within the requirements of 10 CFR 50.46
4. Peak suppression pool temperature less than 185 °F
5. Peak containment pressure less than 15 psig

The ATWS analysis was performed for the original power level and for the uprated power level to demonstrate the impact of the power uprate on the ATWS acceptance criteria. ATWS analyses were performed for the events described in Reference 7. RBS increased the product of boron concentration (C) and enrichment (E) to a value of 570 ppm from the previous value of 413 ppm to comply with the ATWS rule for the power uprate conditions. The NRC approved ODYN methodology (Reference 12), used for the analysis. The NRC staff concludes that the results of the analysis demonstrate that the RBS meets the ATWS acceptance criteria for uprated power conditions.

3.8.3.2 Station Blackout (SBO)

The plant response and coping capabilities for SBO are affected slightly by operation at the uprated power level due to the increase in operating temperature of the primary coolant system, increase in the decay heat (e.g., effects of increase in decay heat on the condensate water requirements), increase in the main steam SRV set points, and the ambient temperature increase in the areas which contain equipment necessary to mitigate the SBO event. The SBO was reevaluated using the guidelines of NUMARC 87-00 (Reference 32). The systems and equipment used to respond to SBO remain acceptable and the required coping time remains the same.

The ECCS and RCIC equipment room temperatures are affected by an increase in the calculated peak suppression pool temperature due to power uprate because of the increase in the piping heat losses and the heat transferred from the suppression pool through the walls to the RHR rooms. The licensee performed an evaluation of the response during SBO conditions for power uprate. The results of that evaluation showed that responses are bounded due to conservatism in the existing qualification of the equipment. The containment drywell area temperature increased by a small amount; however, the equipment necessary for event mitigation is qualified for these temperatures. The systems used to respond after power is restored are designed for the uprate suppression pool peak temperature.

Drywell, containment, and suppression pool design limits remain bounding during the coping period. The condensate water requirement increases; however, the current condensate storage tank design ensures that adequate water volume is available.

In summary, the NRC staff concludes that the power uprate conditions do not result in changes that significantly affect the previous evaluation or conclusion for the SBO and the RBS will continue to meet the requirements of the SBO rule for power uprate conditions.

3.8.3.3 Maine Yankee Lessons Learned

The RBS power uprate amendment was reviewed with regard to the recommendations from the report of the Maine Yankee lessons learned task group dated December 5, 1996. This report is documented in SECY-97-042 (Reference 33).

The NRC staff requested that the licensee identify all codes and methodologies used to obtain SLs and OLs and how they verified that these limits were correct for the appropriate uprate core. The licensee was also requested to identify and discuss any limitations associated with these codes and methodologies that may have been imposed by the staff. In Reference 2, EOI identified all the codes and methodologies used for the RBS power uprate analyses and confirmed that all the models and methodologies are used appropriately for the power uprate evaluation.

EOI confirmed that they had reviewed the results of the GE analyses to assure that the codes were used by GE correctly for power uprate conditions and the limitations and restrictions were followed appropriately by GE.

The main findings centered around the use and applicability of the Code methodologies used to support the uprated power. EOI verified that the codes are appropriate and applicable to the plant in the uprated conditions. EOI confirmed that the LOCA and transients analyses conform to the generic analyses approved by the NRC staff for power uprate.

3.9 <u>Additional Aspects of Power Uprate</u>

3.9.1 High-Energy Line Break (HELB)

The slight increase in the reactor operating pressure and temperature resulting from the plant operations at the proposed uprated power level will cause a small increase in the mass and energy release rates following certain HELBs. This results in a small increase in the subcompartment pressure and temperature profiles. The licensee stated that the HELB analysis evaluation was made for all systems (e.g., main steam system, feedwater system, RCIC system) evaluated in the RBS USAR. The evaluation shows that the affected buildings and cubicles that support the safety-related functions are designed to withstand the resulting pressure and thermal loading following a HELB. The equipment and systems that support a safety-related function are also qualified for the environmental conditions imposed upon them.

Based on the review of the licensee's evaluation, and the experience gained from the NRC staff's review of power uprate applications for similar BWR plants, the NRC staff concludes that the existing analysis for HELB remains bounding and is acceptable for plant operations at the proposed uprated power level.

3.9.2 Moderate-Energy Line Break (MELB)

The licensee performed an evaluation and concluded that the original MELB analysis is not affected by plant operations at uprated power level.

Based on the NRC staff review of the licensee's evaluation and the experience gained from the NRC staff's review of power uprate applications for similar BWR plants, the NRC staff concludes that the existing analysis for MELB remains bounding and is acceptable for plant operations at the proposed uprated power level.

3.9.3 Environmental Qualifications

3.9.3.1 Electrical Equipment

Environmental qualification (EQ) of electrical equipment important to safety located inside the containment is based on MSLB and/or design-basis LOCA conditions and their resultant temperature, pressure, humidity, and radiation consequences, and includes the environments expected to exist during normal plant operation. The licensee evaluated the EQ of electrical equipment important to safety located inside and outside the containment and determined that current accident and normal conditions for temperature, pressure, and humidity inside containment are nearly unchanged for the uprated power conditions.

The NRC staff requested the licensee to describe why the current accident and normal temperature, pressure, and humidity profiles for inside and outside of the primary containment do not change for the power uprate and why the power uprate has no impact on the EQ of electrical equipment important to safety. In Reference 2, the licensee stated that the design basis normal (not accident) temperature, pressure, and humidity profiles for both inside and outside of the primary containment remain unchanged from the preuprate profiles. This is due to the existing margin between actual and design basis conditions and to existing margins in the ventilation systems.

The licensee stated that power uprate added approximately 5 percent to the heat loads in the containment from piping, fuel pool, etc. For normal operation, the containment unit coolers have a design margin of 25 percent in cooling capacity. The 25 percent design margin bounds the 5 percent increase in heat gains as a result of containment pool temperature increase, piping heat gains, etc. The containment coolers will continue to maintain design environmental conditions (temperature and humidity) during normal conditions. The containment coolers are recirculating type and do not affect the containment pressure. The annulus pressure control system maintains a negative pressure of 3 inches water gauge in the annulus with respect to atmosphere during normal operation. The annulus pressure control system will not be impacted since there is no change in the environmental conditions in the containment.

The licensee determined that radiation levels under normal plant conditions are nearly unchanged for the uprated conditions.

With regard to the accident profile inside and outside primary containment, the licensee states that power uprate conditions will increase the blowdown mass and energy releases by less than 5 percent. The peak accident temperature, pressure, and humidity values inside the primary containment remain bounded by the existing profiles. The time histories of mass and energy release rates inside containment continue to be evaluated by the licensee against the existing temperature, pressure, and humidity profiles. The requirement of 10 CFR 50.49, "Environmental qualification of equipment important to safety for nuclear power plants," is that the licensee show that the subject equipment continues to be environmentally qualified prior to operating RBS at the uprated (flow-only) conditions. For this reason, the licensee must evaluate impact of the time histories of mass and energy release rates inside containment against the existing temperature, pressure, and humidity profiles, before implementing the power uprate. Any change to the EQs for safety-related equipment as a result of the power uprate, would be reflected in the RBS EQ files that are required to be maintained for RBS by 10 CFR 50.49; in this regard, the power uprate is no different from other changes in plant design or operation which result in potential changes to EQs.

With regard to the energy release from high-energy lines outside containment, the licensee states that due to the conservatism in the original design basis analyses, the calculated mass and energy release rates with power uprate for several of the evaluated high-energy lines outside containment were determined to be bounded by the original design basis analysis values. For those HELBs that are not bounded by the original design basis analysis, time histories of mass and energy release rates were generated. The temperature and pressure profiles were found to remain within the existing environmental design criteria envelope.

The radiation levels under accident conditions were conservatively evaluated to increase from 3 percent to 8 percent. The reevaluation of the uprated power conditions identified some equipment located inside and outside the containment which is potentially affected by the higher accident radiation level and required further actions to support equipment qualification. The licensee states that the qualification of this equipment will be maintained by refining radiation calculations (to make them location specific) or by slightly reducing qualified life.

The NRC staff concludes that the power uprate has only a minor effect on the environmental conditions currently used for qualifying electrical equipment important to safety inside and outside the primary containment. The electrical equipment EQ profiles continue to bound the calculated environmental conditions associated with the power uprate, subject to the acceptable resolution of issues involving time histories of mass and energy release rates inside containment. These issues will be addressed by the licensee prior to implementation of the power uprate.

3.9.3.2 Mechanical Equipment With Nonmetallic Components

Operation at the uprated power level increases the normal process temperatures up to 10 °F. The normal and accident radiation levels also increase slightly due to power uprate. The licensee performed an evaluation of the effects of plant operations at the proposed uprated power level on the nonmetallic components of safety-related mechanical equipment and stated that certain systems would be potentially affected by the slight increases in radiation levels due to plant operations at the proposed uprated power level; however, the effects of these increases in radiation levels are within the original EQ allowances and have been adequately addressed in the existing maintenance and surveillance programs.

Based on the review of the licensee's evaluation and the experience gained from NRC staff review of power uprate applications for similar BWR plants, the NRC staff concludes that the existing EQ of mechanical equipment with nonmetallic components remains bounding and is acceptable for plant operations at the proposed uprated power level.

3.9.4 Required Testing

Regulatory provisions for the testing of structures, systems, and components are identified under Criterion XI, "Test Control," of 10 CFR Part 50, Appendix B. The program for implementing these requirements is described in Section B.8 of the Entergy Quality Assurance Program Manual (QAPM), CRNO-98/00025, "Test Control." The QAPM description follows the guidance of Regulatory Guide 1.33, Revision 2 (Reference 34), which conditionally endorses ANS-3.2/ANSI N18.7-1976 (Reference 35), with respect to the development of test procedures, conduct of testing, and documentation and evaluation of test results.

Additionally, a summary report of the power uprate program will be submitted after the completion of the uprate test program, as required by RBS TS 6.9.1.1.

The generic test guidelines for GE BWR Power Uprate is contained in Reference 7, Section 5.11.9. It reads as follows:

> A testing plan will be included in the uprate licensing application. It will include pre-operational tests for systems or components which have revised performance requirements. It will also contain a power increase test plan.

Guidelines to be applied during the approach to and demonstration of uprated operating conditions are provided in Section L.2 of Reference 6; Reference 8 provides the required additional information relative to power uprate testing.

With regard to the startup test plan, the licensee will conduct limited startup testing at the time of implementation of power uprate. The tests will be conducted in accordance with the guidelines of Reference 6 to demonstrate the capability of plant systems to perform their designed functions under uprated conditions.

The tests will be similar to some of the original startup tests, described in Section 14.2.12.2 of the licensee's USAR. Testing will be conducted with established controls and procedures, which have been revised to reflect the uprated conditions. Revised plant procedures, reflecting the uprate conditions, will be used to the extent practicable during the test program.

The tests consist essentially of steady-state, baseline testing between 90 percent and 100 percent of the currently licensed power level, in increments of 5 percent power. At least one set of data will be obtained between 100 percent and 103 percent power, and a final set of data at the uprated (105 percent) power level. The tests will be conducted in accordance with a site-specific test procedure currently being developed by the licensee. The test procedure will be developed in accordance with written procedures as required by 10 CFR Part 50, Appendix B, Criterion XI.

The following power increase test plan is provided in Reference 8, Section 10.4, "Required Testing."

1. Surveillance testing will be performed on the instrumentation that requires re-calibration for power uprate in addition to the testing performed according to the RBS TS schedule.

2. Steady-state data will be taken at points from 90% up to the previous rated thermal power, so that system performance parameters can be projected for uprated power before the previous power rating is exceeded.

3. Power increases beyond the previous rating will be made along an established flow control/rod line in increments of ≤ 3% power. Steady-state operating data including fuel thermal margin will be taken and evaluated at each step.

4. Control system tests will be performed for the recirculation flow controls, feedwater/reactor water level controls and pressure controls. The operational tests will be made at the appropriate plant conditions for that test and at each power increment above the previous rated power condition, to show acceptable adjustments and operational capability. The performance criteria will be used as in the original power ascension tests, unless they have been replaced by updated criteria since the initial test program.

The licensee's test plan follows the guidelines of Reference 6 and the NRC staff position regarding individual power uprate amendment requests.

With regard to performance testing, Reference 6, Section 5.11.9 guidelines specify that pre-operational tests will be performed for systems or components which have revised performance requirements. The licensee plans to conduct tests during the ascension to power uprate conditions. The performance tests and associated acceptance criteria are based on the RBS original startup test specifications and previous GE BWR power uprate test programs. Table 4 shows the systems identified for performance testing.

With regard to the recirculation pump testing, vibration testing is not required because RBS is equipped with FCV and there is no change in the maximum core flow for the power uprate.

Of the systems that will be tested during uprate power ascension, only the reactor feedwater/reactor level control system and reactor pressure control system, including subsystems for pressure regulation, turbine bypass, and TCVs, are deemed to have substantive changes in performance requirements; therefore, these systems will be dynamically tested, as part of the power ascension test procedure, to ensure that they will perform adequately at the new higher flow rates and power levels. These systems will be tested in accordance with the methodology employed during the original startup testing. The original acceptance criteria will be used, except where they have been superceded by new criteria as a result of evaluations which were performed while dispositioning test exceptions during original startup testing.

The results from the uprate test program will be used to revise the operator training program to more accurately reflect the effects of the power uprate.

The licensee's programs for startup testing follows the guidelines of Reference 6, which have been accepted by the NRC as the generic basis for power uprate amendment requests. The submittal provides a test program that follows Reference 6 guidelines for uprate testing and meets 10 CFR Part 50, Appendix B, requirements for test control; therefore, the NRC staff concludes that the licensee's power uprate test program is acceptable.

Table 4

Performance Testing

The licensee identified the following systems for performance testing during ascension to power uprate conditions:

System	Test Purpose/Function
Intermediate Range Neutron Monitors	Assure Source Range Neutron Monitors and Average Power Range Monitors Overlap
Average Power Range Monitors	Calibration
Pressure Regulator System	Setpoint steps, Failures, Incremental Regulation
Feedwater Control System	Setpoint Changes, Incremental Regulation
Recirculation Flow Control	Step and Ramp Changes
Recirculation Flow	Calibration
Radiation Measurements	Survey

3.9.5 Probabilistic Risk Assessment (PRA) and Individual Plant Evaluation (IPE)

The licensee addressed the impact of the proposed 5 percent power uprate on plant risk at RBS. The licensee's submittal referred to Reference 7, which determined the effects of a 5 percent power uprate on BWR PRA. The licensee concluded that the finding of this generic evaluation, that risk associated with 5 percent power uprate will have negligible impact on plant risk, is applicable to RBS.

The NRC staff reviewed Section 10.5, "Individual Plant Evaluation," of Reference 8, as well as Section 2.4, "Probabilistic Safety Assessment," of Reference 7. Reference 7 listed initiating event frequency, success criteria, component failure rates, and time available for operator action as PRA parameters and inputs that could potentially be affected by the proposed power uprate. Reference 7 concluded that for the proposed 5 percent uprate, there would be negligible, if any, change in these parameters and inputs and that the resulting change in plant risk would be insignificant. The NRC staff concludes that the generic evaluation is applicable to RBS.

The NRC staff believes that the review of the quality of the PRA should be commensurate with the role that the PRA results play in the NRC staff's decision process and with the degree of rigor needed to provide a valid technical basis for the staff's decision. In this case, the licensee is not requesting relaxation of any deterministic requirements for the proposed power uprate application, and the NRC staff approval is based on the licensee meeting the current deterministic requirements; therefore, the NRC staff considered the NRC staff's original evaluation of the RBS IPE (Reference 36), the PRA portion of the power uprate submittal (Section 10.5 of Reference 8), and Reference 7, and concludes that they constitute sufficient indication of the PRA quality.

The NRC staff concludes that based on the current analysis, no significant change would be expected for initiating event frequencies, success criteria, component failure rates, and operator reaction time. Based on the reported analysis and results, the NRC staff concludes that the impact of the proposed 5 percent power uprate on RBS risk is negligible.

3.9.6 Operator Training and Human Factors

The NRC staff reviewed the application, as amended (References 1, 2, 3, 4, and 5) for the power uprate. The NRC staff's review of the licensee's responses relative to five operator licensing and human performance evaluation topics is provided below.

Topic 1 - Discuss whether the power uprate will change the type and scope of plant emergency and abnormal operating procedures. Will the power uprate change the type, scope, and nature of operator actions needed for accident mitigation and will new operator actions be required?

The licensee stated in Reference 1 that "the plant EOPs [Emergency Operating Procedures] will be reviewed for any effects of power uprate, and the EOPs will be updated, as necessary. The review will be based on Section 2.3 of Reference [7]."

The licensee further stated that "the operator actions in the EOPs will not change as a result of increasing rated power; only the conditions at which some of the actions are specified will change. No new operator actions are required." Reference 4 indicates that updating of the EOPs is a routine administrative process that is controlled by procedures.

The NRC staff concludes that the licensee's response is satisfactory.

Topic 2 - Provide examples of operator actions that are particularly sensitive to the proposed increase in power level and discuss how the power uprate will affect 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 complete 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.

The licensee stated in its response to the NRC staff's request for additional information (Reference 4) that its "Operations Training staff performed simulator observations for the purpose of comparing simulator response at current 100 [percent] power to the expected simulator response at 105 [percent] (Uprated Power)...." MSIV closure with an ATWS at equilibrium xenon conditions with no operator actions assumed was simulated. "Based on the observed response of the simulator, no appreciable time frame or parameter differences were noted [between the 100 percent and 105 percent condition]."

The licensee described a human reliability analysis (HRA) it performed, using the guidance in NUREG/CR-1278 (Reference 37) and NUREG/CR-4772 (Reference 38). From the HRA, human error probabilities (HEPs) were determined to assess the impact of the power uprate on operator reliability. The HRA was performed for at-power and SD conditions. For the at-power condition, pre-accident, post-accident, and ATWS cases were examined. For the SD condition, the HRA examined pre-accident and post-accident cases.

The licensee determined that, for the at-power condition associated with the post-accident case, HEPs which were derived using a screening process were conservative and offset the potential impact of power uprate. For the HRAs that used the nominal process to determine HEPs, "inputs to the HRA due to power uprate are small and should not impact the final HEP."

For the at-power ATWS case, the HRA performed by the licensee assumed that operators could recognize the ATWS condition and made no diagnosis errors. "Nearly all the HEPs considered are errors of omission." Appropriate levels of stress were also considered as affecting operator performance during the at-power ATWS case.

"To illustrate how little impact that power uprate is expected to have on CDF [core damage frequency], the RBS PRA model assumes that the HEP for failure to inject SLC [standby liquid control] is 1.0E-3, which produces a CDF of 1.066E-11. If the failure to inject SLC is increased 10 fold to 1.0E-2 to represent power uprate, the CDF increases to 1.066E-10."

For the SD case, the licensee indicated that "the power uprate does not impact the HEPs for shutdown... The HRA for pre-power-uprate contains ample conservatism in its assumptions [so] that a [5 percent] increase in power will not change the HEPs for power uprate." The NRC staff concludes that the licensee's response is satisfactory.

Topic 3 - Discuss all changes the power uprate will have on control room alarms, controls, and displays. For example, will zone markings on meters change (e.g., normal range, marginal range, and out-of-tolerance range)? If changes will occur, discuss how they will be addressed.

In its response to the NRC staff's request for additional information (Reference 4), the licensee stated "...there are several parameters that are expected to change during power uprate that will have an impact on control room indicator color banding. These parameters include reactor pressure, main steam flow, RCIC turbine speed, standby liquid control system storage tank level, moisture separator-reheater pressure, and main steam pressure. The zone markings for the indicators affected by these changes will be adjusted to accommodate the uprate conditions."

The licensee does not expect the power uprate to affect control room panel layouts or annunciator window legends. Any affected alarm setpoints (e.g., reactor vessel high-pressure SCRAM, high-pressure ATWS RPT, and main steam high flow isolation) will be adjusted to accommodate the power uprate conditions. The feedwater pump suction header low-pressure setpoint is expected to decrease as a result of the proposed power uprate.

The licensee further indicated that eight instrument loops will require range changes as a result of the uprate: four main steam flow, one turbine load set, one turbine load, and two main steam line pressure loops.

The licensee indicated that changes in instrumentation in the main control room will be prepared in accordance with the plant modification process, which includes a detailed review of the proposed control room design change package. All identified changes to the control room alarms, controls, and displays will be implemented before operating at uprated power.

The staff finds that the licensee's response is satisfactory.

Topic 4 - Discuss all changes the power uprate will have on the Safety Parameter Display System (SPDS) and how they will be addressed.

The licensee stated in Reference 1 that EOP curves and limits may also be included in the SPDS and that it must be updated accordingly. Reference 4 indicates that updating of the SPDS is a routine administrative process that is controlled by procedures.

This NRC staff concludes that this commitment is acceptable.

Topic 5 - Describe all changes the power uprate will have on the operator training program and the plant simulator.

The licensee stated in Reference 1 that "training required to operate the plant following uprate will be conducted prior to restart of the unit at uprated conditions. Data from startup testing will be included in the training as appropriate."

The licensee indicated that "when applicable, the results from the uprate test program will be used to revise the operator training program to reflect the effects of the uprated conditions."

The licensee further states that classroom training will be combined with simulator training and that the simulator training, at a minimum, will include a demonstration of transients that show the greatest change in plant response at uprated power compared to nonuprated power.

Simulator changes and fidelity revalidation will be completed in accordance with the requirements of ANS/ANSI 3.5 (Reference 39).

The NRC staff finds that the licensee's response is satisfactory and consistent with the existing simulation facility certification.

The NRC staff concludes that the previously discussed review topics associated with the proposed RBS uprate have been or will be satisfactorily addressed. The NRC staff further concludes that the power uprate should not adversely affect simulation facility fidelity, operator performance, or operator reliability.

3.9.7 Plant Life (Maintenance Rule)

Reference 8, Section 10.7, "Plant Life," states in its entirety:

> The longevity of most equipment is not affected by power uprate. There are various River Bend Station programs (Equipment Qualification, Flow Accelerated Corrosion) that deal with age-related components. These programs will not change as a result of power uprate. In addition, the Maintenance Rule provides oversight for the other mechanical and electrical components, important to plant safety, to guard against age-related degradation.

The equipment qualification, flow-accelerated corrosion and maintenance rule (10 CFR 50.65) programs detect and mitigate age-related degradation of components at RBS. The NRC staff has reviewed the licensee's submittal regarding plant life and finds that it is consistent with the guidelines of Reference 6, Section 5.11.6, "Plant Life," which have been accepted by the NRC staff as the generic basis for power uprate amendment requests, and is therefore acceptable.

3.10 <u>License and TSs</u>

Based on the considerations discussed above, the increase in rated thermal power (RTP) from 2894 MWt to 3039 MWt is acceptable. This change will be reflected in paragraph C.(1) of RBS FOL NPF-47. In addition, RBS TS Section 1.1, "Definitions," the definition of RTP will be revised to reflect the change in RTP.

The increase in RTP operation requires the following additional changes to the License and TSs:

1. License Condition (LC) 2.C.(13)

    The feedwater temperature would be revised to 326 °F.

    The analyses done for power uprate are consistent with the use of 426 °F as the rated feedwater temperature and a 100 °F temperature reduction as a lower limit of operation at rated conditions. As a result, the change to the LC for feedwater temperature reduction from 320 °F to 326 °F is consistent with the licensing and design basis. Accordingly, the proposed change to the LC is acceptable.

2. Thermal Power SL Margin Reduction

The thermal power SL would be reduced from 25 percent to 23.8 percent to maintain the basis for transient analyses as reflected in the following TSs:

(A) 1.4, "Frequency," Examples 1.4-2 and 1.4-3

(B) 2.1, "SLs," 2.1.1, "Reactor Core SLs," Subsection 2.1.1.1

(C) 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)," and SR 3.2.1.1

(D) 3.2.2, "Minimum Critical Power Ratio (MCPR)," and SR 3.2.2.1

(E) 3.2.3, "Linear Heat Generation Rate (LHGR)," and SR 3.2.3.1

(F) 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," and SR 3.3.1.1.2

(G) TS Table 3.3.1.1-1, "Reactor Protection System Instrumentation," Item 5, "Reactor Vessel Water Level - High, Level 8"

(H) 3.4.3, "Jet Pumps," SR 3.4.3.1

(I) 3.7.5, "Main Turbine Bypass System"

Decreasing this limit assures continued compliance with all SLs at the uprated conditions. Accordingly, these proposed changes to the TSs are acceptable.

3. Control Rod Scram Times

The normal reactor steam dome pressure would be increased from 1025 psig to 1055 psig for power uprate. Since the scram performance and the requirements are already known for a dome pressure of 1050 psig, the effects of incremental change in the dome pressure from 1050 psig to 1059 psig was evaluated and found to be acceptable. Accordingly, the proposed change in scram times for the reactor steam dome pressure in TS Table 3.1.4-1, "Control Rod Scram Times," is acceptable.

4. CRD Pressure

The CRD charging water header pressure would be increased from greater than or equal to 1520 psig to greater than or equal to 1540 psig in the following TSs:

(A) 3.1.5, "Control Rod Scram Accumulators," and SR 3.1.5.1

(B) 3.9.5, "Control Rod Operability - Refueling," SR 3.9.5.2

(C) 3.10.8, "Shutdown Margin (SDM) Test - Refueling," and SR 3.10.8.6

The current scram time requirements apply to a reactor steam dome pressure of 950 psig to 1050 psig. For power uprate, the reactor steam dome pressure range is increased from 950 psig to 1059 psig, which is a 9 psig increase over the current scram performance requirement at 1050 psig. The 20 psig increase in the minimum charging water pressure and scram accumulator pressure is sufficient to offset the reactor steam dome pressure increase of 9 psig and to maintain the scram performance margin that existed at 1050 psig, based upon scram performance predictions. Accordingly, these proposed changes to the TS are acceptable.

5. SLCS

The licensee has proposed that the SLCS product of the E and C in TS 3.1.7, "Standby Liquid Control (SLC) System," and in SR 3.1.7.3 be increased from greater than or equal to 413 to greater than or equal to 570 as described in Section 3.8.3.1 herein. The proposed increase is necessary to meet the ATWS rule and 10 CFR 50.62, and is acceptable.

6. SLCS Surveillance Test Pressure

The licensee has proposed that the SLCS surveillance test pressure in SR 3.1.7.7 be increased from greater than or equal to 1220 psig to greater than or equal to 1250 psig. This pressure increase is the same increase as the SRV pressure setpoint increase and is acceptable.

7. Reactor Vessel Steam Dome Pressure - High, Scram Setpoint

The licensee has proposed that the allowable value for the reactor vessel steam dome pressure - high, scram setpoint, Item 3 in TS Table 3.3.1.1-1, be increased from less than or equal to 1079.7 psig to less than or equal to 1109.7 psig. This proposed increase of 30 psig is consistent with the proposed increased reactor pressure and is acceptable; however, since this change would only be applicable after implementation of Phase Two of the power uprate, a footnote is added to maintain a value of less than or equal to 1079.7 psi until the pressure increase phase of the power uprate.

8. ATWS Pump Trip (ATWS-PT) Instrumentation Surveillance

The licensee has proposed that the ATWS-PT setpoint in SR 3.3.4.2.4, "Anticipated Transients Without Scram (ATWS) Pump Trip (ATWS-PT) Instrumentation," be increased from less than or equal to 1135 psig to less than or equal to less than or equal to 1165 psig. This 30 psig increase is consistent with the 30 psig pressure increase in reactor operating pressure and is acceptable.

9. SRV Relief, Safety, and Low Low Setpoints (LLS) SR 3.3.6.4.3a and b and SR 3.4.4.1.

The licensee has proposed that all SRV relief, safety, and LLS be increased by 30 psig in the following TSs:

(A) 3.3.6.4, "Relief and LLS Instrumentation," SR 3.3.6.4.3

(B) 3.4.4, "Safety/Relief Valves (S/RVs)," SR 3.4.4.1

The proposed increase of 30 psig is consistent with the 30 psig pressure increase. The S/RV setpoint tolerance of plus or minus 36 psig (which represents plus or minus 3 percent) is added only to the safety function setpoint, as described in Section 3.2.2 herein. Accordingly, these proposed changes to the TSs are acceptable.

10. RCIC Surveillance Test Pressure, Maximum

The licensee has proposed an increase in the maximum RCIC surveillance test high pressure from less than or equal to 1045 psig to less than or equal to 1075 psig in SR 3.5.3.3. The proposed pressure increase is consistent with the increase in reactor operating pressure and is acceptable.

11. Automatic Initiation-Primary Containment Isolation-Main Steam Line Flow-High

The licensee has proposed an increase in the Automatic Initiation-Primary Containment Isolation-Main Steam Line Flow-High, Allowable Value, in TS Table 3.3.6.1-1, "Primary Containment and Drywell Isolation Instrumentation." This trip setpoint range would be changed from less than or equal to 151 psid to 169 psid, to less than or equal to 190 psid to 194 psid. This proposed increase reflects the higher main steam flow associated with the power uprate and is acceptable.

12. SLO

The current limitations on SLO in TS 3.4.1, "Reactor Loops Operating," would be maintained. The maximum power for SLO would be limited to the previous level. This would reduce the uprated limit by the ratio of the current rated percentage (100 percent) to the uprated percentage (105 percent). The thermal power range for SLO would be changed from 83 percent to 79 percent. This change would maintain plant operation consistent with the assumptions of the safety analyses for power uprate and is acceptable.

13. RCS Pressure Isolation Valve Leakage Test Pressure

The licensee has proposed that range for reactor coolant pressure isolation valve test pressure, in SR 3.4.6.1, "RCS Pressure Isolation Valve (PIV) Leakage," be increased from greater than or equal to 1010 psig to greater than or equal to 1040 psig and less than or equal to 1040 psig to less than or equal to 1070 psig. This 30 psig increase is consistent with the increased reactor pressure and is acceptable.

14. RCS P/T Limits, TS Figure 3.4.11-1 and SR 3.4.11-1

As noted in Section 3.2.3 herein, the licensee has proposed the adoption of 32 EFPY P/T limits. This proposal involves the replacement of TS Figure 3.4.11-1 and associated changes to SR 3.4.11.1 and SR 3.4.11.2, "RCS Pressure and Temperature (P/T) Limits." As noted in Section 3.2.3 herein, adoption of the 32 EFPY P/T limits is acceptable; accordingly, these proposed TS changes are acceptable.

15. 3.4.12, "Reactor Steam Dome Pressure," and SR 3.4.12.1

The licensee has proposed an increase in the maximum allowable RCS dome pressure in TS 3.4.12, "Reactor Steam Dome Pressure," and SR 3.4.12.1 from less than or equal to 1045 psig to less than or equal to 1075 psig. This 30 psig increase is consistent with the increased reactor pressure and is acceptable.

16. Average Power Range Monitor (APRM) and APRM Flow Biased Simulated Thermal Power–High

The licensee's planned implementation of power uprate is in two phases. Phase One is a flow-only implementation and Phase Two is a pressure increase implementation. Two APRM-related TS changes are needed to allow this phased implementation at power. The licensee has proposed that these TS changes apply for a 30-day period.

(A) The licensee has proposed a change to SR 3.3.1.1.2, "Reactor Protection System (RPS) Instrumentation," which would change the APRM tolerance from "±2%" to "-2% to +7%" for 30 days. SR 3.3.1.1.2 requires the confirmation every 7 days that the absolute difference between the APRM channels and the calculated power is less than or equal to 2 percent RTP.

At the beginning of the flow-only implementation phase, the RTP would be changed from 2894 MWt to 3039 MWt. The APRM calibration would be to the pre-uprate RTP (2894 MWt), which is approximately 95 percent of the uprated RTP. The APRMs would be reading approximately 3 percent to 7 percent higher, which would not be within the required 2 percent tolerance. As a result, at the initiation of uprate, all APRMs would be out of calibration and must be declared inoperable. Limiting Condition for Operation 3.3.1.1 actions would require the inoperable channels to be restored to operable status or the plant to be in Mode 2 within 6 hours. The recalibration of the APRMs within this time period would not be practical.

The proposed 30-day increase in the tolerance band of "-2% to +7%" during the flow-only implementation period would allow time for each of the APRMs to be recalibrated to the uprated RTP, and thereby restore compliance with the required 2 percent tolerance. As each APRM channel is recalibrated, the tolerance for that channel would revert to the 2 percent tolerance.

The proposed time limit to complete the uprate implementation is 30 days; therefore, the temporary increase in tolerance would be only for a period of 30 days. During the flow-only phase, with the nonrecalibrated APRM channels reading at an equivalent 95 percent of the uprated power, the resulting peak power would be no greater than that calculated in the accident analyses. The nonrecalibrated APRMs would initiate a reactor trip at a level below that assumed in the analyses relative to the uprated RTP, which would be conservative and, accordingly, acceptable.

(B) The licensee has proposed a change to allow administrative control of SR 3.3.1.1.3 and TS Table 3.3.1.1-1, Item 2.b for 30 days. SR 3.3.1.1.3 requires the adjustment of the FCTR card to conform to reactor flow once within 7 days after reaching equilibrium conditions following a refueling outage. This TS requires the FCTR cards to conform to limits contained in the COLR. At the time of initiation of the uprate, the FCTR limits would be based on an RTP of 2894 MWt. The flow-only uprate implementation would change the basis for the COLR from 2894 MWt to 3039 MWt, thus making the FCTR cards out of calibration. As the adjustments to the FCTR cards are being made, the cards would be brought into calibration with the uprated COLR.

During the transition period, the SRs of SR 3.3.1.1.3 and TS Table 3.3.1.1-1, Item 2.b, would be administratively controlled from the 2894 MWt to 3039 MWt limits. This control would track the correct setpoints and limits of the individual instrument channels to the appropriate limits. All of the FCTR cards would be readjusted to the new 3039 MWt limits prior to power ascension above 2894 MWt. The proposed time limit to complete the uprate implementation is 30 days; therefore, the temporary administrative control would be only for a period of 30 days. During the flow-only phase, with the FCTR limits based on an RTP of 2894 MWt, which would be a conservative configuration, plant responses to an event would be within the accident analyses. Based on the above evaluation, the NRC staff concludes that the licensee's proposed changes to the TS for a period of 30 days are conservative, will be administratively controlled, and will be for a limited amount of time, and are, therefore, acceptable.

3.11 Commitments

In reviewing the application dated July 30, 1999, as supplemented, the NRC staff noted that the licensee made commitments regarding activities associated with the proposed RBS power increase. The commitments, which the NRC staff considers to be safety-significant, are as follows:

1. The licensee will maintain the SFP design limits (i.e., maximum temperature and corresponding heat removal capacity) by controlling the rate of discharge to the SFP. This proposal is in lieu of the SRP criteria of Section 9.1.3. This commitment is contained in Reference 4.

2. All identified changes to control room alarms, controls, and displays will be implemented prior to operation at the uprated conditions, as applicable to Phase One or Phase Two. This commitment is contained in References 4 and 8.

3. Power uprate testing will be performed as described on pages 38 and 39 of Enclosure 2 of Reference 4, as applicable to Phase One or Phase Two. This commitment is contained in Reference 4.

4. The modifications to MOVs necessary to support the pressure increase portion of power uprate will be completed prior to implementation of this phase (Phase Two). The modifications to the program MOVs will include both uprate conditions and Reference 26 recommendations in accordance with EOI programs. This commitment is contained in Reference 4.

5. Training required to operate the plant following uprate will be conducted prior to restart of the unit at uprated conditions. Data from startup testing will be included in the training as appropriate. When applicable, the results from the uprate test program will be used to revise the operator training program to reflect the effects of the uprated conditions.

   Classroom training will combine simulator training and the simulator training will include a demonstration of transients that show the greatest change in plant response at uprated power compared to nonuprated power. These commitments are contained in Reference 8.

6. Simulator changes and fidelity revalidation will be completed in accordance with the requirements of Reference 39. These commitments are contained in Reference 8.

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above regulatory commitments are best provided by the licensee's administrative processes, including its commitment management program. The above regulatory commitments do not warrant the creation of regulatory requirements. The staff notes that pending industry and regulatory guidance pertaining to 10 CFR 50.71(e) may call for some information related to the above commitments to be included in a future update of the RBS UFSAR.

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

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an Environmental Assessment and Finding of No Significant Impact was published in the *Federal Register* on September 28, 2000 (65 FR 58298). Accordingly, based upon the Environmental Assessment, the Commission has determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

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.

7.0 REFERENCES

1. Letter from Randall K. Edington, EOI, to NRC, "License Amendment Request (LAR) 99-15, Changes to Technical Specifications for Power Uprate of River Bend Station," July 30, 1999.

2. Letter from Rick J. King, EOI, to NRC, "Additional Information Related to License Amendment Request (LAR) 99-15, Changes to Technical Specifications for Power Uprate of River Bend Station," April 3, 2000.

3. Letter from Randall K. Edington, EOI, to NRC, "Request to Conduct Flow Only Uprate via Phased Implementation of License Amendment Request (LAR) 99-15, Changes to Technical Specifications for Power Uprate of River Bend Station," May 9, 2000.

4. Letter from Rick J. King, EOI, to NRC, "Additional Information Related to License Amendment Request (LAR) 99-15, Changes to Technical Specifications for Power Uprate of River Bend Station," July 18, 2000.

5. Letter from Randall K. Edington, EOI, to NRC, "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," August 24, 2000.

6. GE Nuclear Energy Topical Report, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDO-31897, Class I (Non-Proprietary), February 1992; and NEDC-31897P-A, Class III (Proprietary), May 1992.

7. GE Nuclear Energy Topical Report, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," Licensing Topical Report NEDC-31984, Class I (Non-Proprietary), March 1992, and Supplements 1 and 2; and NEDC-31984P, Class III (Proprietary), July 1991.

8. GE Nuclear Energy Topical Report, "Safety Analysis Report for River Bend 5% Power Uprate," Licensing Topical Report NEDC-32778P, Class 3 (Proprietary), July 1999.

9. GE Nuclear Energy Topical Report, "Safety Analysis Report For River Bend 5 Percent Power Uprate, Supplement 1, Changes for A Flow Increase Only Power Uprate," Licensing Topical Report NEDC-32778P, Class III (Proprietary), April 2000.

10. Letter from NRC (Robert Fretz) to EOI (Randall Edington), "River Bend Station, Unit 1-Issuance of Amendment Re: Reactor Stability Long Term Solution (TAC No. MA3888)," May 5, 1999.

11. Letter from A.C. Thadani (NRC) to C.L.Tully (BWR Owner's Group (OG)),"Acceptance for Referencing of Licensing Topical Report NEDC-31753P, "BWROG In-Service Pressure Relief Technical Specification Licensing Report," dated March 8, 1993.

12. GE Nuclear Energy Topical Report, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," Licensing Topical Report NEDO-24154-P-A, Class III (Proprietary), August 1986.

13. (Deleted)

14. Letter from Rick J. King, EOI, to NRC, "Additional Information Related to License Amendment Request (LAR) 99-15, Regarding Fluence Determination," May 8, 2000.

15. GE Nuclear Energy, Service Information Letter No. 377, "RCIC Startup Transient Improvement with Steam Bypass," June 1982.

16. GE Nuclear Energy Topical Report, "The GE Pressure Suppression Containment System Analytical Model," NEDM-10320, March 1971.

17. GE Nuclear Energy Topical Report, "The General Electric Mark III Pressure Suppression Containment System Analytical Model," NEDO-20533, June 1974.

18. GE Nuclear Energy Topical Report, "General Electric Model for LOCA Analysis In Accordance With 10 CFR Part 50, Appendix K," NEDE-20566-P-A, September 1986.

19. U.S. NRC Regulatory Guide 1.49, "Power Levels of Nuclear Power Plants," Revision 1, December 1973.

20. ANS/ANSI Standard 5.1, "Decay Heat Power In Light Water Reactors," 1994.

21. U.S. NRC NUREG-0783, "Suppression Pool Temperature Limits for BWR Containment," September 1988.

22. GE Nuclear Energy Topical Report, "Elimination of Limit on Local Suppression Pool Temperature for SRV Discharge With Quenchers," NEDO-30832A, Class I, May 1995.

23. Letter from G. Halahan (NRC) to R. Pinelli (BWROG), transmitting "Safety Evaluation Report by the Office of Nuclear Reactor Regulation on the Review of Two GE Topical Reports for the Elimination of Local Temperature Limits and Raising Pool Temperature Technical Specification Limits," August 29, 1994.

24. NRC Generic Letter 89-10 to All Licensees of Operating Nuclear Power Plants and Holders of Construction Permits for Nuclear Power Plants, "Safety-Related Motor-Operated Valve Testing and Surveillance," June 28, 1989.

25. EOI, RBS "Generic Letter 89-10 MOV Program Power Uprate Report," Revision 0, February 1999.

26. Limitorque Corporation, Technical Update (LTU) 98-01 and LTU 98-01, Supplement 1, "Actuator Output Torque Calculation," July 17, 1998.

27. NRC Generic Letter 96-06 to All Holders of Operating Licenses for Nuclear Power Reactors, Except for Those Licenses That Have Been Amended to Possession-Only Status, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," September 30, 1996.

28. Letter from Rick J. King, EOI, to NRC, "Response to NRC Generic Letter 96-06," January 28, 1997.

29. GE Nuclear Energy Topical Report, "River Bend Station, SAFER/GESTAR-LOCA, Loss-of-Coolant Licensing Topical Report," NEDC-32640P, June 1997.

30. GE Nuclear Energy Topical Report, "General Electric Instrument Setpoint Methodology," Class III (Proprietary), NEDE-31336-P, October 1996.

31. GE Nuclear Energy Topical Report, "GESTAR II–General Electric Standard Application for Reactor Fuel," Amendment II, Revision 8, NEDE-24011-P-A, November 5, 1990.

32. Nuclear Utilities Management and Resource Council (NUMARC), "Guidelines and Technical Basis for NUMARC Initiatives Addressing Station Blackout and Light Water Reactors," NUMARC 87-00, November 1987.

33. NRC SECY-97-042, "Response to OIG [Office of the Inspector General] Event Inquiry 96-045 Regarding Maine Yankee," February 18, 1997.

34. NRC Regulatory Guide 1.33, "Quality Assurance Program," Requirements (Operation), Revision 2, February 1978.

35. ANS-3.2/ANSI-N18.7 Standard, "Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants," 1976.

36. Letter from W. H. Odell, Gulf States Utilities Company, to the NRC, "Individual Plant Evaluation (IPE) in response to Generic Letter 88-20," February 1, 1993.

37. NRC NUREG/CR-1278, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," October 31, 1983.

38. NRC NUREG/CR-4772, "Accident Sequence Evaluation Program Human Reliability Procedure," February 28, 1987.

39. ANS/ANSI Standard 3.5 (1985), "Nuclear Power Plant Simulators for Use in the Operator Training and Examination," 1986.

40. Letter from Randall K. Edington, EOI, to NRC, "Proposed License Condition to License Amendment Request (LAR) 99-15, Changes to Technical Specifications for Power Uprate of River Bend Station," October 2, 2000.

Principal Contributors:

G. Thomas	R. Goel
J. Lee	D. Shum
J. Bongarra	M. Bugg
M. Mitchell	S. Lee
C. Carpenter	B. Marcus
O. Chopra	C. Wu
C. Hinson	D. Jaffe
J. Harold	

Date: October 6, 2000

UNITED STATES NUCLEAR REGULATORY COMMISSION

ENTERGY GULF STATES, INC.

AND

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-458

NOTICE OF ISSUANCE OF AMENDMENT TO RIVER BEND STATION, UNIT 1

FACILITY OPERATING LICENSE NPF- 47

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 114 to Facility Operating License No. NPF- 47 issued to Entergy Gulf States, Inc. and Entergy Operations, Inc. (EOI, or the licensee), which revised the Technical Specifications for operation of the River Bend Station, Unit 1, located in Saint Francisville, Louisiana. The amendment is effective as of the date of issuance and shall be implemented no later than the start-up following the next refueling outage.

The amendment modified the Technical Specifications to increase the maximum allowable thermal power from 2894 megawatts thermal (MWt) to 3039 MWt.

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

Notice of Consideration of Issuance of Amendment to Facility Operating License and Opportunity for a Hearing in connection with this action was published in the FEDERAL REGISTER on June 14, 2000 (65 FR 37413). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of the amendment will not have a significant effect on the quality of the human environment (65 FR 58298).

For further details with respect to the action see (1) the application for amendment dated July 30, 1999, as supplemented by letters dated April 3, May 9, July 18, August 24, and October 2, 2000, (2) Amendment No. 114 to License No. NPF-47, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. Documents may be examined, and/or copied for a fee, at the NRC's Public Document Room, located at One White Flint North, 11555 Rockville Pike (first floor), Rockville, Maryland, and accessible electronically through the ADAMS Public Electronic Reading Room link at the NRC Web site (http://www.nrc.gov).

Dated at Rockville, Maryland, this 6th day of October 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

Jefferey F. Harold

Jefferey F. Harold, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation