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

June 1, 2000

Mr. John K. Wood
Vice President - Nuclear, Perry
FirstEnergy Nuclear Operating Company
P.O. Box 97, A200
Perry, OH 44081

SUBJECT: PERRY NUCLEAR POWER PLANT, UNIT 1 - ISSUANCE OF AMENDMENT (TAC NO. MA6459)

Dear Mr. Wood:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 112 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit 1. This amendment revises the Technical Specifications in response to your application dated September 9, 1999 (PY-CEI/NRR-2420L), as supplemented by submittals dated March 1 (PY-CEI/NRR-2470L), March 13 (PY-CEI/NRR-2477L), and May 11, 2000 (PY-CEI/NRR-2499L).

This amendment increases the present 100 percent authorized rated thermal power level of 3579 megawatts thermal to 3758 megawatts thermal. This represents a power level increase of five percent for the Perry Nuclear Power Plant.

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

Sincerely,

Douglas V Pickett

Douglas V. Pickett, Sr. Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosures: 1. Amendment No. 112 to License No. NPF-58
2. Safety Evaluation

cc w/encls: See next page

Mr. John K. Wood
Vice President - Nuclear, Perry
FirstEnergy Nuclear Operating Company
P.O. Box 97, A200
Perry, OH 44081

June 1, 2000

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Sincerely,
/RA/
Douglas V. Pickett, Sr. Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-440
Enclosures: 1. Amendment No. 112 to License No. NPF-58
2. Safety Evaluation

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*See memorandum to D Pickett
**See memoranda to A Mendiola
***See memoranda to S Bajwa

cc:

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

FIRSTENERGY NUCLEAR OPERATING COMPANY

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 112
License No. NPF-58

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

    A. The application for amendment by the FirstEnergy Nuclear Operating Company (the licensee) dated September 9, 1999, as supplemented by submittals dated March 1, March 13, and May 11, 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, 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 amendment will not be inimical to the common defense and security or to the health and safety of the public; and

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-58 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 112 are hereby incorporated into this license. The FirstEnergy Nuclear Operating Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented not later than 45 days after completion of the next scheduled refuel outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: June 1, 2000

ATTACHMENT TO LICENSE AMENDMENT NO. 112

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

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

Remove	Insert
Operating License, Page 4	Operating License, Page 4
1.0-5	1.0-5
2.0-1	2.0-1
3.1-9	3.1-9
3.1-18	3.1-18
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
3.3-6	3.3-6
3.3-7	3.3-7
3.3-8	3.3-8
3.3-9	3.3-9
3.3-16	3.3-16
3.3-17	3.3-17
3.3-19	3.3-19
3.3-26	3.3-26
3.3-27	3.3-27
3.3-28	3.3-28
3.3-54	3.3-54
3.7-13	3.7-13
5.0-15a	5.0-15a

renewal. Such sale and leaseback transactions are subject to the representations and conditions set forth in the above mentioned application of January 23, 1987, as supplemented on March 3, 1987, as well as the letter of the Director of the Office of Nuclear Reactor Regulation dated March 16, 1987, consenting to such transactions. Specifically, a lessor and anyone else who may acquire an interest under these transactions are prohibited from exercising directly or indirectly any control over the licenses of PNPP Unit 1. For purposes of this condition the limitations of 10 CFR 50.81, as now in effect and as may be subsequently amended, are fully applicable to the lessor and any successor in interest to that lessor as long as the license for PNPP Unit 1 remains in effect; these financial transactions shall have no effect on the license for the Perry Nuclear facility throughout the term of the license.

(b) Further, the licensees are also required to notify the NRC in writing prior to any change in: (i) the terms or conditions of any lease agreements executed as part of these transactions; (ii) the PNPP Operating Agreement; (iii) the existing property insurance coverage for PNPP Unit 1; and (iv) any action by a lessor or others that may have an adverse effect on the safe 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

FENOC is authorized to operate the facility at reactor core power levels not in excess of 3758 megawatts thermal (100% power) in accordance with the conditions specified herein.

(2) Technical Specifications

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

(3) Antitrust Conditions

a. Cleveland Electric Illuminating Company, Ohio Edison Company, OES Nuclear, Inc., Pennsylvania Power Company, and the

1.1 Definitions (continued)

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 3758 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. Exceptions are stated in the individual surveillance requirements.

(continued)

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 ≥ 1.09 for two recirculation loop operation or ≥ 1.11 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)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. ---------NOTE--------- Not applicable when THERMAL POWER > 19.0% RTP. ---------------------- Two or more inoperable control rods not in compliance with banked position withdrawal sequence (BPWS) and not separated by two or more OPERABLE control rods.	D.1 Restore compliance with BPWS. <u>OR</u> D.2 Restore control rod to OPERABLE status.	4 hours 4 hours
E. Required Action and associated Completion Time of Condition A, C, or D not met. <u>OR</u> Nine or more control rods inoperable.	E.1 Be in MODE 3.	12 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 Control Rod Pattern

LCO 3.1.6 OPERABLE control rods shall comply with the requirements of the banked position withdrawal sequence (BPWS).

APPLICABILITY: MODES 1 and 2 with THERMAL POWER ≤ 19.0% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more OPERABLE control rods not in compliance with BPWS.	A.1 --------NOTE--------- Affected control rods may be bypassed in Rod Action Control System (RACS) in accordance with SR 3.3.2.1.9. --------------------- Move associated control rod(s) to correct position.	8 hours
	OR	
	A.2 Declare associated control rod(s) inoperable.	8 hours

(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 <u>AND</u> 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 < 38% 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 while operating at ≥ 23.8% RTP.	7 days
SR 3.3.1.1.3	Adjust the channel to conform to a calibrated flow signal.	7 days
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

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
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 ≥ 38% RTP.	18 months
SR 3.3.1.1.17	Calibrate 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	≤ 0.628 W + 63.8% RTP and ≤ 113% RTP(b)

(continued)

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

(b) Allowable Value is ≤ 0.628 W + 43.5% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

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	≤ 1079.7 psig
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	≥ 177.1 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	≤ 220.1 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 psig

(continued)

Table 3.3.1.1-1 (page 3 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
8. Scram Discharge Volume Water Level — High					
a. Transmitter/Trip Unit	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	≤ 38.87 inches
	5(a)	2	I	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	≤ 38.87 inches
b. Float Switch	1,2	2	H	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 626 ft 11.5 inches elevation
	5(a)	2	I	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15	≤ 626 ft 11.5 inches elevation
9. Turbine Stop Valve Closure	≥ 38% RTP	4	E	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.18	≤ 7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure — Low	≥ 38% RTP	2	E	SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.15 SR 3.3.1.1.16 SR 3.3.1.1.18	≥ 465 psig
11. Reactor Mode Switch — Shutdown Position	1,2	2	H	SR 3.3.1.1.12 SR 3.3.1.1.15	NA
	5(a)	2	I	SR 3.3.1.1.12 SR 3.3.1.1.15	NA
12. Manual Scram	1,2	2	H	SR 3.3.1.1.5 SR 3.3.1.1.15	NA
	5(a)	2	I	SR 3.3.1.1.5 SR 3.3.1.1.15	NA

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

SURVEILLANCE REQUIREMENTS

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

1. Refer to Table 3.3.2.1-1 to determine which SRs apply for each Control Rod Block 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 control rod block capability.

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.1	------------------NOTE------------------ Not required to be performed until 1 hour after THERMAL POWER is > 66.7% RTP. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.2	------------------NOTE------------------ Not required to be performed until 1 hour after THERMAL POWER is > 33.3% RTP and ≤ 66.7% RTP. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.3	------------------NOTE------------------ Not required to be performed until 1 hour after any control rod is withdrawn in MODE 2. -- Perform CHANNEL FUNCTIONAL TEST.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.1.4	------------------NOTE------------------ Not required to be performed until 1 hour after THERMAL POWER is ≤ 19% RTP in MODE 1. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.1.5	Calibrate the low power setpoint trip units. The Allowable Value shall be > 19% RTP and ≤ 33.3% RTP.	92 days
SR 3.3.2.1.6	Verify the RWL high power Function is not bypassed when THERMAL POWER is > 66.7% RTP.	92 days
SR 3.3.2.1.7	Perform CHANNEL CALIBRATION.	184 days
SR 3.3.2.1.8	------------------NOTE------------------ Not required to be performed until 1 hour after reactor mode switch is in the shutdown position. -- Perform CHANNEL FUNCTIONAL TEST.	18 months

(continued)

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS
1. Rod Pattern Control System			
a. Rod withdrawal limiter	(a)	2	SR 3.3.2.1.1 SR 3.3.2.1.6 SR 3.3.2.1.9
	(b)	2	SR 3.3.2.1.2 SR 3.3.2.1.5 SR 3.3.2.1.7 SR 3.3.2.1.9
b. Rod pattern controller	1[(c)],2	2	SR 3.3.2.1.3 SR 3.3.2.1.4 SR 3.3.2.1.5 SR 3.3.2.1.7 SR 3.3.2.1.9
2. Reactor Mode Switch — Shutdown Position	(d)	2	SR 3.3.2.1.8

(a) THERMAL POWER > 66.7% RTP.

(b) THERMAL POWER > 33.3% RTP and ≤ 66.7% RTP.

(c) With THERMAL POWER ≤ 19.0% RTP.

(d) Reactor mode switch in the shutdown position.

3.3 INSTRUMENTATION

3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

LCO 3.3.4.1 Two channels per trip system for each EOC-RPT instrumentation Function listed below shall be OPERABLE:

a. Turbine Stop Valve (TSV) Closure; and

b. Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure—Low.

APPLICABILITY: THERMAL POWER ≥ 38% RTP with any recirculation pump in fast speed.

ACTIONS

-------------------------------------NOTE-------------------------------------
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Restore channel to OPERABLE status. <u>OR</u> A.2 --------NOTE--------- Not applicable if inoperable channel is the result of an inoperable breaker. --------------------- Place channel in trip.	72 hours 72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One or more Functions with EOC-RPT trip capability not maintained.	B.1	Restore EOC-RPT trip capability.	2 hours
C. Required Action and associated Completion Time not met.	C.1	Remove the associated recirculation pump fast speed breaker from service.	4 hours
	<u>OR</u>		
	C.2	Reduce THERMAL POWER to < 38% RTP.	4 hours

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 EOC-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.1	Perform CHANNEL FUNCTIONAL TEST.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.2	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. TSV Closure: ≤ 7% closed; and b. TCV Fast Closure, Trip Oil Pressure—Low: ≥ 465 psig.	18 months
SR 3.3.4.1.3	Perform LOGIC SYSTEM FUNCTIONAL TEST, including breaker actuation.	18 months
SR 3.3.4.1.4	Verify TSV Closure and TCV Fast Closure, Trip Oil Pressure—Low Functions are not bypassed when THERMAL POWER is ≥ 38% RTP.	18 months
SR 3.3.4.1.5	------------------NOTE------------------ Breaker arc suppression time may be assumed from the most recent performance of SR 3.3.4.1.6. -- Verify the EOC-RPT SYSTEM RESPONSE TIME is within limits.	18 months on a STAGGERED TEST BASIS
SR 3.3.4.1.6	Determine RPT breaker arc suppression time.	60 months

Table 3.3.6.1-1 (page 1 of 6)
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 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 14.3 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 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 795.2 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 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 256.5 psid
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 3.3.6.1.4 SR 3.3.6.1.5	≥ 7.6 inches Hg vacuum
e. Main Steam Line Pipe Tunnel Temperature - High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 158.9°F
f. Main Steam Line Turbine Building Temperature-High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 138.9°F
g. Manual Initiation	1,2,3	2	G	SR 3.3.6.1.5	NA
2. Primary Containment and Drywell Isolation					
a. Reactor Vessel Water Level - Low Low, Level 2	1,2,3	2(b)	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 127.6 inches

(continued)

(a) With any turbine stop valve not closed.

(b) Required to initiate the associated drywell isolation function.

3.7 PLANT SYSTEMS

3.7.6 Main Turbine Bypass System

LCO 3.7.6 The Main Turbine Bypass System shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 23.8% RTP.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.6.1 Verify one complete cycle of each main turbine bypass valve.	31 days
SR 3.7.6.2 Perform a system functional test.	18 months
SR 3.7.6.3 Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	18 months

5.5 Programs and Manuals

5.5.12 Primary Containment Leakage Rate Testing Program (continued)

- BN-TOP-1 methodology may be used for Type A tests.
- The corrections to NEI 94-01 which are identified on the Errata Sheet attached to the NEI letter, "Appendix J Workshop Questions and Answers," dated March 19, 1996 are considered an integral part of NEI 94-01.
- The containment isolation check valves in the Feedwater penetrations are tested per the Inservice Testing Program (Technical Specification 5.5.6).

The peak calculated primary containment internal pressure for the design basis loss of coolant accident is 6.40 psig. For conservatism P_a is defined as 7.80 psig.

The maximum allowable primary containment leakage rate, L_a, shall be 0.20% of primary containment air weight per day at the peak containment pressure (P_a).

Leakage rate acceptance criteria are:

a. Primary containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. However, during the first unit startup following testing performed in accordance with this Program, the leakage rate acceptance criteria are $< 0.6\ L_a$ for the Type B and Type C tests, and $\leq 0.75\ L_a$ for the Type A tests;

b. Air lock testing acceptance criteria are:

   1) Overall air lock leakage rate is $\leq$ 2.5 scfh when tested at $\geq P_a$.

   2) For each door, leakage rate is $\leq$ 2.5 scfh when the gap between the door seals is pressurized to $\geq P_a$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Primary Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Primary Containment Leakage Rate Testing Program.

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 112 TO FACILITY OPERATING LICENSE NO. NPF-58

FIRSTENERGY NUCLEAR OPERATING COMPANY

PERRY NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-440

I INTRODUCTION

By letter dated September 9, 1999 (Reference 1), and supplemented by submittals dated March 1 (Reference 2), March 13, (Reference 3), and May 11, 2000 (Reference 15) FirstEnergy Nuclear Operating Company (the licensee) requested Nuclear Regulatory Commission (NRC) approval of a license amendment to increase the 100 percent authorized rated thermal power for Perry Nuclear Power Plant (Perry) from 3579 megawatts (MWt) to 3758 MWt. This represents an increase of 5 percent. Included in the amendment request is proprietary General Electric (GE) Nuclear Energy Topical Report NEDE-32907P (Reference 4) presenting the safety analysis results supporting the proposed power uprate. The licensee based the amendment request on the boiling water reactor (BWR) generic power uprate guidelines presented in NEDC-31897P-A (Reference 5).

The supplemental submittals of March 1, March 13, and May 11, 2000, contained clarifying information and did not change the initial no significant hazards consideration determination and did not expand the scope of the original Federal Register notice.

Overview

The staff has approved a number of BWR facilities for power uprate using the NRC-approved generic format and content found in NEDC-31897P-A (Reference 5). An increase in electrical output of a BWR plant is accomplished primarily by the generation and supply of higher steam flow for the turbine-generator. Perry, like most BWR plants as originally licensed, has an as-designed equipment and system capability to accommodate steam flow rates at least 5 percent above the original rating. The licensee's submittal was characterized by the following:

- All safety aspects of the plant that are affected by the increase in thermal power, including nuclear steam supply and balance-of-plant systems, were evaluated.

- There is no change in the established 10 CFR 50.2 design basis of the plant or licensing basis acceptance criteria of the plant.

- Evaluations were performed using NRC-approved analysis methods.

- The Nuclear Steam Supply System (NSSS) was reviewed to confirm that it continues to comply with the acceptance criteria in the Perry Updated Safety Analysis Report (USAR).

- No modification to a safety-related system is expected; however, any modifications will be implemented via 10 CFR 50.59.

- An assessment against the 10 CFR 50.92 significant hazards consideration criteria as required by 10 CFR 50.91(a) was performed.

- All systems and components impacted by power uprate were reviewed to assure there are no significant challenges to safety systems.

- Compliance with existing plant environmental regulations was reviewed.

The Perry facility was originally licensed at 3579 MWt. The original safety analysis assumed that the reactor had been operating continuously at a power level at least 1.02 times the licensed power level (i.e., 102 percent rated power or 3651 MWt). Many of the original analyses had already been performed at 105 percent steam flow (equivalent to 104.2 percent thermal power). The power uprate included in this evaluation is a 5 percent thermal power uprate which corresponds to 3758 MWt. The power uprate safety analyses are based on a power level of at least 1.02 times 3758 MWt or 3833 MWt. However, some analyses are performed at 100 percent rated power because the Regulatory Guide 1.49, "Power Levels of Nuclear Power Plants," 2 percent power factor is already accounted for in the analysis methods.

The licensee plans to implement the power uprate within 45 days following completion of the next scheduled refuel outage (Reference 15). The licensee plans to achieve the higher power level by increasing core thermal power, steam flow, and feedwater flow, but without increasing maximum core flow or reactor operating dome pressure. Plant-specific and generic evaluations were used to support the proposed power uprate. The generic evaluations are based on a slightly smaller power increase than is requested for Perry (4.2 vs. 5 percent).

The licensee's engineering evaluation indicates that modifications to the high pressure turbine nozzles may be required to pass full uprated steam flow. Since the high pressure turbine cannot be modified while at power, this could limit turbine operation to approximately 104 percent power if the power uprate is implemented during the current Cycle 8 operation. If necessary, the high pressure turbine could be modified by minor grinding of the first stage nozzles during the next refueling outage to permit increased steam flow.

II <u>EVALUATION</u>

1.0 TECHNICAL SPECIFICATION CHANGES

The following technical specification (TS) changes have been proposed:

Increase the value for rated thermal power (RTP) from 3579 to 3758 megawatts to reflect the uprated power level.

- Operating License paragraph 2.C(1)
- TS 1.1, "Definitions," Rated Thermal Power

Revise the value for the reactor core safety limit at low pressure and low flow conditions (from 25 to 23.8% RTP) to maintain the pre-uprate thermal power basis in terms of absolute power.

- TS 2.1, "Safety Limits"
- TS 3.2.1, "Average Planar Linear Heat Generation Rate (APLHGR)"
- TS 3.2.2, "Minimum Critical Power Ratio (MCPR)"
- TS 3.2.3, "Linear Heat Generation Rate (LHGR)"
- TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"
- TS 3.7.6, "Main Turbine Bypass System"

Revise the Allowable Value for the Rod Pattern Controller - low power setpoint (from 20 to 19%) to maintain the pre-uprate thermal power basis in terms of absolute power.

- TS 3.1.3, "Control Rod OPERABILITY"
- TS 3.1.6, "Control Rod Pattern"
- TS 3.3.2.1, "Control Rod Block Instrumentation"

Revise the value for the automatic bypass of turbine stop valve closure and turbine control valve fast closure signals that initiate scram and end-of-cycle recirculation pump trip (EOC-RPT) (from 40 to 38% RTP) to maintain the pre-uprate thermal power basis in terms of absolute power

- TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"
- TS 3.3.4.1, "End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation"

Revise the Allowable Values for the two-loop and single-loop Average Power Range Monitor Flow Biased Simulated Thermal Power - High setpoints (from 0.66W + 67% to 0.628W + 63.8% RTP and from 0.66W + 45.7% to 0.628W + 43.5% RTP, respectively) to maintain the pre-uprate thermal power basis in terms of absolute power.

- TS 3.3.1.1, "Reactor Protection System (RPS) Instrumentation"

Revise the Allowable Values for the Rod Withdrawal Limiter - high power and low power setpoints (from 70 to 66.7% RTP and from 35 to 33.3% RTP, respectively) to maintain the pre-uprate thermal power basis in terms of absolute power

- TS 3.3.2.1, "Control Rod Block Instrumentation"

Revise the Allowable Value for the Main Steam Line Flow - High setpoint (from 189.3 to 256.5 psid) to maintain the pre-uprate steam flow basis in terms of percent rated steam flow

- TS 3.3.6.1, "Primary Containment and Drywell Isolation Instrumentation"

Revise the discussion of peak calculated containment pressure to reflect the results of the new containment analysis (6.40 psig) and retain the current value of P_a (7.80 psig) for leakage rate testing.

- TS 5.5.12, "Primary Containment Leakage Rate Testing Program"

2.0 REACTOR CORE AND FUEL PERFORMANCE

2.1 Fuel Design and Operation

Power uprate increases the power density of the plant and has effects on operating flexibility, reactivity characteristics, and energy requirements. The core power distribution would be changed to achieve increased core power while limiting the absolute power in any individual fuel bundle.

At uprated power conditions, all fuel and core design limits continue to be met by configuration of fuel enrichment and burnable poison, supplemented by core management control rod pattern or core flow adjustments. The power uprate evaluations considered GE10, GE11, and GE12 fuel types. More advanced fuel types may be used to provide operating flexibility and to maintain cycle length. Core configurations will be evaluated on a cycle-specific basis in accordance with the plant 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 power uprate. The licensee conducted parametric studies that indicated that the uprate can be accommodated. The studies consisted of a standard reload licensing analysis of the current Cycle 8 loading pattern at uprate conditions and a fuel cycle analysis of Cycle 9 using GE12 fresh fuel at uprated power conditions. Thermal-hydraulic design and operating limits assure an acceptably low probability of boiling transition-induced cladding failure, 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 reload core designs for operation at the uprated power will take into account the applicable limits to assure acceptable margins between the licensing limits and their corresponding operating values.

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. The impact of higher power operation on radiation sources and design basis accident doses are discussed in the licensee's submittal.

Perry is currently operating with minor fuel leaks that have been controlled through flux suppression by insertion of selected control rods. The licensee has analyzed the effects of inserted control rods on fuel performance at uprated power conditions. Operation at uprated power conditions is not expected to have an effect on existing fuel leaks or any discernable impact on overall fuel integrity. The licensee will continue monitoring activities to confirm that fuel leakage remains controlled.

2.2 Thermal Limits Assessment

The reference loading pattern for Perry Cycle 8 at uprated power conditions was used for the uprate evaluation. The topical report states that cycle-specific core configurations will be evaluated for future reloads to confirm power uprate capability and to establish and confirm cycle-specific limits.

2.2.1 Minimum Critical Power Ratio (MCPR) Operating Limit

The operating limit MCPR is determined on a cycle-specific basis from the results of reload analysis, as described in GE report NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactors Power Uprate," (Reference 6). No significant change in operation is anticipated due to the uprate based on experience from operating BWR uprates. The operating limit MCPR for the uprated power condition was determined for limiting fuel type as discussed in Section 9.1, Reactor Transients.

2.2.2 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Maximum Linear Heat Generation Rate (LHGR) Operating Limits

No significant change in operation is anticipated due to the power uprate based on experience from operating BWR uprates. The emergency core cooling system (ECCS) evaluation for power uprate addressed in Section 4.3 shows that no change in the MAPLHGR limits is needed for the uprate. The LHGR limits are dependent on fuel type, apply regardless of power level, and therefore do not change under uprate conditions.

2.3 Reactivity Characteristics

The reference loading pattern for Perry Cycle 8 at uprated power conditions was used for the uprate evaluation. As is current practice, cycle-specific core configurations will be evaluated for future reloads to confirm core capabilities and cycle-specific limits.

Operation at higher power could reduce the excess reactivity during the fuel cycle. The loss of reactivity is not expected to significantly degrade the ability to manage the power distribution through the cycle to achieve the target power level. Lower reactivity would result in an earlier all-rods-out condition. Fuel cycle redesign can obtain sufficient excess reactivity to match the desired cycle length, therefore, lower reactivity is not a concern.

The increase in hot reactivity may result in less hot-to-cold reactivity difference and, therefore, smaller cold shutdown margins. However, this loss in margin can be accommodated through core design. If needed, a bundle design with improved shutdown margin characteristics can be used to preserve the flexibility between hot and cold reactivity requirements for future cycles. Current design and TS cold shutdown margin requirements are maintained for power uprate operating conditions.

2.3.1 Power/Flow Operating Map

The Perry reactor operating domain for two reactor recirculation loop operation at the uprated power condition is described in Reference 4. It is defined by the proposed rated power corresponding to 105 percent of the current licensed power, the existing maximum extended load line limit (MELLL) upper load line extended to the uprated 100 percent power, and the existing increased core flow (ICF) line extended up to the uprated 100 percent power. The boundaries define an increase in the extent of the operating domain above the currently licensed RTP between the extended MELLL upper load line and the ICF line. There is no change in the extent of the single reactor recirculation loop operating domain due to the power uprate.

The licensee re-scaled the power axis for the two-loop operating domain, consistent with the generic methodology in References 5 and 6, as well as previously approved plant-specific power uprates.

2.4 Stability

Perry is currently operating under the Boiling Water Reactors Owners Group (BWROG) reactor stability interim corrective actions (ICA) as detailed in the licensee response to Generic Letter 94-02, "Long-Term Solutions and Upgrade of Interim Operating Recommendations for Thermal-Hydraulic Instabilities in Boiling Water Reactors" (Reference 7). The licensee is in the process of implementing reactor long-term solution Option III. Long-term solution Option III is discussed in Reference 8. The licensee addressed the effect of the power uprate on both the interim corrective actions and on the long-term stability solution.

The licensee conducted an evaluation to determine the effect of the power uprate on core stability interim corrective actions following the guidelines in Reference 6. To ensure the same level of protection against the occurrence of thermal-hydraulic instability, the instability exclusion region boundaries are unchanged with respect to absolute power level. The licensee provided a reactor operating domain for two reactor recirculation loop operation illustrating the ICA regions for the uprated power conditions. The approach used by the licensee is consistent with the generic method discussed in Reference 6.

Perry is implementing long-term stability Option III. Under this option, oscillation power range monitor signals are monitored to determine when a reactor scram is needed to disrupt an oscillation. When Option III is implemented, the power-to-flow operating map will be defined in plant procedures to include an armed region that is used in Option III. The armed region will be modified for uprated power conditions to maintain the current absolute power and flow coordinates. The licensee indicates that its stability-based MCPR calculations show no significant changes from current conditions. This discussion does not constitute a staff review of the licensee's proposed implementation of Option III.

2.5 Reactivity Control

2.5.1 Control Rod Drive (CRD) Hydraulic System

The control rod drive (CRD) system changes core reactivity by positioning neutron absorbing control rods within the reactor. It must scram the reactor by rapidly inserting withdrawn rods into the core. The CRD system was evaluated at the uprated operating conditions.

The CRD scram performance meets current TS requirements under power uprate conditions. The CRD scram performance was evaluated for a reactor dome pressure of 1025 psig and an additional 35 psid for the vessel bottom head.

For CRD insertion and withdrawal, the required minimum differential pressure between the hydraulic control unit and the vessel bottom head is 250 psid. The CRD pumps were evaluated against this requirement and were found to have sufficient capacity. The flows needed for CRD cooling and function are assured by automatic operation of the system flow control valve.

The licensee stated that the control rod drive mechanisms (CRDMs) have been designed in accordance with the code of record, the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section III, 1974 Edition with addenda up to and including Winter 1975. The components of the CRDM, which form part of the primary pressure boundary, have been designed for a bottom head pressure of 1250 psig, which is higher than the reactor bottom head pressure of 1060 psig for normal and uprated power conditions.

Furthermore, the maximum calculated stress for the CRDM indicator tube is 23,830 psi which is less than the allowable stress limit of 31,050 psi. The analysis of cyclic operation of the CRDM resulted in a maximum cumulative fatigue 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 staff concludes that the CRD system will continue to meet its design basis and performance requirements at uprated power conditions.

3.0 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

3.1 Nuclear System Pressure Relief

The nuclear system pressure relief system prevents over-pressurization of the nuclear system during abnormal operating transients. The plant safety/relief valves (SRVs) and scram functions provide this protection. The setpoints for the SRVs at Perry are not changed for power uprate.

The nominal SRV setpoints ensure that adequate differences between operating pressure and SRV setpoints are maintained (i.e., the "simmer margin"), and that an increase in the number of unnecessary SRV actuations does not result from operation at uprated power conditions. Based on its analysis, the licensee concluded that the nuclear system pressure relief system was capable of providing sufficient overpressure protection at uprated power conditions. The conclusion is consistent with the corresponding discussion in References 5 and 6.

3.1.1 SRV Setpoint Tolerance

SRV setpoint tolerance is independent of power uprate. The power uprate evaluations are performed using the 3 percent SRV setpoint (tolerance) analytical limits as in the safety analysis basis. In-service surveillance testing of Perry's SRVs has not shown a significant propensity for high setpoint drift of more than 3 percent. Of 114 SRV tests from the "as-found" setpoint lift verification tests performed from 1989 to 1997, only 4 SRVs were found to vary from their setpoint by more than ±3 percent.

3.2 Reactor Overpressure Protection Analysis

The design pressure of the reactor pressure vessel (RPV) remains at 1250 psig. The ASME code allowable peak pressure for the reactor vessel 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, which is described in the General Electric Standard Application for Reactor Fuel (GESTAR) topical report (Reference 9). For power uprate, the analysis assumes the event initiates at a reactor dome pressure of 1045 psig, which is the TS limiting condition for operation (LCO) maximum dome pressure. Six of the dual mode safety/relief valves were assumed to be out of service for the overpressure protection analysis. The analysis conservatively assumes that the 13 dual mode safety/relief valves having the highest pressure relief setpoints are operable. At uprated power conditions, a peak reactor bottom head pressure of 1295 psig was calculated, which is within the 1375 psig ASME limit.

3.3 Reactor Vessel Evaluation

The staff's review focused on whether or not the proposed licensing action would reduce the margins of safety that have been established in the licensing basis to ensure the structural integrity of the Perry reactor coolant pressure boundary, and in particular, to ensure the integrity of the RPV. The only reactor coolant pressure boundary component affected by the power uprate is the RPV, because it is the only component that receives significant amounts of neutron radiation.

Plant parameters that could be affected from a power uprate include: pressure-temperature (P-T) limits and adjusted reference temperature (ART) calculations, upper shelf energy (USE) drop for the RPV materials, and the surveillance capsule withdrawal schedule.

The staff evaluates the P-T limits based on the following NRC regulations and guidance:

- 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements";
- Generic Letter (GL) 88-11, "NRC Position on Radiation Embrittlement Of Reactor Vessel Materials And Its Impact On Plant Operations";
- GL 92-01, Revision 1, Supplement 1, "Reactor Vessel Structural Integrity";
- Regulatory Guide (RG) 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials"; and
- Standard Review Plan (SRP) 5.3.2, "Pressure-Temperature Limits and Pressurized Thermal Shock."

GL 88-11 advised licensees that the staff would use RG 1.99, Revision 2, to review P-T limit curves. RG 1.99, Revision 2, contains methodologies for determining the increase in transition temperature and the decrease in USE from neutron radiation. GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the staff's review of the P-T limit curves. Appendix G to 10 CFR Part 50 requires that P-T limit curves for the RPV be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Code.

The staff evaluates the surveillance program based upon Appendix H to 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements." Appendix H to 10 CFR Part 50 includes criteria to monitor changes in the fracture toughness properties of ferritic materials in the RPV beltline region of the light-water nuclear power reactors which result from exposure of these materials to neutron irradiation and the thermal environment. Appendix H to 10 CFR Part 50 endorses American Society for Testing and Materials (ASTM) E185, "Surveillance Tests for Nuclear Reactor Vessels." Appendix H states that "the design of the surveillance program and the withdrawal schedule must meet the requirements of the edition of ASTM E185 that is current on the issue date of the ASME Code to which the reactor vessel was purchased."

The licensee evaluated the integrity of the RPV at the revised design conditions in terms of impact due to the neutron fluence. More specifically, the licensee provided an assessment on the impact of the power uprate to: (1) the ART of the limiting RPV material, (2) the need to revise the P-T limit curves, (3) the change in the predicted USE drop for the RPV materials, and (4) determine whether changes in the RPV surveillance program are necessary.

3.3.1 Reactor Vessel Integrity/Neutron Irradiation

Several analyses are performed to determine the impact that the neutron irradiation has on the integrity of the reactor vessel. The most critical area is the beltline region of the reactor vessel since it is predicted to be most susceptible to neutron damage. In regard to the power uprate and the reactor vessel integrity, the analyses should include an evaluation of the (1) ART calculations, (2) heat-up and cooldown P-T limit curves, (3) USE, and (4) surveillance capsule withdrawal schedule. It should be noted that these evaluations could be affected by changes in the neutron fluences and operating temperatures and pressures that result from a power uprate.

In regard to the ART calculation, the licensee stated that the highest current ART end of license (EOL) value for the RPV (for the axial weld, heat number 627260) remains as 84°F. With a nominal 5 percent increase in fluence, the licensee determined that the change in the ART value would not be significant, and therefore, revised P-T curves are not required.

In addition, the licensee found that the revised design conditions showed continued compliance with the existing design and licensing criteria for the Perry RPV. The licensee further explained that with regard to the application of the requirements of 10 CFR Part 50, Appendices G and H, to the Perry RPV materials:

(a) The USE values would still remain above the 50 ft-lb value throughout the life of the vessel.

(b) There is no significant change in the 32 effective full power year (EFPY) shift in adjusted reference temperature, and therefore, the existing P-T curves remain bounding for power operation up to 3758 MWt.

(c) No changes in the Appendix H program (the RPV surveillance program) are required.

The staff independently calculated the ARTs at the 1/4 thickness (1/4T) position in the vessel wall for the Perry RPV beltline materials, considering the 5 percent power uprate. The staff verified that the licensee used a fluence increase that was calculated to be proportional to the 5 percent power increase. In calculating the ARTs for the beltline materials of Perry, the staff used the higher fluence of 5×10^{18} n/cm^2, as proposed by the licensee for the 5 percent power uprate. The staff independently verified that the limiting material for Perry was the axial weld, heat number 627260. In addition, the staff independently calculated the shift in RT_{NDT} as a result of the power uprate, and determined that the uprate had a negligible effect on the value. The shift in the RT_{NDT} remained at 58°F, which resulted with an ART at EOL remaining at 84°F.

The staff also independently evaluated the USE, based upon the revised fluence value of 5 x 10^{18} n/cm^2, as a result of the power uprate. The staff determined that the minimum USE at EOL for the beltline materials of Perry is 75 ft-lb. Therefore, the staff verified that the USE remains greater than 50 ft-lb for the design life of the Perry RPV and maintains the margin requirements of 10 CFR 50, Appendix G.

In evaluating the surveillance program, the staff determined that the predicted transition temperature shift at vessel inside surface, as a result of the 5 percent power uprate, remained below 100°F. Therefore, the staff determined that the minimum number of capsules to be withdrawn and the capsule withdrawal schedule for Perry still meets the ASTM E 185-82 Standard. Since the minimum number of capsules and withdrawal schedule meet the ASTM E 185-82 Standard, the surveillance program is in compliance with Appendix H, 10 CFR Part 50, and is acceptable.

Based on the staff's review of the licensee's submittal, the staff found that the issues regarding the integrity and operation of the RPV had been adequately addressed. The staff also determined that as a result of the power uprate, the Perry RPV still meets the requirements of Appendices G and H of 10 CFR Part 50.

3.3.2 Reactor Internals Evaluation

3.3.2.1 Reactor Internal Pressure Differences

The licensee evaluated the reactor vessel and internal components in accordance with the current licensing basis. Load combinations include reactor internal pressure difference (RIPD), SRV actuations, LOCA, annulus pressurization (AP), jet reaction (JR), acoustic loads, thermal loads, seismic, and fuel lift loads. The seismic loads are unaffected by the power uprate. The fuel lift loads due to the combined effects of the uplift pressure and dynamic loads are bounded by the design-basis loads. There is no increase in the dynamic loads due to LOCA, SRV, and AP because the existing design loads are bounding for the power uprate. The licensee indicated that the acoustic loads in the vessel annulus, as a result of recirculation line break, are not affected by the power uprate. In its evaluation of the power uprate, the licensee takes into consideration the effects of changes in thermal and flow conditions, the jet load and the RIPDs. The licensee recalculated RIPDs for the proposed power uprate in Reference 4, for normal, upset and faulted conditions.

The stresses and cumulative fatigue usage factors (CUFs) for the reactor internal and vessel components were evaluated by the licensee in accordance with the code of record at Perry, the ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition with addenda to and including Winter 1972, with certain exceptions and modifications as specified in Perry USAR. The load combinations for normal, upset and faulted conditions were considered in the evaluation. The maximum stresses for critical components of the reactor internals were calculated in Reference 4 for the power uprate conditions. The licensee indicated that the stresses were determined by scaling the existing (pre-uprate) stresses based on bounding uprated conditions (pressure, temperature and flow). The licensee concluded that the calculated stresses are less than the allowable code limits.

3.3.2.2 Structural Evaluation

In Reference 4, the licensee indicated that the evaluations of structural integrity of the RPV were performed considering operating conditions such as feedwater flow and temperature, and steam flow, that are affected by the power uprate. The licensee provided the calculated CUFs in Reference 4. The licensee concluded that maximum calculated stresses are within allowable limits. The staff finds that the methodology used by the licensee is consistent with the NRC-approved methodology in Appendix I of Reference 5, and is, therefore, acceptable.

3.3.2.3 Flow Induced Vibration

The licensee assessed the potential for flow-induced vibration based on the vibration data recorded during startup testing at Perry, the GE prototype BWR/6 plant vibration data for the reactor internal components, and on operating experience from other GE BWR plants. The vibration levels were calculated by extrapolating the recorded vibration data to power uprate conditions and compared to the plant allowable limits. The stresses at critical locations were calculated based on the extrapolated vibration peak response displacements and found to be within the GE allowable design criteria of 10 ksi for acceptance. The licensee indicated that stress values less than 10 ksi are within the endurance limit without the need to compute the CUF for the component due to flow-induced vibration. The licensee concluded that vibration levels of all safety-related reactor internal components are within the acceptance criteria. The staff finds this acceptable in comparison to the ASME criteria of 13.6 ksi peak vibration response.

Based on our review of the information provided by the licensee, the staff finds that the maximum stresses and fatigue usage factors are within the code-allowable limits, and concludes that the reactor vessel and internal components will continue to maintain their structural integrity for the power uprate condition.

3.4 Reactor Recirculation System

The licensee's evaluation of the reactor recirculation system performance at uprated power determined that the effects on the recirculation system and its components are acceptable. The system pressures and temperatures remain virtually unchanged. The drive flow increases by approximately 0.6 percent at maximum thermal power and core flow, which is within the capability of the recirculation system.

The cavitation protection interlock for the recirculation pumps and jet pumps is expressed in terms of a temperature difference between the reactor vessel dome temperature and the recirculation suction temperature in each recirculation loop. The cavitation protection interlock for the flow control valves is expressed in terms of feedwater flow. These interlocks prevent cavitation during low power conditions and are affected only slightly at increased power levels. An evaluation of the net positive suction head (NPSH) for the recirculation pumps, jet pumps, and flow control valves concluded that power uprate does not significantly increase the NPSH required or reduce the NPSH margin. The cavitation protection interlocks remain the same in terms of absolute thermal power, but their representations on the power/flow map change because of the scale change on the power axis.

Because power uprate has a negligible effect on recirculation system operating conditions, differences in the operation of system components is acceptably small. The staff concurs with the licensee's conclusion that operation at uprated power would be well within the capability of the recirculation system.

3.5 Reactor Coolant Pressure Boundary Piping

The licensee evaluated the effects of the power uprate condition, including higher flow rate, temperature, pressure, fluid transients and vibration effects on the reactor coolant pressure boundary (RCPB) and the balance of plant (BOP) piping systems and components. The components evaluated included equipment nozzles, anchors, guides, penetrations, pumps, valves, flange connections, and pipe supports (including snubbers, hangers, and struts). The evaluation was performed using the original code of record specified in the Perry USAR, the code allowables, and analytical techniques similar to those used in the original and existing design-basis analysis. The licensee indicated that no new assumptions were introduced that were not in the original analyses.

The RCPB piping systems evaluated include the main steam piping, reactor recirculation piping, feedwater piping, RPV bottom head drain line, reactor water cleanup (RWCU), reactor vessel head vent line, reactor core isolation cooling (RCIC), core spray piping, high pressure core spray piping (HPCS), residual heat removal (RHR), safety/relief valve discharge line (SRVDL) piping and CRD piping. The licensee compared the increase in pressure, temperature and flow rate due to the power uprate against the same parameters used as input to the original design-basis analyses. The comparison resulted in the bounding percentage increases in stress for affected limiting piping systems. The bounding percentage increases are compared to the design margin between calculated stresses and the ASME allowable limits. As a result of such comparison, the licensee concluded that there are sufficient design margins to justify operation at the power uprate condition. The bounding percentage increases were applied to the highest calculated stresses, displacements, and the CUF at applicable piping system locations to determine the maximum power uprate calculated stresses, displacements and usage factors. This approach is consistent with the methodology as provided in Appendix K to Reference 5, which was approved by the NRC (Reference 14).

The licensee provided the calculated maximum stresses and CUFs at critical locations of the piping systems evaluated for the power uprate. Based on the information provided by the licensee, all calculated stresses are within ASME allowable limits and the calculated CUFs are less than the allowable limit of 1.0. The licensee also concluded that the evaluation showed compliance with all appropriate Code requirements for the piping systems evaluated and that power uprate will not have an adverse effect on the reactor coolant piping system design. The staff reviewed relevant portions of the evaluation and finds that the licensee's conclusions are acceptable.

3.6 Main Steamline Flow Restrictors

Regarding the assessment of the main steamline flow restrictor, the licensee stated that there is no impact on the structural integrity of the restrictor for the power uprate. In Section 3.2 of the power uprate license amendment request, the licensee indicated that a higher peak RPV transient pressure of 1295 psig results from the Perry plant operation at 3758 MWt conditions. This value remains below the ASME code limit of 1375 psig. Therefore, the main steamline flow restrictor will maintain its structural integrity following the power uprate since the restrictor was designed for a differential pressure of 1375 psig which exceeds that for uprated power conditions. The licensee evaluated the MSIVs by referring to the GE generic evaluation in Section 4.7 of Reference 6, which is applicable to the Perry power uprate. Also, the operating pressure and temperature for the Perry power uprate remain unchanged. The licensee concluded that the existing design pressure and temperature for the MSIVs are bounding for the power uprate and the ability of the MSIVs to perform their isolation function is not affected following the uprate condition. The staff concurs with the licensee's conclusion.

3.7 Main Steam Isolation Valves (MSIVs)

The MSIVs are part of the reactor coolant pressure boundary and perform a safety function (steamline isolation). The MSIVs must be able to close within the specified limits at all design and operating conditions upon receipt of a closure signal. They are designed to satisfy leakage limits set forth in the plant TSs. The licensee indicated that there are no changes in the operating conditions (primarily pressure) associated with power uprate when compared to the original operating conditions. The existing design pressure and temperature for the MSIVs is the same as the RCPB, and continue to bound the maximum operating pressure and temperature under power uprate conditions. Because reactor operating pressure does not increase with power uprate, the ability of the MSIVs to perform their isolation function is not affected.

The staff concurs with the licensee's conclusion that the plant operations at the proposed uprated power level will not affect the ability of the MSIVs to perform their isolation function because the reactor operating pressure does not increase with power uprate.

3.8 Reactor Core Isolation Cooling System (RCIC)

The RCIC system provides core cooling when the RPV is isolated from the main condenser, concurrent with loss of all feedwater flow, and the RPV pressure is greater than the maximum allowable for initiation of a low pressure core cooling system.

The design basis for the maximum required injection pressure is based on an SRV reopening setpoint. Since the reactor system pressure and the SRV setpoints remain unchanged for the power uprate, there is no change to the RCIC high pressure injection process parameters. The calculated minimum required RCIC injection rate at power uprate conditions remains within the specified system design injection flow rate of 700 gpm.

Operation of the RCIC system at uprated power conditions does not have any effect on the availability or the reliability of the system and does not invalidate the original design temperature and pressure conditions for system components. The RCIC startup transient response is dependent on reactor pressure and the initial responsiveness of the turbine control system. Because there is no increase in the reactor pressure, there is no increase in the potential for higher peak transient speeds on the startup and no increase in the potential for overspeed trip. Consequently, RCIC turbine operation with power uprate does not result in any changes to the startup transients or to system reliability.

Reevaluation of the RCIC turbine startup performance indicates acceptable transient speed peaks without implementation of the startup control modifications described in GE SIL 377, "RCIC Startup Transient Improvement with Steam Bypass" (Reference 10). Power uprate does not decrease the NPSH available for the RCIC pump or change the NPSH required above the specified design value. Surveillance testing and the infrequent demands for system injection would occur at the same reactor pressure, therefore, there is no change to RCIC system reliability rates. The RCIC system was evaluated for loss of feedwater transient events and is consistent with the bases and conclusions of the generic evaluation in Reference 6. Therefore, the staff concludes that the RCIC system is acceptable for power uprate.

3.9 Residual Heat Removal System (RHR)

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

3.9.1 Shutdown Cooling Mode (SDC)

The SDC mode is designed to remove heat from the reactor coolant system during a normal reactor shutdown. The operational objective is to reduce the bulk reactor temperature to 125°F in approximately 20 hours. The system is also designed to cool the reactor to 212°F using one RHR heat exchanger loop after the most limiting single failure. At the uprated power level the decay heat is increased proportionally, thus slightly increasing the time required to reach the shutdown temperature. The increased shutdown cooling times have no effect on plant safety and are within acceptable limits.

3.9.2 Suppression Pool Cooling Mode (SPC)

The SPC mode is designed to remove heat discharged into the suppression pool to maintain pool temperature below the 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 and 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 4.

Based on our review of the licensee's rationale and evaluation, the staff concurs with the licensee's assessment that plant operations at the proposed uprated power level will have an insignificant impact on the suppression pool cooling mode.

3.9.3 Containment Spray Cooling Mode (CSC)

The CSC mode is designed to spray water from the suppression pool via spray headers into the containment airspace, to reduce containment pressure and temperature during post-accident conditions. The licensee indicated that the power uprate would slightly increase the containment spray water temperature. This increase has a negligible effect on the calculated values of drywell pressure, drywell temperature, and suppression chamber pressure because these parameters reach peak values prior to actuation of the containment spray.

Based on our review of the licensee's rationale and evaluation, the staff concurs with the licensee's conclusion that plant operations at the proposed uprated power level will have an insignificant impact on the CSC mode.

3.9.4 Fuel Pool Cooling Assist Mode

In the event that the spent fuel pool (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 to the SFP. Heat loads on the RHR system SFP cooling assist mode will increase proportionally to the increase in reactor operating power level. The licensee performed evaluations and stated that the combined existing design heat removal capability of SFP cooling system and the RHR system in supplemental SFP cooling mode is sufficient to maintain the SFP temperature below 150°F during a normal (planned) refueling outage or an abnormal (unplanned) full core offload event resulting from the proposed power uprate. Since the design temperature of the SFP cooling system is 180°F, sufficient margin will be maintained with the additional heat loads of the power uprate. Therefore, the proposed power uprate has no impact on this mode of RHR system operations.

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

3.9.5 Steam Condensing Mode

This is not applicable because Perry has previously eliminated this mode of operation.

3.10 Reactor Water Cleanup (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 reactor coolant system. System temperature and pressure during operation are not changed at the uprated power level.

The licensee reviewed RWCU system functional capability. Based on 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 flow. The present reactor water conductivity limits are unchanged with the power uprate.

The system piping and component integrity were 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 staff finds the licensee's evaluation to be acceptable, and concludes that the RWCU system is capable of performing its function at the uprated power level.

3.11 Balance-of-Plant (BOP) Piping Evaluation

The licensee evaluated the stress levels for BOP piping and appropriate components, connections and supports in a manner similar to the evaluation of the RCPB piping and supports based on increases in 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 4, 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 existing design analyses of the affected BOP piping systems were reviewed against the uprated power conditions. The licensee concluded that in all cases there is a sufficient margin between the calculated stresses and the code-allowable limits to accommodate the increase in stresses due to the increase in pressure, temperature, and flow as a result of the power uprate. The staff finds that the stress ratios provided 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 for the supports to accommodate the load increase and as such, 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 staff finds the licensee's evaluation to be acceptable.

Based on the above review, the staff concludes that the design of piping, components and their supports will be adequate to maintain the structural and pressure boundary integrity of the BOP and reactor coolant piping, components and supports in the proposed power uprate.

4.0 ENGINEERED SAFETY FEATURES

4.1 Containment System Performance

The Perry 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 3579 MWt to 3758 MWt would change some of the conditions and assumptions of the containment analyses. Topical Report NEDC-31897 "Generic Guidelines For General Electric Boiling Water Reactor Power Uprate," Section 5.10.2 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 safety-relief valve containment dynamic loads. Appendix G of NEDC-31897 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 states that the applicant will analyze short-term containment pressure and temperature response using the GE M3CPT code (current analyses). 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) may be used to determine more realistic RPV break flow rates for input to the M3CPT code. The use of LAMB code has been reviewed and accepted by the NRC for application to LOCA analysis in accordance with 10 CFR 50, Appendix K. The results from these analyses will also be used for input to the LOCA dynamic loads evaluation.

Appendix G of NEDC-31897 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 containment suppression pool design temperature, ECCS NPSH and equipment qualification temperatures. These analyses can be performed using the GE computer code SHEX. SHEX 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.

4.1.1 Containment Pressure and Temperature Response

Short-term and long-term analyses of the containment pressure and temperature response following a large break inside the drywell are documented in the Perry 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 large break inside the drywell (DBA LOCA), while the long-term analysis was performed to determine the peak pool temperature response considering decay heat addition.

The licensee indicated that the containment analyses were performed in accordance with RG 1.49 and NEDC-31897 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 the LAMB model is justified in "General Electric Company Analytical Model for Loss-of Coolant Accident Analysis in Accordance with 10 CFR Part 50 Appendix K," NEDE-20566-P-A, September 1986. We find the use of the LAMB model detailed RPV break flow input to the M3CPT code in the containment analysis for power uprate acceptable. The licensee also indicated that the SHEX code was used to model the long-term containment pressure and temperature response. As indicated above, the SHEX computer code has been used by GE on all BWR power uprates and has been accepted by the staff for such applications.

The licensee reanalyzed the long-term peak containment pressure response for a main steamline break at 102 percent of the uprated power condition. A new peak calculated containment pressure was determined to be 6.40 psig (the previous value was 6.25 psig). The new value continues to be bounded by the containment design pressure of 15.0 psig.

Based on our review of the licensee's evaluation and the staff's review of power uprates for similar BWR plants, the staff finds the use of the SHEX computer code and the calculated peak containment pressure values acceptable for the Perry power uprate.

4.1.1.1 Long-Term Suppression Pool Temperature Response

(1) Bulk Pool Temperature

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 uprate power (3758 MWt) using the SHEX code and the more realistic decay heat model (ANS/ANSI 5.1+two sigma) than was used in the current USAR analysis. The staff has determined that the use of the ANS/ANS 5.1-1979 decay heat model with an uncertainty adder 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 the current methods and decay heat model to show the difference in containment pressure and temperature due to uprated power. These analyses calculated the peak suppression pool temperature of 180.3°F at uprated power level and 178.1°F at the current power level. The present USAR value for the above case was 182.7°F with the previous methods and decay heat model. The peak calculated suppression pool temperature of 180.3°F at uprate power remains below the suppression pool design temperature 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 staff concludes that the peak bulk suppression pool temperature response remains acceptable from both the NPSH and structural design standpoints for the power uprate.

(2) Local Pool Temperature with SRV Discharge

The local pool temperature limit for SRV discharge is specified in NUREG-0783, because of concerns resulting from unstable condensation observed at high pool temperatures in plants without quenchers. Elimination of this limit for plants with quenchers on the SRV discharge lines is justified in GE report NEDO-30832, "Elimination of Limit on Local Suppression Pool Temperature for SRV Discharge with Quenchers." In a safety evaluation report dated August 29, 1994, the staff eliminated the maximum local pool temperature limit for plants with quenchers on the SRV discharge lines, provided the ECCS suction strainers are below the quencher elevation. Perry has the ECCS suction strainers below the quenchers, so no evaluation of this limit is necessary. The licensee indicated that with a 5 percent power uprate, the power level increase is more than offset by the switch to the ANS 5.1+2 sigma decay heat model and, therefore, the current maximum local pool temperature for the NUREG-0783 analysis remains bounding.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff agrees with the licensee's conclusion that the plant operations at the uprated power will have an insignificant impact on the local pool temperature with SRV discharge.

4.1.1.2 Containment Air Temperature Response

The licensee indicated that the limiting DBA analyses for MSLB and RSLB were performed to calculate peak drywell and containment airspace temperatures. The results of the analyses show that power uprate did not produce significant changes in the peak drywell and containment gas temperatures. The analyses calculated the peak drywell temperature of 329°F at uprated power level. The peak calculated drywell temperature of 329°F at uprated power remains below the drywell design temperature of 330°F.

The analyses also calculated the peak containment wetwell temperature of 148.6°F at uprated power. The peak calculated containment wetwell temperature of 148.6°F at uprated power remains below the containment wetwell 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 our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff agrees with the licensee's conclusion that the power uprate will not adversely affect the containment drywell and wetwell air temperature response following a postulated LOCA.

4.1.1.3 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.

The revised analyses calculated a maximum drywell pressure of 23.45 psig which is essentially unchanged from the 23.43 psig calculated for the existing power level. The calculated maximum drywell pressure remains below the design value of 30 psig.

The revised analyses calculated a maximum containment pressure of 11.37 psig which is essentially unchanged from the 11.35 psig calculated for the existing power level. The calculated maximum containment pressure remains below the design value of 15 psig.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar BWR plants, the staff agrees with the licensee's conclusion that the power uprate will not adversely affect the containment and drywell pressure response following a postulated LOCA.

4.1.2 Containment Dynamic Loads

4.1.2.1 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, which 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, 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 pool swell and condensation oscillation loads for the plant. The long-term response conditions with power uprate, in which chugging would occur, are within the conditions used to define the chugging loads. Therefore, the LOCA dynamic loads for Perry are not affected by the power uprate.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar BWR plants, the staff agrees with the licensee's conclusion that the power uprate will not adversely affect the LOCA containment dynamic loads.

4.1.2.2 Safety Relief Valve Loads

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. The licensee indicated that for the first SRV actuations following an event involving RPV pressurization, the only parameter change which can affect the SRV loads is an increase in SRV opening setpoint pressure. The proposed power uprate does not include a SRV opening setpoint pressure increase. Therefore, the power uprate will not impact the SRV load definitions.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar BWR plants, the staff agrees with the licensee's conclusion that the plant operation at uprated power will not impact the SRV containment loads.

4.1.2.3 Subcompartment Pressurization

The licensee indicated that the energy release from the design basis pipe break will increase slightly (0.2 percent) due to operation at increased feedwater temperature and that there is no significant increase in the pressure-time histories that are used to calculate the asymmetric loads on the reactor vessel, attached piping, and biological shield wall. In addition, the biological shield wall and component design adequacy is not challenged because the design values of the pressure-time histories, which in turn were determined from the baseline mass and energy releases, bound the mass and energy releases at uprated power.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff agrees with the licensee's conclusion that plant operation at the proposed uprated power level will have an insignificant impact on the subcompartment pressurization.

4.1.3 Containment Isolation

The licensee indicated that the system designs for containment isolation are not affected by the power uprate. The capability of the actuation devices to perform at power uprate conditions has been evaluated and determined to be acceptable.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff agrees with the licensee's conclusion that plant operations at the proposed uprated power will have an insignificant impact on the containment isolation system.

4.1.4 GL 89-10 Program

The licensee indicated that the power uprate will not significantly increase any system operating pressure. There is no change in reactor pressure, or safety relief valve setpoints. The licensee confirmed that all safety-related valves will perform their intended function(s) following the uprated power conditions that affect the fluid flow, line pressures and temperature, valve differential pressures, and ambient temperature conditions. The licensee concluded that all motor-operated valves (MOVs) in the Perry MOV program will continue to comply with GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance."

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff concurs with the licensee's conclusion that their MOV program will continue to comply with GL 89-10 and will remain acceptable after the power uprate.

4.1.5 GL 96-06

The licensee reviewed the plant-specific information on Perry systems and components for the power uprate to determine its potential effect on the performance of mechanical components. The licensee concluded that there will be no mechanical components such as heat exchangers, pumps and valves for which operability could not be confirmed at the power uprate condition. The licensee also indicated that the proposed power uprate conditions are bounded by the current containment analysis and thus, has no impact on the evaluation in response to GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," on potential over-pressurization of isolated piping segments for Perry.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff concurs with the licensee's conclusion that their evaluation in response to GL 96-06 will remain acceptable after the power uprate.

4.2 Emergency Core Cooling Systems (ECCS)

The effect of power uprate on ECCS systems, including compliance with net positive suction head requirements, is addressed in this section. ECCS performance evaluation is discussed in Section 4.3.

4.2.1 High Pressure Core Spray (HPCS)

The HPCS system is automatically initiated in the event of a LOCA. When operating in conjunction with other ECCS, HPCS provides required core cooling for all LOCA events.

The maximum injection pressure for the HPCS system is based on the upper analytical setpoint for the lowest available group of SRVs. The SRV setpoints are unchanged for the power uprate. Operation of the HPCS system at power uprate conditions does not change, thus, there is no effect on the availability of the system, and the original design pressures or temperature for the system components are unchanged. The ECCS performance evaluation for power uprate (Section 4.3) shows that existing HPCS capability, with other ECCS, is sufficient to meet post-LOCA core cooling requirements for uprated power conditions. Power uprate does not decrease the NPSH available for the HPCS pump or increase the required NPSH. Surveillance testing of the HPCS system is not affected.

The HPCS system was evaluated by the licensee and found to be consistent with the bases and conclusions contained in Reference 6, the generic evaluation for power uprate.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff concurs with the licensee's conclusion that the power uprate will not adversely affect the HPCS system.

4.2.2 Low Pressure Core Injection System (LPCI) mode of RHR

The LPCI mode of operation of the RHR system is automatically initiated during a LOCA. Along with other ECCS, LPCI provides adequate core cooling for all LOCA events.

The higher decay heat because of power uprate could slightly increase the peak clad temperature following a postulated LOCA. The ECCS performance evaluation indicates that the existing LPCI performance capability, along with the other ECCS, is adequate to meet post-LOCA cooling requirements for the uprated power conditions. The hardware capability of the LPCI system to perform its function under power uprate conditions was evaluated by the licensee and found to be consistent with the generic evaluation in Reference 6.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff concurs with the licensee's conclusion that the power uprate will not adversely affect the LPCI system.

4.2.3 Low Pressure Core Spray (LPCS) System

The LPCS mode of operation of the RHR system is automatically initiated during a LOCA. Along with other ECCS, LPCS provides adequate core cooling for all LOCA events.

The higher decay heat because of power uprate could slightly increase the peak clad temperature following a postulated LOCA. The ECCS performance evaluation indicates that the existing LPCS performance capability, along with the other ECCS, is adequate to meet post-LOCA cooling requirements for the uprated power conditions. The capability of the LPCS system to perform its function under power uprate conditions was evaluated by the licensee and found to be consistent with generic evaluation in Reference 6.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff concurs with the licensee's conclusion that the power uprate will not adversely affect the LPCS system.

4.2.4 Automatic Depressurization System (ADS)

The ADS uses safety/relief valves to reduce reactor pressure following a small break LOCA. This function allows LPCI and LPCS to inject coolant to the vessel. The ADS initiation logic and ADS valve control are unchanged for uprate conditions. ECCS design requires a minimum flow capacity for the SRVs and also requires that ADS initiates on low water level plus high drywell pressure, or low water level alone. The licensee has concluded that the required flow capacity and ability to initiate ADS on appropriate signals are not affected by power uprate.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff concurs with the licensee's conclusion that the power uprate will not adversely affect the ADS.

4.2.5 Net Positive Suction Head (NPSH)

Power uprate increases the reactor decay heat, which increases the heat input to the suppression pool in the event of a LOCA. This increased heat load could increase the peak suppression pool water temperature and containment pressure during post-LOCA long term RHR, LPCS, and HPCS pump operation. The NPSH requirements for the ECCS pumps are conservatively based on 0 psig containment pressure and a peak post-LOCA suppression pool water temperature of 185°F. Because the peak post-accident suppression pool temperature does not exceed 185°F, NPSH requirements are maintained under power uprate conditions.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff concurs with the licensee's conclusion that the power uprate will not adversely affect the ECCS pump NPSH.

4.3 ECCS Performance Evaluation

The ECCS is designed to provide protection against hypothetical LOCAs caused by ruptures in 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 Appendix K. The results of the ECCS-LOCA analysis using NRC-approved SAFER/GESTR-LOCA methodology were presented by the licensee. Perry implemented SAFER/GESTR methodology for pre-uprate conditions during 1999, as reported to the NRC in Reference 11.

The SAFER/GESTR-LOCA analysis for pre-power uprate conditions was performed at conditions within 3 percent of the power level proposed by the uprate. The analysis addressed the same performance improvement programs as for power uprate (i.e., alternate operating modes such as maximum extended operating domain (MEOD), ICF, feedwater temperature reduction, and single-loop operation). The power uprate and the fuel reload for uprate will not change the limiting break, single failure assumption, or break spectrum as compared to the existing analysis. The performance improvement programs are also unchanged.

The SAFER/GESTR-LOCA analysis for power uprate conditions was conducted for the limiting case, the DBA recirculation suction line break with HPCS diesel generator failure. The licensing basis peak clad temperature (PCT) is determined based on the calculated Appendix K PCT at rated core flow with an adder to account for uncertainties. The value is then converted to a higher licensing basis PCT for MEOD.

For 105 percent power uprate, the licensing basis PCT based on the limiting fuel is 1340°F at rated core flow and 1370°F for MEOD. The comparable licensing basis PCT for the pre-uprate conditions is 1300°F at rated core flow and 1330°F for MEOD.

The estimated upper bound PCT for MEOD is less than 1250°F for the uprated power conditions. This is an increase of less than 40°F from the pre-uprate conditions and is significantly below the 1600°F limit required by the NRC safety evaluation for SAFER/GESTR.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff concurs with the licensee's conclusion that the power uprate will not adversely affect the ECCS performance evaluation.

4.4 Main Control Room Atmosphere Control System

The licensee indicated that this system is not significantly affected by power uprate. The engineered safety feature (ESF) filtration system for the Control Room Emergency Recirculation was found to have iodine loading levels well below the limit established in RG 1.52, and thus, the charcoal is not adversely affected by power uprate.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff agrees with the licensee's conclusion that plant operations at the proposed uprated power will have an insignificant impact on the Main Control Room Atmosphere Control System.

4.5 Annulus Exhaust Gas Treatment System (AEGTS)

The AEGTS is designed to process the air in the annular space between the shield building and the primary containment vessel to limit the release to the environment of radioisotopes which may leak from primary containment under accident conditions. The capacity of the AEGTS was selected to provide a negative differential pressure between the annulus and the outside of at least 0.25 inch of water. The licensee indicated that this capability is not affected by the power uprate. The post-LOCA total iodine loading at uprated power remains well below the limits of RG 1.52, "Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar BWR plants, the staff agrees with the licensee's conclusion that the plant operations at the proposed uprated power level will have an insignificant impact on the AEGTS.

4.6 Main Steam Isolation Valve Leakage Control System

This is not applicable because Perry has previously eliminated this system.

4.7 Post-LOCA Combustible Gas Control System

The combustible gas control system is designed to maintain the hydrogen concentration of the drywell and containment atmospheres below the lower flammability limit of 4 volume percent following LOCA. The post-LOCA production of hydrogen by radiolysis will increase in proportion to power. The licensee indicated that the increase in hydrogen generation due to 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. The time required for operator initiation of the mixing compressors prior to reaching 3.0 volume percent hydrogen is reduced from 18.4 hours for current rated power to 17 hours for uprated power. After initiation of mixing, the time required to reach 3.5 volume percent hydrogen in the drywell is reduced from 287 hours to 255 hours. The above timing change does not affect the ability of the operator to take action. Power uprate has no impact on recombiner maximum operating temperature which is dependent only on the containment hydrogen concentration when the recombiners are started.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar BWR plants, the staff agrees with the licensee's conclusion that plant operations at the proposed uprated power level will have an insignificant impact on the post-LOCA combustible gas control system.

5.0 INSTRUMENTATION AND CONTROL

In NEDC-32907P (Reference 4), GE stated that most BWR plants, as originally licensed, have an assigned equipment and system capability to accommodate steam flow rates at least five percent above the original rating. In addition, improvements in analytical techniques, plant performance feedback, fuel and core designs have resulted in 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 to 20 percent without major NSSS hardware modifications. GE further stated that the NEDC-32907P analysis are based on the guidelines and evaluations provided in the NRC approved GE Topical Report NEDC-31897P-A, Class III, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate" dated May 1992 and/or NEDC-31984P-A, "Generic Evaluations of General Electric Boiling Water Reactor Power Uprate," dated July 1991 for Class III (Proprietary) and March 1992 for Class I (Non-proprietary), and Supplements 1 & 2.

In addition, GE used the NRC approved GE Topical Report NEDE-3133P-A, "General Electric Instrument Setpoint Methodology" dated September 1996 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 NEDE-32907P that prior to operation at the uprated power level, the Perry plant will be subjected to power uprate testing to Section 5.11.9 and Appendix L, Section L.2 of NEDC-31897P-A. 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 ≤ 3 percent power. These tests will be specifically conducted for IRM Neutron Monitors, Average Power Range Monitors, Pressure Regulator System, Feedwater Control System, Recirculation Flow Control, Recirculation Flow, and Radiation Measurements.

The staff has compared the proposed changes against the guidelines provided in Appendix F of NEDC-3189P-1 and finds them acceptable. Specifically, most of the changes involve 5 percent reduction in the allowable values in terms of the uprate power which means that in terms of absolute power the setting remains unchanged. As pointed out in NEDC-3189P-1, these changes will avoid any hardware modifications. The NRC has approved similar TS changes for other nuclear plants seeking power uprates.

The proposed TS changes are in conformance with the NRC approved GE reports NEDE-31336P-A, NEDC-31897P-A, and NEDC-31897P-A. Most of the changes will leave the allowable values unchanged in terms of absolute power. The proposed setpoint changes will provide adequate margins between analytical limits and setpoints and will not significantly increase the likelihood of a false trip or failure to trip on demand.

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

6.0 ELECTRICAL POWER AND AUXILIARY SYSTEMS

6.1 AC POWER

6.1.1 Offsite Power

The staff has reviewed information provided by the licensee to determine the impact of the power uprate on offsite power. Areas included in the review were grid stability analysis and related electrical systems.

6.1.1.1 Grid Stability and Reliability Analysis

The licensee performed a grid stability uprate review to determine the adequacy of grid stability for the Perry power uprate. The grid stability studies, which considered the increase in electrical output, demonstrated conformance to 10 CFR 50, Appendix A, General Design Criterion (GDC) 17. GDC 17 addresses onsite and offsite electrical supply requirements. A sensitivity analysis was performed at higher levels than the uprate power level in order to demonstrate adequate stability margin. No grid stability or reliability issues were identified by the stability analysis.

The staff requested that the licensee discuss what this grid stability uprate review consisted of and include in this description the major assumptions for this review and the resulting primary review findings and conclusions. In response to the staff request, the licensee stated that the impact of the proposed power was assessed through an analysis of a variety of probable and severe scenarios, reflecting the requirements contained in the North America Electric Reliability Council Planning Standards. The power flow analysis considered thermal loading of transmission line and transformer branches and bus voltage violations under normal and contingency operating conditions. With the 5 percent uprate, no additional branch loading or bus voltage violations were observed, and no violations were intensified by the uprate conditions. Responses for the stability analysis, which evaluated both first swing stability and system damping, found that all contingencies were stable and damped for the power uprate case model. The existing protective relay settings at Perry are based on the full generator output of 1446 MVA. Given that the 5 percent uprate does not exceed 1446 MVA, no relay setting changes are required and the probability of losing electric power to the unit is unchanged.

On the basis of this information, the staff concludes that the proposed power uprate at Perry will not adversely affect the grid stability and reliability.

6.1.1.2 Related Electrical Systems

The licensee performed a power uprate review to determine the adequacy of electrical systems associated with the main turbine-generator auxiliary systems. The review determined that the electrical system's configuration and operating voltage ranges are unchanged and remain adequate for operation at the higher output. The review determined that the isophase bus rating, the main power transformer ratings, the unit auxiliary power transformer ratings, the system auxiliary power transformer ratings, the 345-kV switchyard equipment ratings and operating voltage ranges, the generator voltage and current ratings and operating voltage ranges bound the uprate operating conditions.

The staff requested that the licensee discuss the technical basis for the increase in the main transformers rating from 1394.4 MVA to 1580 MVA. The licensee stated that the uprated Perry main transformer rating of 1580 MVA was calculated using the Electric Power Research Institute (EPRI) transformer loadability program, "PTLoad - A Numerical Model for Power Transformer Load Planning." The Perry main transformer thermal characteristics were modeled using the methodology described in Institute of Electrical and Electronic Engineers (IEEE) C57.91-1995, "Guide for Loading Mineral-Oil-Immersed Transformers," and were based on test report (heat run) data supplied by the manufacturer and actual operating voltages. Historical ambient temperature data for the licensee's service territory were used to model the environmental conditions. The resulting PTLoad calculations determined that the main transformer rating could be revised to 1580 MVA in order to utilize the full thermal capabilities of the transformers under normal operating conditions while remaining within the hot spot and top oil temperature limits.

Therefore, the licensee has concluded that the turbine/generator and major electrical components extending from the isophase bus to the switchyard will remain adequate for operation at the higher output.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff concurs with the licensee's conclusion that the power uprate will not adversely affect the electrical systems associated with the main turbine-generator auxiliary systems.

6.1.2 Onsite Power

The onsite power distribution system consists of transformers, buses, switchgear, and distribution panels. The alternate current (ac) power to the distribution system is provided from the transmission system or the onsite emergency diesel generators. Station batteries provide direct current (dc) power to the dc distribution system. The licensee noted that operation at the uprated level is achieved by utilizing existing equipment operating at or below the nameplate ratings. Station loads under emergency operating conditions (powered via the emergency diesel generators) are evaluated based on equipment nameplate data, except for the HPCS pump which is based on operating data.

The staff requested that the licensee address the impact of the power uprate on the load, voltage, and short circuit current values for all levels of the plant auxiliary electrical distribution system (ac and dc). In response to the staff's request, the licensee stated that the expected switchyard, generator and battery voltage conditions along with the ac and dc electrical distribution configuration and characteristics are unchanged as a result of the power uprate change. The licensee stated that the only change in load demand which is due to the power uprate is associated with the hotwell, condensate booster and feedwater booster pumps. Although these pumps experience increased demand due to the increased flow under uprate power conditions, the motor demand for each of these loads remains bounded by the existing design basis calculations. The licensee stated that no electrical distribution calculation changes are required and the voltage and short circuit studies are unaffected by the power uprate change. Therefore, the existing diesel generator load calculations are unchanged by the uprated conditions, and the current emergency power system design remains adequate. The system has sufficient capacity to support the required loads for safety shutdown, to maintain a safe-shutdown condition, and to operate the required engineered safeguards equipment following a postulated accident. Therefore, the licensee has concluded that power uprate has no impact on the emergency onsite power system.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff concurs with the licensee's conclusion that the power uprate will not adversely affect the onsite power distribution system.

6.2 DC Power

The staff has reviewed information provided by the licensee to determine the impact of the power uprate on the dc power system. The dc power distribution system provides control and motive power for various systems and components within the plant. The licensee noted that operation at the uprated level is achieved by utilizing existing equipment operating at or below the nameplate ratings. The licensee stated that no electrical distribution calculation changes are required and the voltage and short circuit studies are unaffected by the power uprate change.

On the basis of this information, the staff concludes that the proposed power uprate at Perry has no impact on the dc power system.

6.3 Spent Fuel Pool (SFP)

6.3.1 Fuel Pool Cooling System

The SFP heat load at uprated power level was evaluated and determined to be higher as compared to the pre-power uprate heat load. The SFP heat load was determined by calculating the heat load generated by a full core off-load plus remaining spaces filled with spent fuel discharged at regular intervals, and calculating bulk pool temperatures.

The SFP cooling system, which consists of two cooling trains, is designed to maintain the SFP water temperature at or below 130°F using both trains[1] during planned refueling outages. As discussed in Section 3.9.4, supplemental fuel pool cooling is provided by the RHR[2] system to maintain the SFP water temperature at or below 150°F during a planned refueling outage or an unplanned full core offload event.

The licensee has stated that power uprate does not affect the heat removal capability of the SFP cooling system. Power uprate results in slightly higher core decay heat loads during refueling. The power uprate analysis assumes 24 month fuel cycle lengths. Each reload will affect the decay heat generation in the SFP after discharge of fuel from the reactor. The licensee's evaluation considered the expected heat load in the spent fuel storage pool at the uprated conditions, and confirmed the capability of the SFP cooling system to maintain adequate cooling.

The licensee performed evaluations which show that the maximum heat load in the SFP for power uprate increases, but is still below the pre-uprate design basis heat load output. The

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1 If one pump and one heat exchanger were lost, the SFP water temperature would rise to 160°F maximum for a short time. Under this condition, the RHR would be used to maintain the SFP temperature below 150°F until the normal SFP cooling system could maintain the temperature below 150°F.

2 As stated in the Perry Final Safety Analysis Report, Perry Technical Specifications will not allow reactor startup whenever the RHR system is being used for SFP cooling.

combination of the SFP cooling system heat exchangers and the availability of the RHR system is sufficient to remove the decay heat such that the SFP water temperature remains at or below 150°F during a planned refueling outage or an unplanned full core offload event. Since this remains well below the SFP design temperature of 180°F, sufficient margin will be ensured.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar BWR plants, the staff concludes that plant operations at the proposed uprate power level will not adversely affect the capability of the SFP cooling system and the RHR system in the fuel pool cooling assist mode.

6.4 Water Systems

Water systems are designed to provide cooling water to various systems (both emergency and non-emergency service water systems). All heat removed by these systems is rejected to the ultimate heat sink. The licensee stated that the environmental effects of uprate will be controlled at the same level as is presently in place. That is, the plant operation will be managed such that none of the present limits such as maximum allowed ultimate heat sink temperature will be increased as a result of power uprate. The staff finds this to be acceptable.

6.4.1 Emergency Water Systems

The emergency water systems, which consist of the emergency service water system and the emergency closed cooling system, are designed to provide reliable supplies of cooling water to various safety-related equipment during and following a design basis accident.

6.4.1.1 Emergency Service Water Systems (ESWSs)

The ESWSs remove heat from emergency closed cooling heat exchangers, RHR heat exchangers, spent fuel pool cooling heat exchangers, diesel generator coolers, SFP emergency makeup, and HPCS diesel generator coolers and pump room coolers. The licensee performed evaluations and stated that plant operations at the proposed uprated power level will have an insignificant impact on the ESWSs and that power update does not require a modification of the ESWSs.

Based on our review of the licensee's evaluation and the staff's review of power update applications for similar BWR plants, the staff agrees with the licensee that plant operations at the proposed uprated power level will have an insignificant impact on the ESWSs.

6.4.1.2 Emergency Closed Cooling System

The emergency closed cooling system removes heat from control complex chillers, hydrogen analyzers, emergency core cooling system room coolers and RHR pump seal coolers following a LOCA. The licensee performed an evaluation and concluded that the cooling loads for the emergency closed cooling system remain virtually the same as that for the current rated power level operation because the heat loads from the above equipment remain essentially unchanged for LOCA conditions following plant operations at the proposed uprated power level.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar BWR plants, the staff agrees with the licensee's conclusion that plant operations at the proposed uprated power level will have an insignificant impact on the emergency closed cooling system.

6.4.2 Non-Emergency Water Systems

6.4.2.1 Service Water System

The licensee stated that the major service water heat load increases from power uprate reflect an increase in main generator losses rejected to the stator water coolers, generator hydrogen coolers, and exciter coolers. The increase in service water heat loads from these sources due to uprated operation is approximately proportional to the power update. The licensee performed evaluations which demonstrate that the service water system is adequate for power update conditions.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff concurs with the licensee's conclusion that the power uprate will not adversely affect the service water system.

6.4.2.2 Nuclear Closed Cooling Water (NCCW) System

The NCCW system is designed to cool various auxiliary equipment during normal plant operations. The licensee indicated that the increase in heat load due to uprated power level operation has an insignificant impact on the NCCW system and that the NCCW system is adequate for power uprate conditions.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff concurs with the licensee's conclusion that power uprate will not adversely affect the NCCW system.

6.4.2.3 Turbine Building Closed Cooling (TBCC) System

The TBCC system supplies cooling water to auxiliary plant equipment in the turbine building. The licensee indicated that heat loads related to the turbine-generator are power dependent and will increase the overall heat loads on the TBCC system. If the TBCC water temperature increases to an unacceptable level as a result of increased Lake Erie temperature and/or plugged condenser tubes, the licensee will need to take actions to either increase service water flow to the TBCC heat exchanger or reduce reactor thermal power. The licensee has concluded that these actions will be sufficient to maintain operability of the TBCC system.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff concurs with the licensee's conclusion that power uprate will not adversely affect the TBCC system.

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

The main condenser, auxiliary 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. The licensee stated that the performance of the main condenser, auxiliary condenser, circulating water, and normal heat sink systems was evaluated and found adequate for plant operations at the proposed uprated power level.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff concurs with the licensee's conclusion that power uprate will not adversely affect the condensers, circulating water system, and cooling tower systems.

6.4.3.1 Discharge Limits

The licensee evaluated the impact of power uprate on the effluent discharge from the cooling tower to the atmosphere. The effluent discharge results in cooling tower evaporation and cooling tower drift from the circulating water system.

The effect on cooling tower evaporation, makeup, and blowdown was evaluated and found to be acceptable. An increase in steam and condensate flow will result in a corresponding increase in the net heat rejection to the cooling tower. The cooling tower evaporation is calculated to increase from 14,554 gallons per minute (gpm) to 15,587 gpm, whereas the cooling tower drift and blowdown temperature are predicted to remain unchanged. In NUREG-0884 (Final Environmental Statement Related to the Operation of Perry Nuclear Power Plant, Units 1 and 2), the staff concluded that cooling tower induced icing and fogging with two cooling towers in operation would not adversely affect driving conditions, airports, shipping ports, or waterways in the vicinity of the plant. Considering that only one unit was completed at the Perry site, any increase in icing and fogging from the additional cooling tower evaporation would be bounded by the original two-unit analyses. There are no state regulated limits for cooling tower parameters.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar plants, the staff concurs with the licensee's conclusion that power uprate will not adversely affect the effluent discharge from the cooling tower.

6.4.4 Ultimate Heat Sink (UHS)

The ultimate heat sink for the Perry Plant is Lake Erie. Power uprate will not impact the inlet temperatures of the UHS. The UHS inlet temperature limit will remain at 81.5°F.

The service water system at Perry was originally designed to support the operation of two units. Therefore, the design discharge temperature into Lake Erie is based on two unit operation. As a result of power uprate to 105 percent of current licensed core power, there will be a slight increase in the normal heat loads rejected to the plant service water system. For normal operation, the maximum service water heat loads occur during peak summer months. The licensee calculates that the maximum summer discharge temperature for the service water system will be increased by 0.34°F, or from 90.1°F to 90.44°F. This increase in service water temperature will not exceed the original design discharge temperature.

Based on the review of the licensee's evaluation, the staff agrees with the licensee's conclusion that plant operations at the proposed uprated power level will have an insignificant impact on the UHS.

6.5 Standby Liquid Control System (SLCS)

The SLCS is designed to shut down the reactor from rated power to cold shutdown assuming that all or some of the control rods cannot be fully inserted. It is a manually operated system that will pump a solution of borax and boric acid into the vessel to provide neutron absorption and achieve a subcritical reactor condition.

The ability of the SLCS boron to achieve and maintain safe shutdown is not a direct function of core thermal power, and therefore, is not affected by power uprate. SLCS shutdown capability (in terms of required boron concentration) is reevaluated for each core reload. The SLCS is designed for injection at a maximum reactor pressure equal to the upper analytical setpoint for the lowest group of SRVs operating in relief mode. For power uprate, the reactor operating dome pressure and the SRV relief setpoints remain unchanged. Consequently, SLCS process parameters do not change. The capability of the SLCS to provide its backup shutdown and anticipated transient without scram (ATWS) functions is not affected by power uprate.

6.6 Power Dependent Reactor Building and Plant Heating, Ventilation, and Air Conditioning (HVAC) Systems

The licensee indicated that heating ventilation and air conditioning systems consist mainly of heating, cooling supply, exhaust and recirculation units in the turbine building, reactor building and the 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. The areas which are affected by power uprate are the feedwater heater bay and condenser areas in the turbine building. Heat loads in the drywell are not significantly affected because the reactor coolant temperature and recirculation drive flow are not significantly changed. Other areas are unaffected by power uprate because the process temperatures remain relatively constant. The heat loads represent an increase of less than 2 percent in the main steam tunnel and heater bay area total heat loads, and less than 0.1 percent in the drywell. Based on a review of the design basis calculations, these increases are within the design capability of the HVAC system. Therefore, the design of the HVAC is not adversely affected by power uprate.

Based on our review of licensee's rationale and the staff's review of power uprate applications for similar BWR plants, the staff agrees with the licensee's conclusion that plant operations at the proposed uprated power level will have an insignificant impact on the HVAC systems.

6.7 Fire Protection

Fire suppression or detection is not expected to be impacted due to 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 shutdown systems and equipment used to achieve and maintain cold shutdown 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 shutdown function.

Based on our review of the licensee's rationale and the staff's review of power update applications for similar BWR plants, the staff agrees with the licensee that the post-fire safe shutdown capability will not be affected by plant operations at the proposed uprated power level.

7.0 POWER CONVERSION SYSTEMS

7.1 Turbine-Generator

The 5 percent thermal power uprate increases the steam flow by approximately 6 percent. An engineering evaluation by the licensee indicates that a modification to the high pressure turbine nozzles may be required to ensure the turbine-generator is capable of achieving the proposed steam flow rate. If the licensee implements the power uprate during the existing Cycle 8 operation, power may be limited to approximately 104 percent. This is because the high pressure turbine cannot be modified during power operation. If modifications are needed for the high pressure turbine, these would be made during the next refueling outage.

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 nor associated turbine missile production due to plant operations at the proposed uprated power level.

Based on our review of the licensee's rationale and the staff's review of power update applications for similar BWR plants, the staff agrees with the licensee that the turbine can continue to be operated safely at the proposed uprated power levels.

7.2 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 concluded that the existing equipment for these systems are acceptable for plant operations at the proposed uprated power level.

Based on our review of the licensee's rationale and the staff's review of power update applications for similar BWR plants, the staff agrees with the licensee that the power uprate will not adversely affect the power conversion systems and their associated components.

8.0 RADWASTE SYSTEMS AND RADIATION SOURCES

8.1 Liquid Waste Management

The single largest source of liquid waste is from the backwash of the condensate demineralizers. The licensee stated that with the 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. Reactor coolant cleanup flows, leaks, laboratory drains, dry solid waste, and spent resin quantities will remain essentially the same after power uprate. 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.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar BWR plants, the staff agrees with the licensee's conclusion that the power uprate will not adversely affect the liquid radwaste system.

8.2 Gaseous Waste Management

Gaseous wastes generated during normal and abnormal operation are collected, controlled, processed, stored, and disposed 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 standby gas treatment system, as well as other building ventilation systems. Various devices and processes, such as radiation monitors, filters, isolation dampers, and fans, are used to control airbome radioactive gases. Results of licensee 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 our review of the licensee's evaluation and the staff's review of power uprate applications for similar BWR plants, the staff concludes that plant operations at the proposed uprated power level will have an insignificant impact on the above systems.

8.2.1 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 H_2 flow rate due to power update 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 our review of the licensee's evaluation and the staff's review of power uprate applications for similar BWR plants, the staff agrees with the licensee's conclusion that plant operations at the proposed uprated power level will have an insignificant impact on the offgas system.

9.0 REACTOR SAFETY PERFORMANCE EVALUATIONS

9.1 Reactor Transients

The Perry USAR describes the results of analyses of plant transients caused by a malfunction or single failure of equipment or operator error. Transients are investigated according to the type of initiating event, consistent with Regulatory Guide 1.70. The generic guidelines for BWR power uprate (Appendix E of Reference 5) identify the limiting events to be considered in each category. The generic guidelines also identify the analytical methods to be used, the operating conditions to be assumed, and the criteria to be applied.

The licensee provided a summary of the safety evaluation results for each of the limiting events. The analyses were conducted for a representative core based on Reload 7/Cycle 8 and used the GEMINI transient analysis methods listed in Reference 5.

The licensee analyzed the following events for the power uprate:

- Turbine Trip without Bypass
- Load Reject without Bypass
- Pressure Regulator Downscale Failure
- Feedwater Controller Failure

- Loss of 100°F Feedwater Heating
- Rod Withdrawal Error
- Slow Recirculation Flow Increase

Most of the transient events were analyzed at the full uprate power and maximum allowed core flow operating point on the power/flow map. Direct or statistical allowance for 2 percent power uncertainty is included in the analysis. Nominal dome operating pressure is used, as defined in the ODYN and GEMINI methodology. The safety limit minimum critical power ratio (MCPR) was used to calculate the MCPR operating limits for the analyzed events. The two lowest setpoint safety/relief valves are assumed to be out-of-service in the transient analysis. For the MSIV closure overpressurization analysis, six safety valves are assumed out-of-service.

The safety limit MCPR for two-loop operation was analyzed and is not affected by power uprate. The single loop safety limit MCPR value of 1.11 is also not affected by power uprate. In single loop conditions, the MCPR operating limit is increased by the difference between the single loop and two-loop safety limit MCPR.

The limiting transients for each category were analyzed to determine their sensitivity to core flow, feedwater temperature, and cycle exposure. Core flow was varied from 75 percent to 105 percent flow. Feedwater temperature of 420°F was used for all the transients except for feedwater controller failure, which used 250°F. The increased core flow cases were more limiting than the low flow cases. Only end-of-cycle exposure cases were valuated for pressurization transients. The loss of feedwater heating transient was evaluated at beginning-of-cycle, middle-of-cycle, and end-of-cycle. The rod withdrawal error event was evaluated at middle-of-cycle. The results from these analyses developed the licensing basis for transient analyses at the uprated power. No change to the basic characteristics of any of the limiting events is caused by power uprate.

The severity of transients at less than rated power including the slow recirculation flow increase transient were reviewed and found to be insignificantly affected by the power uprate. Adequate protection is provided by the current flow-dependent thermal limits (e.g., MCPR(f)). Adjustment is required to the power-dependent thermal limits (e.g., MCPR(p)) because the rod control and information system high power setpoint and direct scram bypass power level remains at the same absolute power level under the proposed uprate. Therefore, the limits in terms of percent of uprated power are lower under the power uprate.

The licensee did not explicitly analyze the loss of feedwater flow (LOFW) transient, since it is generically evaluated in Reference 6. During a LOFW transient assuming a single failure (loss of RCIC or HPCS) reactor water level is automatically maintained above the top of active fuel by RCIC or HPCS without any operator action required. Because of the additional decay heat under uprated power conditions, slightly more time will be needed for the automatic systems to restore water level. Operator action is only needed for long-term plant shutdown once water level is restored. These sequences do not require any new operator actions or shorter operator response times. Therefore, operator actions for a LOFW transient do not significantly change for power uprate.

Based on our review of the licensee's rationale and the staff's review of power update applications for similar BWR plants, the staff agrees with the licensee that power uprate does not result in changes which significantly affect the previous evaluations or conclusions for the reactor transients.

9.2 Special Events

9.2.1 Anticipated Transients Without Scram (ATWS)

Perry meets the ATWS mitigation requirements defined in 10 CFR 50.62:

- Installation of an Alternate Rod Insertion (ARI) system,
- Boron injection equivalent to 86 gpm,
- Installation of automatic recirculation pump trip logic.

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

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

Key inputs to the ATWS analysis are:

- Reactor power of 3758 MWt,
- Reactor dome pressure of 1040 psia,
- SRV opening setpoints at current values,
- ATWS high pressure setpoint at current value,
- 2 SRVs out-of-service.

ATWS analyses were performed for the events in Reference 6 and the results presented show that Perry meets the ATWS acceptance criteria for the uprated power conditions. Therefore, the ATWS analysis is acceptable for Perry.

9.2.2 Station Blackout

The staff has reviewed information provided by the licensee to determine the impact of the power uprate on the existing analysis performed for station blackout (SBO). The licensee reevaluated its SBO analysis using the guidelines of Nuclear Management and Resources Council (NUMARC)-8700 (Reference 12). The licensee evaluated the changes for the existing SBO analysis under the power uprate conditions, particularly as they relate to issues such as heat-up analysis, equipment operability, and battery capacity.

The staff requested that the licensee (1) provide the numerical estimate for the increase in decay heat and associated temperature rise in the plant areas relevant in coping with station blackout conditions; (2) discuss the potential impact if additional SRV actuations occur due to the increased decay heat; and (3) discuss and verify that the results of suppression pool temperature transient analyses show that ECCS equipment will not be adversely impacted given a maximum allowable cooldown rate during the reactor pressure vessel depressurization.

In response to the staff's request, the licensee stated the numerical estimate for the increase in decay heat following a SBO is roughly consistent with the degree of the uprate (i.e., approximately 5 percent) without any adjustment for additional uncertainty. The temperature response for areas other than the suppression pool, such as the battery area and HPCS room is not expected to change due to the SBO event under power uprate conditions. The licensee's analysis confirmed that the suppression pool temperature will remain below the 185°F temperature limit. The increased decay heat will result in a slightly larger number of SRV cycles prior to depressurization. However, since the pneumatic supply is sufficient and the low-low set logic is active, the number of SRV cycles is anticipated to be lower than the design basis requirement. The licensee verified for the HPCS diesel and pump, which are the only ECCS equipment used during the SBO event, that operation is not impacted by the depressurization and that the system is designed to operate at the anticipated low reactor pressures.

Based on our review of the licensee's rationale and the staff's review of power update applications for similar BWR plants, the staff agrees with the licensee that power uprate does not result in changes which significantly affect the previous evaluation or conclusions for the SBO event.

10.0 ADDITIONAL ASPECTS OF POWER UPRATE

10.1 High Energy Line Breaks (HELBs)

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, reactor core isolation cooling system, etc.) evaluated in the Perry 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 our review of the licensee's evaluation and the staff's review of power uprate applications for similar BWR plants, the staff concludes that the existing analysis for HELB remains bounding and is acceptable for plant operations at the proposed uprated power level.

10.2 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 because the piping system operating temperatures, pressures and flows, along with the environmental zone conditions, remain essentially unchanged.

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

10.3 Equipment Qualifications (EQ)

10.3.1 Electrical Equipment

The licensee evaluated the safety-related electrical equipment to ensure qualification for the normal and accident conditions expected in the area in which the devices are located.

10.3.1.1 Inside Containment

EQ for safety-related electrical equipment located inside the primary containment is based on a steam line break and/or design basis LOCA conditions. The resultant temperature, pressure, humidity, and radiation consequences bound the environment expected to exist during normal plant operation. The licensee evaluated the EQ for safety-related electrical equipment located inside the primary containment and determined that the current accident and normal plant conditions for temperature, pressure, and humidity inside containment are "effectively unchanged" for the power uprate conditions.

The staff requested that the licensee provide a discussion to clearly explain how the current accident and normal temperature, pressure, and humidity profiles for inside the primary containment change for the power uprate conditions and why these changes have no impact on the EQ of electrical equipment. In response to the staff's request, the licensee provided a detailed discussion of the changes to the normal and accident temperature, pressure, and humidity profiles under power uprate conditions. Normal service temperatures are expected to increase little, if any, and there are no changes expected in the normal operating pressure as a result of power uprate conditions. Since the normal operating temperature in the drywell is expected to increase little under power uprate conditions, and leakage into the drywell is not affected, it is concluded that drywell humidity will remain unaffected. Following an accident, relative humidity increases to 100 percent for the pre-uprate condition. Since this is the maximum value for relative humidity, there is no change in the power uprate case. For accident conditions, in the power uprate case, the calculated short term peak pressure used for qualification of equipment inside the drywell is greater than the pre-uprate calculations and there are only minor changes to the shape of the pre-uprate temperature and pressure profiles.

The staff noted that the reevaluation of the EQ conditions under the uprated power conditions identified some electrical equipment located inside the primary containment and mechanical equipment with non-metallic components which are affected by the higher accident radiation level. The staff requested that the licensee (1) identify the subject equipment and discuss how this equipment will be requalified for the new radiation values; (2) provide the current, the revised, and bounding radiation level values and; (3) provide numerical values for specific equipment exposure under these new radiation conditions.

In response to the staff's request, the licensee initially stated that there are five Auditable File Packages (AFPs) pursuant to 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," where equipment appeared to not have the required 10 percent accident margin (radiation) as a result of the power uprate conditions. The AFPs identified were as follows: Power Range Detectors; Intermediate Range Detectors; Residual Heat Removal system pump motors; the RCIC Turbine Assembly; and the Fuel Handling Building Ventilation System Exhaust Filter Plenums. However, after a detailed review of the subject AFPs, the licensee confirmed that at least a 10 percent accident margin exists for the affected equipment.

In summary, the staff concludes that the power uprate has a negligible affect on normal plant operating environmental conditions and has no significant effect on the environmental conditions currently used for the safety-related electrical equipment EQ program inside the primary containment.

10.3.1.2 Outside Containment

Accident temperature, pressure, and humidity environments used for qualification of equipment outside primary containment result from a main steamline break in the pipe tunnel, or other HELBs. The accident temperature, pressure, and humidity conditions resulting from a LOCA do not change with power level, but some of the HELB profiles do increase by a small amount.

The staff requested that the licensee provide a discussion similar to the item raised in Section 10.3.1.1 for the temperature, pressure, and humidity profiles for HELB areas outside of primary containment. In response to the staff's request, the licensee stated that the only HELB transient outside containment affected by power uprate conditions is the rupture of a Reactor Water Cleanup line. Subsequent analysis of the HELB transient indicated that the calculated temperature and pressure would not vary significantly (i.e., less than 0.1 psi and less than 1 °F) under the power uprate blowdown conditions. Therefore, the previous analyses remains valid for power uprate conditions. Given that the previous analyses remains valid there is no anticipated impact on the environmental qualification of electrical equipment due to power uprate conditions.

In summary, the power uprate has a negligible effect on the environmental conditions currently used for the safety-related electrical equipment EQ program outside the primary containment. Therefore, the staff finds the licensee's evaluation to be acceptble.

10.3.2 Mechanical Equipment With Non-Metallic Components

The licensee performed an evaluation of the effects of plant operations at the proposed uprated power level on the non-metallic components of safety-related mechanical equipment. The licensee stated that certain systems would be affected by the slight increases in temperatures, pressures, and in some cases, flows due to plant operations at the proposed uprated power level. However, these increases in temperatures, pressures and flows are within the original EQ allowances.

Based on our review of the licensee's evaluation and the staff's review of power uprate applications for similar BWR plants, the staff concurs with the licensee that the existing EQ of mechanical equipment with non-metallic components remain bounding and is acceptable for plant operations at the proposed uprated power level.

10.3.3 Mechanical Component Design Qualification

The dynamic loads such as SRV discharge and LOCA loads (including pool swell, condensation oscillation, and chugging loads) that were used in the equipment design will remain unchanged as discussed in Reference 4. This is because the plant-specific hydrodynamic loads that are based on the range of test conditions for the design-basis analysis at Perry, are bounding for the power uprate condition.

Based on our review of the proposed power uprate amendment, the staff finds that the original seismic and dynamic qualification of the safety-related mechanical and electrical equipment is not affected by the power uprate conditions for the following reasons:

(a) Seismic loads are unchanged for the power uprate;

(b) No new pipe break locations or pipe whip and jet impingement targets are postulated as a result of the uprated conditions;

(c) Pipe whip and jet impingement loads do not increase for the power uprate; and

(d) SRV and LOCA dynamic loads used in the original design basis analyses are bounding for the power uprate.

10.4 Required Testing

TEST CONTROL

1. Regulatory Basis

Regulatory provisions for the testing of structures, systems, and components are identified under Criterion XI, "Test Control," of Appendix B to 10 CFR 50. The program for implementing these requirements is described in Section 6 of the Perry Operations Manual, PAP-1121, "Conduct of Infrequently Performed Tests or Evolutions," Revision 1, 1997. The PAP-1121 description follows the guidance of ANSI N45.2-1977, with respect to the development of test procedures, conduct of testing, and documentation and evaluation of test results.

Additionally, the startup test results are documented in a suitable test package, including deviations and adverse conditions, and actions taken to resolve condition in accordance with the licensee's Operational Requirements Manual, PDB-R0001, Section 7.6, "Reporting Requirements," Revision 0. Documentation of test results are maintained as lifelong records, in accordance with the guidelines provided in RG 1.28, Revision 2, 1979, to which the licensee has committed.

2. Generic Test Guidelines for GE BWR Power Uprate

NEDC-31984 (Reference 6), Section 5.11.9, provides the general guidelines for power uprate testing.

> 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 GE proprietary report NEDC-31897P-A (Reference 5). GE report NEDC-32907P (Reference 4), submitted with the licensee's application, provides the required additional information relative to power uprate testing.

3. Startup Test Plan

The license 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 5 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 13.5 of the Perry 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 and 100 percent of the currently licensed power level. Following testing at 100 percent currently rated power, the licensee will implement a series of five (5) 1 percent incremental tests through 105 percent.[3] During each incremental test, data will be taken and evaluated for acceptance to Level 1 and/or Level 2 acceptance criteria prior to escalation to the next power level. Level 1 criteria are associated with TS requirements and Level 2 Criteria are associated with design or desired parameters. 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 PAP-1211 which implements the licensee's QA program test control requirements.

The following power increase test plan is provided in NEDC-32907P, Section 10.4 "Testing." This testing philosophy will accommodate power uprate during Cycle 8.

- Surveillance testing will be performed on the instrumentation that requires recalibration for power update in addition to the testing performed according to the plant TS schedule.
- Steady-state data will be taken at points from 90 percent 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.
- Power increases beyond the previous rating will be made along an established flow control/rod line at increments less than or equal to 3 percent power. Steady-state operating data, including fuel thermal margins, will be taken and evaluated at each step.
- Control system tests will be performed for the feedwater/reactor water level controls and pressure controls. These 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 same performance criteria shall 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 5 and the staff position regarding individual power uprate amendment requests.

4. Performance Testing

a. Systems/Components with Revised Performance Requirements

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[3]As discussed in section I, Introduction, of this safety evaluation, physical limitations of the high pressure turbine may restrict the power uprate if implemented during the current Cycle 8 operation. If rated steam flow for 105 percent power level cannot be reached, the incremental test program described above will be suspended. The test program will resume following the next scheduled refueling outage when modifications to the high pressure turbine nozzles can be made.

Reference 5, Section 5.1 1.9, includes guidelines specifying that pre-operational tests will be performed for systems or components which have revised performance requirements.

The licensee's submittal, NEDO-32907P, did not identify systems or components that have revised performance requirements. The power uprate for Perry does not have a reactor pressure increase. Therefore, none of the plant equipment requires revised performance criteria that is normally associated with a power uprate which implement a concurrent reactor pressure increase. The equipment at Perry is capable of meeting the power uprate requirements within the existing design requirements.

b. Planned Performance Tests

The licensee plans to conduct tests during the ascension to power uprate conditions. The performance tests and associated acceptance criteria are based on the Perry original startup test specifications and previous GE BWR power uprate test programs. The licensee has identified performance tests for the following main systems (Condensate, Feedwater, Main Steam and Off Gas) and associated components:

- Intermediate Range Monitors - testing will assure proper Source Range and Average Power Range monitoring overlap,
- Average Power Range Monitors - setpoint calibration will be conducted,
- Pressure Regulator System - testing will assure incremental regulation and setpoint steps,
- Feedwater Control System - testing will assure proper setpoint changes and incremental regulation,
- Recirculation Flow Control - testing will assure proper step and ramp control,
- Recirculation Flow - setpoint calibration,
- Radiation Measurement - testing will assure proper function of survey equipment including chemical and radiochemical monitoring.

During power uprate testing, numerous plant parameters associated with these systems will be evaluated including, but not limited to: turbine control valve response, turbine vibration, actual power generation, generator cooling, main transformer response, drywell temperature, turbine building temperature, and main steam line flow.

The licensee's program for startup testing follows the guidelines of Reference 5, 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 5 guidelines for uprate testing and meets 10 CFR Part 50, Appendix B requirements for test control. Therefore, the staff concludes that the licensee's power uprate test program is acceptable.

10.5 Operator Training and Human Factors

The operator licensing and human performance evaluation focused on the following five review topics.

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 its letter dated September 9, 1999 (Reference 1), that the power uprate would require some revisions to emergency procedures. The licensee has committed to reviewing the Plant Emergency Instructions (PEI's) for any effects due to power uprate and update as necessary. This review will be based on Section 2.3 of the "Licensing Topical Report Generic Evaluations of GE Boiling Water Reactor Power Uprate (Volume 1)," which includes a list of operator action levels which are sensitive to power uprate.

For uprated power conditions, operator responses to transient, accident and special events are not affected. Most abnormal events result in automatic plant shutdown (scram). All events result in safety-related structures, systems, and components remaining within their design allowables. Power uprate does not change any of the automatic safety functions. After the applicable automatic responses have initiated, the follow-on operator actions for plant safety do not change for the power uprate. The staff finds that the licensee's responses are 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 effect operator reliability or performance. Identify all operator actions that will have their response times changed because of the power uprate. Specify the expected response times before the power uprate and the new (reduced/increased) response times. Discuss why any reduced operator response times are needed. Discuss whether any reduction in time available for operator actions, due to the power uprate, will significantly affect the operator's ability to 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 the power uprate.

By letter dated September 9, 1999, the licensee stated that PEIs include variables and limit curves which define conditions where operator actions are required. Some of these variables and limit curves depend upon the rated reactor power. The operator actions in the PEIs do not change as a result of increasing rated reactor power; only the conditions at which some of the actions are taken will change. No new operator actions will be required. The staff finds 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.

The licensee did not identify any changes to control room alarms, controls, and displays in its letter dated September 9, 1999 or any later submittals. The power uprate does not change any of the automatic safety functions.

A setpoint change for the Main Steam Line High Flow will not change any alarms, controls, or displays. Changes to some Nuclear Instrumentation setpoints will also not change any alarms, controls, or displays; only certain rod block lights will come on at slightly different times. The power axis of the reactor power/flow map has been rescaled such that the maximum uprated core power is defined to be 100 percent uprated power. The physical layout of the screens will remain the same, except some label changes.

The staff finds that the licensee's responses are 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 its letter dated September 9, 1999, that PEI curves and limits are included in the SPDS and will be updated in accordance with the plant procedures and setpoints. The physical layout of the screens is not changing. The staff finds this response to be acceptable.

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

The licensee stated in its letter dated September 9, 1999, that additional training required to operate the plant in an uprated condition is expected to be minimal. The changes to the plant have been identified and the operator training program is being evaluated to determine the specific changes required for operator training. This evaluation includes the effect on the simulation facility.

The licensee committed that 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 staff finds that the licensee's responses are satisfactory and consistent with the existing simulation facility certification.

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

10.6 Plant Life

The licensee's submittal regarding plant life indicated that most equipment is not affected by the power uprate. For equipment that is potentially affected, various Perry programs including, but not limited to, EQ and Flow Accelerated Corrosion that deal with age-related components, will be reviewed and updated as a result of the power uprate. Additionally, the licensee's maintenance rule implementation program provides a mechanism to monitor other electrical and mechanical components important to plant safety to guard against age-related degradation.

The staff has reviewed the licensee's submittal regarding plant life and finds that it is consistent with the guidelines of NEDC-31897P-1, Section 5.11.6, "Plant Life," which have been accepted by the NRC as the generic basis for power uprate amendment requests, and is therefore, acceptable.

11.0 DESIGN BASIS ACCIDENT RADIOLOGICAL CONSEQUENCES

The licensee evaluated the radiological consequences of four postulated DBAs. The analyzed DBAs are (1) LOCA, (2) fuel-handling accident (FHA), (3) control rod drop accident, and (4) instrument line break accident. The licensee stated in the amendment request that the MSLB accident outside containment was not reanalyzed because the mass flow rate from the postulated MSLB is unchanged from the current power level. The licensee concluded that the radiological consequences of a DBA subsequent to implementation of the Perry power uprate remain well below the dose criteria specified in 10 CFR Parts 50 and 100 and GDC 19 of Appendix A to 10 CFR Part 50. The staff finds the licensee's rationale to be acceptable.

(1) LOCA Accident

In Perry License Amendment No. 103, issued on March 26, 1999, the staff evaluated the radiological consequences resulting from a postulated LOCA using the revised accident source term (NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants"). This reanalysis was performed at a reactor power level of 3758 MWt (i.e., the same power level as proposed in the Perry power uprate amendment request). The licensee submitted this amendment request as the lead pilot plant application with the endorsement of the Nuclear Energy Institute. In review of this amendment request, the staff performed confirmatory radiological consequence calculations for four potential fission product release pathways following the postulated LOCA: (1) MSIV leakage, (2) containment leakage, (3) containment bypass leakage, and (4) post-LOCA leakage from engineered safety feature systems outside containment. In addition, the staff evaluated (1) control room habitability, (2) post-LOCA containment water chemistry management, (3) atmospheric relative concentrations (X/Q values), and (4) EQ and plant shielding. The staff's confirmatory calculations performed as part of their review for Amendment No. 103 concluded that the radiological consequences analyzed and submitted by the licensee were acceptable and met the dose criteria specified in 10 CFR Part 50 and GDC 19.

In determining the radiological consequence analyses for the power uprate, the staff assumed a reactor core power level of 3834 MWt, which is equal to 1.02 times the proposed reactor power level of 3758 MWt. This assumption allows for possible instrument errors in determining the reactor power level, as described in RG 1.49, "Power Levels of Nuclear Power Plants." Rather than perform new confirmatory calculations for a power level of 3834 MWt, the staff scaled up the confirmatory analyses from Amendment No. 103 by 2 percent. The staff believes that scaling the dose calculations proportionally to the increase in reactor power level will result in a conservative dose estimate for the power uprate. Using this methodology, the staff finds the resulting doses are still within the dose criteria specified in 10 CFR Part 50 and GDC 19 and are, therefore, acceptable. The resulting doses at the uprated reactor core power level of 3834 MWt are provided in Table 1.

(2) Fuel-Handling Accident

In Perry License Amendment No. 102, issued on March 11, 1999, the staff evaluated the radiological consequences resulting from a fuel handling accident assuming a reactor power level of 3758 MWt (i.e., the same power level as proposed in the Perry power uprate amendment request). The staff concluded that the control room radiological consequences analyzed and
submitted by the licensee were acceptable and met the dose acceptance criteria specified in GDC 19. In review of that request, the staff independently confirmed the licensee's analysis with its own dose calculations.

The offsite dose consequence results were previously submitted to the NRC by letter dated March 16, 1990, in support of a request for approval of TS changes to allow opening of up to six 3/4-inch vent and drain line pathways during refueling activities. In reviewing that request (Perry License Amendment No. 35 dated September 28, 1990), the staff confirmed the licensee's offsite dose consequence results and found they met 10 CFR Part 100 dose acceptance criteria.

For the Perry power uprate amendment request, the staff assumed a core power level of 3834 MWt, which is equal to 1.02 times the proposed reactor power level of 3758 MWt for the radiological consequence analyses to allow for possible instrument errors in determining the reactor power level. Similar to the above LOCA analysis, the staff conservatively scaled up 2 percent from the radiological consequence analyzed in Perry License Amendment Nos. 35 and 102. Using this methodology, the staff finds the resulting doses are still within the dose criteria specified in 10 CFR Part 100 and GDC 19 and are, therefore, acceptable. The resulting doses at an uprated reactor core power level of 3834 MWt are given in Table 1.

(3) Control Rod Drop and Instrument Line Break Accidents

The radiological consequences resulting from a control rod drop accident and an instrument line break accident were analyzed in NUREG-0887, "Safety Evaluation Report related to the Operation of Perry Nuclear Power Plant, Units 1 and 2," dated May 1982 at a reactor core power level of 3834 MWt (which is equal to 1.02 times the proposed reactor power level of 3579 MWt). The staff concluded in NUREG-0887 that the Perry plant is effectively designed to control the release of fission products following the postulated control rod drop accident and the postulated instrument line break accident, and that it met the dose guideline values given in 10 CFR Part 100 and GDC 19. Therefore, the staff performed no new or different dose consequences for this reactor power uprate amendment request.

TABLE 1

Radiological Consequences of Design Basis Accidents
at the
Uprated Reactor Core Power Level
(3834 MWt)

Postulated Accidents	EAB[1]	LPZ[2]	Control Room
Loss of Coolant (TEDE)[3]	23	7	3.5
10 CFR 50 Part Dose Criteria	25	25	5.0

	EAB		LPZ		Control Room	
	Thyroid	WB[4]	Thyroid	WB	Thyroid	WB
Fuel Handling Accident (rem)	48	<1	5.4	<1	24	<1
SRP Acceptance Criteria	75	6	75	6	30	5

[1] Exclusion area boundary
[2] Low population zone
[3] Total effective dose equivalent
[4] Whole body

III STATE CONSULTATION

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

IV 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 May 9, 2000 (65 FR 26858), in connection with the proposed technical specification changes. Accordingly, based upon the environmental assessment, the Commission has determined that issuance of this amendment will not have a significant effect on the quality of the human environment.

V CONCLUSION

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

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REFERENCES

1. FirstEnergy Nuclear Operating Company Letter from John K. Wood to NRC, "License Amendment Request Pursuant to 10 CFR 50.90: Implementation of Power Uprate," September 9, 1999.

2. FirstEnergy Nuclear Operating Company Letter from John K. Wood to NRC, "Response to Request for Additional Information Related to a License Amendment Requesting a Power Uprate," March 1, 2000.

3. FirstEnergy Nuclear Operating Company Letter from John K. Wood to NRC, "Response to Request for Additional Information Related to a License Amendment Requesting a Power Uprate," March 13, 2000.

4. GE Nuclear Energy Topical Report NEDE-32907P, "Safety Analysis Report for Perry 5% Thermal Power Uprate," (Proprietary), September 1999.

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

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

7. Centerior Energy Letter from Robert A. Stratman to NRC, "Response to Generic Letter 94-02 Regarding Long-Term Solutions and Upgrade of Stability Interim Corrective Actions," September 9, 1994.

8. GE Nuclear Energy Licensing Topical Report NEDO-31960-A, "BWR Owner's Group Long-Term Stability Solutions Licensing Methodology," Class I, November 1995 and Supplement 1, November 1995.

9. GE, "General Electric Standard Application for Reactor Fuel - GESTAR II, United States Supplement," NEDE-24011-P-A-10-US, (Proprietary), March 1991; and NEDE-24011-A-10-US, (Non-Proprietary), March 1991.

10. GE Service Information Letter (SIL) 377, "RCIC Startup Transient Improvement with Steam Bypass," June 1, 1982.

11. FirstEnergy Letter from Lew W. Myers to NRC, "Implementation of the SAFER/GESTR Loss of Coolant Accident Methodology," June 7, 1999.

12. Nuclear Management and Resources Council, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," NUMARC-8700, November 1987.

13. SECY-97-042, "Response to OIG Event Inquiry 96-04S Regarding Maine Yankee," February 18, 1997.

14. NRC Letter to General Electric Company (W.T. Russell to P.W. Marriott), "Staff Position Concerning Generic Boiling Water Reactor Power Uprate Program," dated September 30, 1991.

15. FirstEnergy Nuclear Operating Company Letter from John K. Wood to NRC, "Implementation of Power Uprate License Amendment Request (TAC No. MA6459)" dated May 11, 2000.