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

May 30, 2012

Mr. Chris Burton, Vice President
Shearon Harris Nuclear Power Plant
Progress Energy Carolinas, Inc.
Post Office Box 165, Mail Code: Zone 1
New Hill, North Carolina 27562-0165

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 – ISSUANCE OF AMENDMENT RE: MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NO. ME6169)

Dear Mr. Burton:

The Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 139 to Renewed Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant Unit 1 (HNP). The amendment is in response to an application submitted by Carolina Power & Light, dated April 28, 2011, as supplemented by letters dated June 23, August 3, August 15, August 25, August 30, August 31, September 6, September 7, October 20, October 21, October 28, November 28, December 20, 2011, February 9, and March 26, 2012.

The amendment revises the HNP renewed facility operating license and certain technical specifications to implement an increase of approximately 1.66 percent in rated thermal power from the current licensed thermal power of 2900 megawatts thermal (MWt) to 2948 MWt. The changes are based on increased feedwater flow measurement accuracy, which will be achieved by utilizing Cameron International Corporation (formerly Caldon) Cameron Leading Edge Flow Meter CheckPlus system to improve the HNP calorimetric heat balance measurement accuracy.

A copy of the related safety evaluation is enclosed. A notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Araceli T. Billoch Colón

Araceli T. Billoch Colón, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Amendment No. 139 to NPF-63
2. Safety Evaluation

cc w/enclosures: Distribution via ListServ

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

CAROLINA POWER & LIGHT COMPANY, et al.

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 139
Renewed License No. NPF-63

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

   A. The application for amendment by Carolina Power & Light Company (the licensee), dated April 28, 2011,[1] as supplemented by letters,[2] 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 Title 10 of the *Code of Federal Regulations* (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.

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1 Agencywide Documents Access and Management System (ADAMS) No. ML11124A180.

2 June 23, 2011 (ML11179A052); August 3, 2011 (ML11221A185); August 15, 2011 (ML11235A516); August 25, 2011 (ML11243A121); August 30, 2011 (ML11250A097); August 31, 2011 (ML11255A132); September 6, 2011 (ML11256A026); September 7, 2011 (ML11256A029); October 20, 2011 (ML11299A023); October 21, 2011 (ML11300A183); October 28, 2011 (ML11308A028); November 28, 2011 (ML11340A078); December 20, 2011 (ML12010A079); February 9, 2012 (ML12052A253) and March 26, 2012 (ML12100A160).

2. Accordingly, the license is amended by changes to the Renewed Facility Operating License and Technical Specifications, as indicated in the attachment to this license amendment; and paragraphs 2.C.(1) and 2.C.(2) of Renewed Facility Operating License No. NPF-63 are hereby amended to read as follows:

   (1) <u>Maximum Power Level</u>

   Carolina Power & Light Company is authorized to operate the facility at reactor core power levels not in excess of 2948 megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

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

   The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 139 , are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. Accordingly, the license is amended by approving changes to the Technical Specifications as indicated in the safety evaluation attached to this license amendment.

4. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Michele G. Evans

Michele G. Evans, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. NPF-63
and the Technical Specifications

Date of Issuance: May 30, 2012

ATTACHMENT TO LICENSE AMENDMENT NO. 139

RENEWED FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace the following page of the renewed facility operating license with the revised page. The revised page is identified by amendment number and contains lines in the margin indicating the areas of change.

Remove	Insert
Page 4	Page 4

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 lines in the margins indicating the areas of change.

Remove	Insert
1-5	1-5
2-2	2-2
2-4	2-4
2-6	2-6
2-10	2-10
3/4 3-4	3/4 3-4
3/4 3-13	3/4 3-13
3/4 7-2	3/4 7-2

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

Carolina Power & Light Company is authorized to operate the facility at reactor core power levels not in excess of 2948 megawatts thermal (100 percent rated core power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No.139 , are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

Carolina Power & Light Company shall comply with the antitrust conditions delineated in Appendix C to this license.

(4) Initial Startup Test Program (Section 14)[1]

Any changes to the Initial Test Program described in Section 14 of the FSAR made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(5) Steam Generator Tube Rupture (Section 15.6.3)

Prior to startup following the first refueling outage, Carolina Power & Light Company shall submit for NRC review and receive approval if a steam generator tube rupture analysis, including the assumed operator actions, which demonstrates that the consequences of the design basis steam generator tube rupture event for the Shearon Harris Nuclear Power Plant are less than the acceptance criteria specified in the Standard Review Plan, NUREG-0800, at 15.6.3 Subparts II(1) and (2) for calculated doses from radiological releases. In preparing their analysis Carolina Power & Light Company will not assume that operators will complete corrective actions within the first thirty minutes after a steam generator tube rupture.

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[1] The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report and/or its supplements wherein the license condition is discussed.

PROCESS CONTROL PROGRAM

1.25 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71 and State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.26 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.27 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.28 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2948 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.29 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage. 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. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

REPORTABLE EVENT

1.30 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

SHUTDOWN MARGIN

1.31 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.32 For these Specifications, the SITE BOUNDARY shall be identical to the EXCLUSION AREA BOUNDARY defined above.

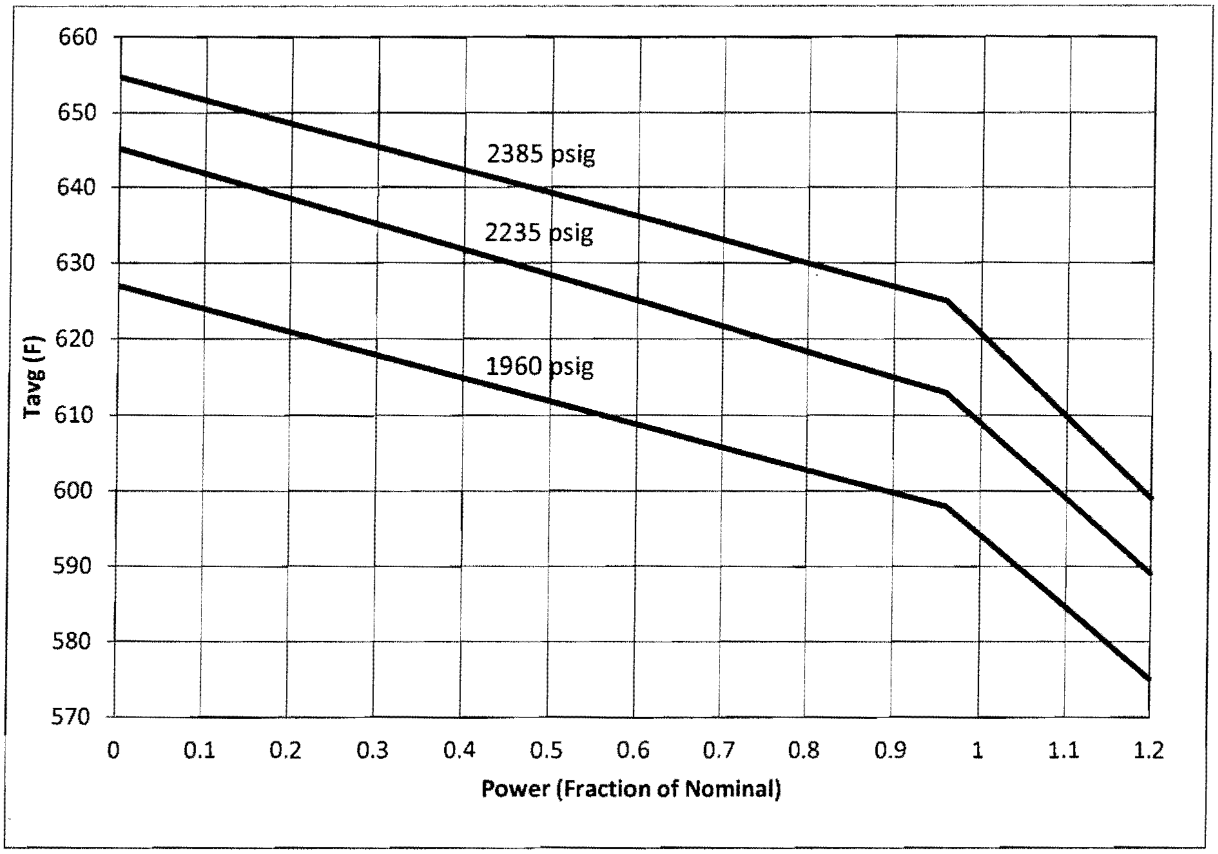

FIGURE 2.1-1

REACTOR CORE SAFETY LIMITS – THREE LOOPS IN OPERATION
WITH MEASURED RCS FLOW ≥ [293,540 GPM X (1.0 + C_1)]

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1.	Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2.	Power Range, Neutron Flux					
	a. High Setpoint	5.83	4.56	0	≤ 108% of RTP** See NOTES 7, 8	≤ 109.5% of RTP**
	b. Low Setpoint	7.83	4.56	0	≤ 25% of RTP** See NOTES 7, 8	≤ 26.8% of RTP**
3.	Power Range, Neutron Flux, High Positive Rate	2.33	0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds See NOTES 7, 8	≤ 6.3% of RTP** with a time constant ≥ 2 seconds
4.	Power Range, Neutron Flux, High Negative Rate	2.33	0.83	0	≤ 5% of RTP** with a time constant ≥ 2 seconds See NOTES 7, 8	≤ 6.3% of RTP** with a time constant ≥ 2 seconds
5.	Intermediate Range, Neutron Flux	17.0	8.41	0	≤ 25% of RTP**	≤ 30.9% of RTP**
6.	Source Range, Neutron Flux	17.0	10.01	0	≤ 10^5 cps	≤ 1.4×10^5 cps
7.	Overtemperature ΔT	9.0	7.31	Note 5	See Note 1	See Note 2
8.	Overpower ΔT	4.0	2.32	1.3	See Note 3	See Note 4
9.	Pressurizer Pressure-Low	5.0	1.52	1.5	≥ 1960 psig	≥ 1948 psig
10.	Pressurizer Pressure-High	7.5	1.52	1.5	≤ 2385 psig	≤ 2397 psig
11.	Pressurizer Water Level-High	8.0	3.42	1.75	≤ 92% of instrument span	≤ 93.5% of instrument span

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT		TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
19.	Reactor Trip System Interlocks					
	a. Intermediate Range Neutron Flux, P-6	N.A.	N.A.	N.A.	≥ 1 x 10-10 amp	≥ 6 x 10-11 amp
	b. Low Power Reactor Trips Block, P-7					
	1) P-10 input	N.A.	N.A.	N.A.	≤ 10% of RTP**	≤ 12.1% of RTP**
	2) P-13 input	N.A.	N.A.	N.A.	≤ 10% RTP** Turbine Inlet Pressure Equivalent	≤ 12.1% RTP** Turbine Inlet Pressure Equivalent
	c. Power Range Neutron Flux, P-8	N.A.	N.A.	N.A.	≤ 49% of RTP**	≤ 51.1% of RTP**
	d. Power Range Neutron Flux, P-10	N.A.	N.A.	N.A.	≥ 10% of RTP**	≥ 7.9% of RTP**
	e. Turbine Inlet Pressure, P-13	N.A.	N.A.	N.A.	≤ 10% RTP** Turbine Inlet Pressure Equivalent	≤ 12.1% RTP** Turbine Inlet Pressure Equivalent
20.	Reactor Trip Breakers	N.A.	N.A.	N.A.	N.A.	N.A.
21.	Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	N.A.
22.	Reactor Trip Bypass Breakers	N.A.	N.A.	N.A.	N.A.	N.A.

**RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 3: (Continued)

K_6	=	0.002/°F for T > T" and K_6 = 0 for T≤ T",
T	=	As defined in Note 1,
T"	=	Reference T_{avg} at RATED THERMAL POWER (≤ 588.8°F)
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	0 for all ΔI.

NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 1.4% of ΔT span for ΔT input and 0.2% of ΔT span for T_{avg} input.

NOTE 5: The sensor error is: 1.3% of ΔT span for $\Delta T/T_{avg}$ temperature measurements; and 1.0% of ΔT span for pressurizer pressure measurements.

NOTE 6: The sensor error (in % span of Steam Flow) is: 1.1% for steam flow; 1.8% for feedwater flow; and 2.4% for steam pressure.

NOTE 7: If the as-found channel setpoint is outside its predefined as-found tolerance, the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

NOTE 8: The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Trip Setpoint in Table 2.2-1 (Nominal Trip Setpoint (NTSP)) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine NTSPs and the as-found and the as-left tolerances are specified in EGR-NGGC-0153, "Engineering Instrument Setpoints." The as-found and as-left tolerances are specified in PLP-106, "Technical Specification Equipment List Program and Core Operating Limits Report."

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
16. Underfrequency - - Reactor Coolant Pumps (Above P-7)	2/pump	2/train	2/train	1	6
17. Turbine Trip (Above P-7)					
a Low Fluid Oil Pressure	3	2	2	1	6
b. Turbine Throttle Valve Closure	4	4	1	1	10
18. Safety Injection Input from ESF	2	1	2	1, 2	13
19. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	7
b. Low Power Reactor Trips Block, P-7					
1) P-10 Input	4	2	3	1	7
or					
2) P-13 Input	2	1	2	1	7
c. Power Range Neutron Flux, P-8	4	2	3	1	7
d. Power Range Neutron Flux, P-10	4	2	3	1, 2	7
e. Turbine Inlet Pressure, P-13	2	1	2	1	7

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
19. Reactor Trip System Interlocks (Continued)						
b. Low Power Reactor Trips Block, P-7	N.A.	R(4)	R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	R	N.A.	N.A.	1
d. Power Range Neutron Flux P-10	N.A.	R(4)	R	N.A.	N.A.	1, 2
e. Turbine Inlet Pressure, P-13	N.A.	R	R	N.A.	N.A.	1
20. Reactor Trip Breaker	N.A.	N.A.	N.A.	M (7, 9, 10)	N.A.	1, 2, 3*, 4*, 5*
21. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M (7)	1, 2, 3*, 4*, 5*
22. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	M (7, 13) R (14)	N.A.	1, 2, 3*, 4*, 5*

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING 3 LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	49
2	32
3	15

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 139 TO RENEWED FACILITY

OPERATING LICENSE NO. NPF-63

CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By letter dated April 28, 2011,[1] to the U.S. Nuclear Regulatory Commission (NRC, the Commission) as supplemented by additional letters,[2] Carolina Power & Light Company (CP&L, the licensee), doing business as Progress Energy Carolinas (PEC) Inc., requested changes to the Renewed Facility Operating License (FOL) No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The amendment revises the FOL and certain technical specifications (TSs) to implement an increase of approximately 1.66 percent in rated thermal power (RTP) from the current licensed thermal power (CLTP) of 2900 megawatts thermal (MWt) to 2948 MWt by means of a measurement uncertainty recapture (MUR) power uprate.

The changes are based on increased feedwater (FW) flow measurement accuracy, which will be achieved by utilizing Cameron International Corporation (formerly Caldon) Leading Edge Flow Meter (LEFM) CheckPlus system instrumentation to improve HNP calorimetric heat balance measurement accuracy. The LEFM system was installed in HNP during the fall 2010 refueling outage (RFO).

The NRC staff's original proposed no significant hazards consideration determination was published in the *Federal Register* on September 13, 2011 (76 FR 56486). The supplemental letters contained clarifying information, did not expand the scope of proposed license amendment request (LAR), and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

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1 Agencywide Documents Access and Management System (ADAMS) No. ML11124A180

2 June 23, 2011 (ML11179A052); August 3, 2011 (ML11221A185); August 15, 2011 (ML11235A516); August 25, 2011 (ML11243A121); August 30, 2011 (ML11250A097); August 31, 2011 (ML11255A132); September 6, 2011 (ML11256A026); September 7, 2011 (ML11256A029); October 20, 2011 (ML11299A023); October 21, 2011 (ML11300A183); October 28, 2011 (ML11308A028); November 28, 2011 (ML11340A078); December 20, 2011 (ML12010A079); February 9, 2012 (ML12052A253); and March 26, 2012 (ML12100A160).

2.0 BACKGROUND

Nuclear power plants are licensed to operate at a specified maximum core thermal power, often called RTP. Appendix K, "[Emergency Core Cooling System] ECCS Evaluation Models," of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, formerly required licensees to assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level when performing loss-of-coolant accident (LOCA) and ECCS analyses. This requirement was included to ensure that instrumentation uncertainties were adequately accounted for in the safety analyses. In practice, many of the design bases analyses assumed a 2 percent power uncertainty, consistent with 10 CFR Part 50, Appendix K.

A revision to 10 CFR Part 50, Appendix K, effective July 31, 2000, allows licensees to use a power level less than 1.02 times the RTP for the LOCA and ECCS analyses, but not less than the licensed power level, based on the use of state-of-the art FW flow measurement devices that provide a more accurate calculation of power. Licensees can use a lower uncertainty in the LOCA and ECCS analyses provided the licensee has demonstrated that the proposed value adequately accounts for instrumentation uncertainties. Because there continues to be substantial conservatism in other Appendix K requirements, sufficient margin to ECCS performance in the event of a LOCA is preserved.

However, the final rule did not allow increases in licensed power levels. Because the licensed power level for a plant is contained in the plant's operating license, proposals to raise the licensed power level must be reviewed and approved under the LAR process. HNP is currently licensed to operate at a maximum power level of 2900 MWt, which includes a 2 percent margin in the ECCS evaluation model to allow for uncertainties in core thermal power measurement as was previously required by 10 CFR Part 50, Appendix K.

Uncertainty in FW flow measurement is the most significant contributor to core power measurement uncertainty. The licensee states that use of the Cameron International Corporation LEFM CheckPlus System, which was installed in HNP during the fall 2010 RFO, provides a more accurate measurement of FW flow compared to the accuracy of the venturi-based instrumentation originally installed at HNP. The purpose of the proposed change is to obtain a power uprate on the basis of a plant modification that would result in an improved accuracy of FW flow measurement, that will be used to calculate reactor thermal power. Installation of an ultrasonic flow meter (UFM), called LEFM CheckPlus System, to measure FW flow would allow the licensee to operate the plant with a reduced instrument uncertainty margin of approximately 0.34 percent and an increased power level of approximately 1.66 percent above the licensed thermal power.

The NRC issued Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," on January 31, 2002,[3] to provide guidance to licensees on the scope and detail of the information that should be provided to the NRC for MUR power uprate applications. While RIS 2002-03 does not constitute an NRC

3 William D. Beckner, NRC, "NRC Regulatory Issue Summary 2002-03: Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," January 31, 2002 (ADAMS Accession No. ML013530183).

requirement, it is available to aid licensees in optimizing an MUR power uprate LAR, and to provide guidance to the NRC staff for conducting the review. The licensee stated in its application dated April 28, 2011, that the LAR was submitted consistent with the guidance of RIS 2002-03.

3.0 EVALUATION

3.1 Human Factors Evaluation

Regulatory Evaluation

The NRC's staff human factor review addresses whether the licensee has adequately considered the effects of the proposed MUR on programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC staff's human factors evaluation was conducted to confirm that operator performance would not be adversely affected as a result of system and procedure changes made to implement the proposed MUR power uprate. The scope of the review included changes to operator actions, human-system interfaces, and procedures and training needed for the proposed MUR power uprate. The human factors evaluation determines conformance to the NRC staff's guidance in Section VII of RIS 2002-03.

Technical Evaluation

RIS 2002-03, Attachment 1, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," describes the review guidance related to human factors (Section VII, Items 1 through 4) for MUR applications. The NRC staff's evaluation of the licensee's responses to these items in the LAR and additional clarifications in the supplemented letters are provided below.

3.1.1 Operator Actions

Section VII.1 of Attachment 1 to RIS 2002-03 requests that the licensee make a statement confirming that operator actions that are sensitive to the power uprate, including any effects on the time available for operator actions, have been identified and evaluated. The licensee stated in its application that the existing operator actions are not affected by the power uprate and there is no reduction in time for required operator actions. The NRC staff requested the licensee to clarify whether this statement refers to the time required for the operator to complete the operator actions credited in the analysis, or the time available to operators as it is outlined in the procedures and accident scenario analysis. The license stated in the letter dated August 30, 2011, that a review of the design basis events where response times are credited, determined that operator actions are not impacted by the MUR power uprate. The NRC staff concludes that the proposed MUR power uprate will not adversely impact operator actions and their response times. The NRC staff finds that the statements provided by the licensee are in conformance with Section VII.1 of Attachment 1 to RIS 2002-03.

3.1.2 Emergency and Abnormal Operating Procedures

The licensee stated in its submittal, dated April 28, 2011, that there are no mitigating actions or step changes as a result of the power uprate. However, the NRC staff review identified two emergency operating procedure (EOP) setpoints that require revision. These EOP setpoints were developed using full power reactor coolant system (RCS) hot leg temperature and the full power RCS hot leg temperature changed with the power uprate. The licensee stated in the April 28, 2011, submittal that these EOP setpoints will be revised to reflect a total core power of 2958 MWt or 102 percent of 2900 MWt, which bounds the power uprate. The licensee concluded that the EOP and abnormal operating procedure (AOP) changes do not significantly affect operator actions and mitigation strategies.

The licensee identified in the April 28, 2011, submittal how these revised AOP and EOP set points would affect the safety margins associated with the credited operator actions by the AOPs requiring revision (FW malfunction, main transformer trouble, and rapid down-power) not being events that contain operator actions included in the safety analyses. Thus, these AOP revisions do not impact safety margins associated with credited operator actions. Safety margins provided by EOP setpoints are established through the definitions of the associated emergency response guideline (ERG) footnotes and the methodology prescribed for the development of the values of those footnotes specified in the ERG footnote basis document. All EOP setpoints have been screened for the impact of the proposed power uprate. The setpoints impacted by the power uprate have been recalculated using the methodologies consistent with those established by the ERG footnote basis document, thus ensuring the inherent safety margin is maintained.

The NRC staff has reviewed the licensee's evaluation of the effects of the MUR power uprate on the HNP EOPs and AOPs. The NRC staff concludes that the proposed MUR power uprate does not present any adverse impacts on the EOPs and AOPs. The NRC staff finds that the statements provided by the licensee are in conformance with Sections VII.2.A, Attachment 1 to RIS 2002-03.

3.1.3 Changes to Control Room Controls, Displays, and Alarms

In its submittal, dated April 28, 2011, the licensee described changes/modifications to control room controls, displays (including the safety parameter display system), and alarms related to the proposed MUR power uprate. Notable proposed modifications to controls, displays, and alarms include:

- Instruments associated with turbine first-stage pressure will require scaling changes.
- Instrument loops are affected by the power uprate (possible indicator replacement, calibration span, and/or scaling).
- Plant computer points will be added and/or changed for the revised calorimetric algorithm and the FW LEFM.
- The FW flow data will be displayed in a new LEFM electronic cabinet. The display provides system status or monitored process parameters.
- The LEFM system will provide input to the secondary calorimetric. LEFM system parameters will be displayed in the main control room through an emergency response facility information system interface.
- System alerts operations personnel to LEFM trouble through main control room annunciator computer alarm reactor when the system loses a plane of operation, has a

channel that reaches an alert or fail condition, a high temperature condition, or other failures.

The licensee also stated that no significant safety parameter display system changes are anticipated as a result of the power uprate. The licensee stated that all modifications to the control room and training on these changes will be provided prior to MUR power uprate implementation.

The NRC staff has reviewed the licensee's evaluation and proposed changes to the control room. The NRC staff concludes that the proposed changes do not present any adverse effects to the operators' functions in the control room. The NRC staff finds that the statements provided by the licensee are in conformance with Sections VII.2.B and VII.3 of Attachment 1 to RIS 2002-03.

3.1.4 Control Room Plant Reference Simulator

The licensee stated that the power uprate is being implemented under the plant modification process administrative controls. Simulator required changes resulting from the power uprate will be evaluated, implemented, and tested per approved procedures. The licensee stated the simulator fidelity will be revalidated per approved procedures. The licensee also stated that any required simulator modifications will be completed in time to support operator training prior to power uprate implementation.

The NRC staff has reviewed the licensee's proposed changes to the control room plant reference simulator and concludes that the changes do not present any adverse effects on the plant's simulator. The NRC staff finds that the statements provided by the licensee are in conformance with Sections VII.2.C of Attachment 1 of RIS 2002-03.

3.1.5 Operator Training Program

The licensee stated in its submittal that the operator training program will be developed and the operations staff trained on the plant modifications, TS changes, new relocated TS and design basis requirements attachment, and procedural changes prior to MUR power uprate implantation.
The NRC staff reviewed the licensee's proposed changes to the control room plant reference simulator and concludes that the changes do not present any adverse effects on the plant's simulator. The NRC staff finds that the statements provided by the licensee are in conformance with Sections VII.2.D of Attachment 1 of RIS 2002-03.

3.1.6 Human Factors Conclusion

The NRC staff has completed its human factors review of the licensee's proposed changes and concludes that the licensee has adequately considered the impact of the proposed MUR power uprate on operator actions, EOPs and AOPs, control room components, plant simulator and operator training programs.

3.2 Dose Consequence Analysis

Regulatory Evaluation

RIS 2002-03 recommends that to improve efficiency of the NRC staff's review, licensees requesting an MUR uprate should identify existing postulated design-basis accident (DBA) analyses of record (AOR), which bound plant operation at the proposed uprated power level. For any existing postulated DBA AOR that does not bound the proposed uprated power level, the licensee should provide a detailed discussion of the reanalysis.

The NRC staff reviewed the impact of the proposed changes on postulated DBA radiological dose consequences. In HNP Amendment No. 107 dated October 12, 2001 (ADAMS Accession No. ML012830516), the NRC staff approved a full-scope implementation of the alternative source term (AST) in accordance with 10 CFR 50.67 "AST," and following the guidance and methodology provided in applicable sections of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors" (ADAMS Accession No. ML003716792).

The NRC staff conducted this evaluation to verify that the results of the DBA radiological dose consequence analyses continue to meet the dose acceptance criterion given in 10 CFR 50.67 for offsite doses, as defined by RG 1.183, and 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 19: "Control Room" with respect to control room habitability. The applicable acceptance criteria are 5 roentgen equivalent man (rem) total effective dose equivalent (TEDE) in the control room, 25 rem TEDE at the exclusion area boundary, and 25 rem TEDE at the outer boundary of the low population zone. The NRC staff utilized the regulatory guidance provided in applicable sections of RG 1.183, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 6.4, "Control Room Habitability System," SRP Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," and the licensing basis provided in HNP Updated Final Safety Analysis Report (UFSAR), Chapter 15, "Accident Analyses," in performing this review.

Technical Evaluation

The NRC staff reviewed the regulatory and technical analyses performed by the licensee in support of its proposed MUR power uprate LAR, as they relate to the radiological dose consequences of postulated DBA analyses. Information regarding these analyses was provided by the licensee in Enclosure 2 to the April 28, 2011, application. The findings of the NRC staff's evaluations are based on the review of the descriptions and results of the licensee's analyses and other supporting information docketed by the licensee.

The NRC staff reviewed the impact of the proposed 1.66 percent MUR power uprate on all postulated DBA radiological dose consequence analyses, as documented in Chapter 15 of the HNP UFSAR. The NRC staff reviewed the following accidents:

- LOCA
- Small Break LOCA (SBLOCA)
- Fuel Handling Accident (FHA)
- Main Steam Line Break (MSLB)

- Control Rod Eject Accident (CREA)
- Steam Generator Tube Rupture (SGTR)
- Loss of Offsite Power (LOOP)
- Locked Rotor (LR)
- Single Rod Control Cluster Assembly (RCCA)
- Chemical and Volume Control System (CVCS)

The radiological dose consequence analyses are based upon the AST, with acceptance criteria as specified above in the Regulatory Evaluation. The radiological dose consequence analyses are potentially impacted in three areas: 1) the core and coolant activities prior to the accident (source term), 2) the fuel failure resulting from the accident, and 3) the secondary side steam releases following the accident. The NRC staff's review of these three areas is provided below.

3.2.1 Core and Coolant Activities Prior to the Accident

The licensee stated that the radiological dose consequence analyses were performed using the core inventory that assumes 2958 MWt, or 102 percent of 2900 MWt, and therefore remain applicable at the proposed power uprate conditions. In Enclosure 2, Table II-1, "FSAR [Final Safety Analysis Report] Accidents, Transients and Other Analyses," of the application, the licensee provided a brief overview of the accident/transient analyses and other analyses contained in the HNP UFSAR. The above referenced table also includes information regarding the assumed core power level in each analysis, whether these analyses remain bounding for the proposed MUR power uprate, as well as a reference to the NRC staff's previous approval of each analysis, if applicable. Enclosure 2, Section II.2, "Discussion of Events" of the application discussed the DBA analyses.

The NRC staff reviewed the above referenced tables and sections of Enclosure 2 of the submittal, and compared the radiological dose consequence analyses with HNP's AST amendment, dated October 12, 2001, and Chapter 15 of the HNP UFSAR. Through the comparative analyses described above, the NRC staff confirmed that HNP's current postulated DBA AOR for the LOCA, SBLOCA, FHA, CREA, SGTR, LOOP, LR, RCCA, and CVCS were all previously performed at 2958 MWt, or 102 percent of the currently licensed thermal power of 2900 MWt. HNP's postulated DBA AOR for the MSLB was previously analyzed at the currently licensed thermal power of 2900 MWt. The NRC staff also determined that the remaining accidents, transients, and/or analyses of Enclosure 2, Section II.2, were either specifically bounded by the applicable above stated postulated DBAs, or by the previously assumed power level of 2958 MWt.

If the LEFM CheckPlus System experiences operational limitations (maintenance or fail mode) the licensee has accounted for a potential increase in measurement uncertainty beyond that assumed in the radiological dose consequence analyses. Within 72 hours of the detection of an LEFM CheckPlus System limitation at a power of 2900 MWt or greater, the licensee will reduce power to account for the potential of a measurement uncertainty increase. The reduction in power will ensure that the current licensing basis dose consequence analyses remain bounding. The acceptability of the proposed LAR, as it pertains to the assumption that there is no increase in uncertainty (beyond the assumed 0.34 percent) up to 72 hours after detecting a system limitation at a power of 2900 MWt or greater, is discussed in Section 3.6 of the safety evaluation (SE).

3.2.2 Fuel Failure and Melting Assumptions

The licensee stated in the application that the fuel failure and melting assumptions do not change and will be verified during the standard core reload process. The acceptability of the proposed license application, as it pertains to the evaluation of fuel failure due to exceeding departure from nucleate boiling ratio (DNBR) and/or fuel centerline melt limits, is discussed in Section 3.6 of this SE.

3.2.3 Secondary Side Steam Releases

As stated by the licensee in Enclosure 2, Section II.2.38, "Radiological Consequences," and further documented in Table II-3, "Revised Accident Doses" of the LAR, the specific postulated DBA analyses that experienced a change (increase) in secondary side steam releases as a result of the proposed MUR power uprate are the MSLB, LOOP, CREA, LR, RCCA, and SBLOCA analyses. To reflect the change in power measurement uncertainty, the steam release and FW flow rates, and the radiological dose consequences for these accidents were recalculated at 102 percent of 2900 MWt, plus 12.4 MWt for reactor coolant pump (RCP) heat input. In Enclosure 2, Section II.2.38, the licensee also stated that the standard steam release for dose calculations at HNP analyzes four steam release events: MSLB, LR, LOOP, and loss of load. The licensee further asserted that the loss of load event is not part of the HNP radiological licensing basis. Therefore, the only postulated DBAs included in the steam mass release analysis are the MSLB, LR, and LOOP. The licensee provided benchmark comparisons in Enclosure 2, Table II-2, "Steam Released and Main Feedwater Flows," of the LAR to support the application.

The NRC staff reviewed the above referenced tables and sections of Enclosure 2 of the submittal dated April 28, 2011. The NRC staff conducted a quantitative analysis of the licensee's revised steam release rates and FW flows by taking a ratio of the newly calculated MUR values against the current AOR as documented in the HNP UFSAR. In Enclosure 2, Section II.2.38, the licensee also stated that the release pathways, radiological atmospheric dispersion factors, and the dose conversion factors for HNP were unchanged from the previously analyzed AST amendment dated October 12, 2001.

The NRC staff found the increase in radiological dose consequences to be representative of the increase in steam to the environment. These analyses were conducted for time intervals of 0-2 hours and 2-8 hours, and repeated for each pair (i.e. proposed MUR and AOR) of data given in Enclosure 2, Table II-2 of the LAR. From the results, the NRC staff determined that all of the licensee's revised steam release and FW flow rates exhibited less than a 5 percent change in value, yielding a limiting change in value ratio of 1.04 (rounded up) for all steam release and FW flow cases.

As mentioned above, similar ratio analyses were conducted for each value given in Enclosure 2, Table II-3. From these results, the NRC staff also determined that all of the licensee's revised postulated DBA analyses exhibited less than an 8 percent change in value, with a limiting ratio of 1.08 for radiological dose consequences at the exclusion area boundary (EAB), low population zone (LPZ) and control room for all analyzed events.

3.2.4 Dose Consequence Analysis Conclusion

The NRC staff reviewed the assumptions and parameters used by the licensee to assess the radiological consequences of the postulated DBA radiological dose consequence analyses at the proposed uprated power level. The NRC staff finds that operating the HNP at the proposed uprated power level will continue to meet the applicable dose acceptance criteria given in 10 CFR 50.67, of 25 rem TEDE for both the EAB and LPZ, and 5 rem TEDE for the control room following implementation of the proposed 1.66 percent MUR power uprate. The NRC staff further finds with reasonable assurance that HNP, as modified by this approved license amendment, will continue to provide sufficient safety margins, with adequate defense-in-depth, to address unanticipated events and to compensate for uncertainties in accident progression, analysis assumptions, and input parameters. Therefore, the NRC staff concludes that the proposed LAR is acceptable with respect to the radiological dose consequences of postulated DBAs.

3.3 Fire Protection

Regulatory Evaluation

The purpose of the fire protection program established by National Fire Protection Association Standard (NFPA) 805 is to provide assurance, through a defense-in-depth philosophy, that the fire protection objectives are satisfied.[4] The NRC staff's review focused on the effects of increased decay heat due to the MUR power uprate and on the plant's ability to achieve and maintain the nuclear safety performance criteria.

The NRC's acceptance criteria for the fire protection program are based on (1) 10 CFR 50.48, "Fire protection," insofar as it requires the development of a fire protection program to ensure, among other things, the capability to safely shutdown the plant; (2) GDC 3 "Fire Protection" of Appendix A to 10 CFR Part 50, insofar as it requires that, structures, systems, and components (SSCs) important to safety be designed and located to minimize the probability and effect of fires, that noncombustible and heat resistant materials be used, and that fire detection and suppression systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; and (3) GDC 5 "Sharing of Structures, Systems, and Components" of Appendix A to 10 CFR Part 50, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

Technical Evaluation

The licensee stated in its application dated April 28, 2011, that the LAR was developed consistent with the guidelines in RIS 2002-03. In the LAR, the licensee re-evaluated the applicable SSCs and safety analyses at the proposed MUR core power level of 2948 MWt against the previously analyzed core power level of 2900 MWt. The NRC staff reviewed Enclosure 2 to the submittal, specifically, the fire protection related portion of the LAR.

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4 On May 29, 2008 the licensee submitted an LAR to adopt NFPA 805 (ADAMS Accession No. ML081560640). On June 28, 2010, the NRC staff issued Amendment No. 133 to complete the HNP transition to NFPA 805 (ADAMS Accession No. ML101130535).

The NRC staff review covered the impact of the proposed MUR power uprate on the results of the plant's ability to achieving and maintaining the nuclear safety performance criteria or safe-shutdown capability as noted in RIS 2002-03, Attachment 1, Sections II and III. The review focused on the effects of the MUR power uprate on the nuclear safety performance criteria and increase in decay heat generation following plant trips. The NRC staff's review of the April 28, 2011, submittal identified areas in which additional information was necessary to complete the review of the proposed MUR power uprate LAR.

The NRC staff noted that Enclosure 2 to the LAR, Section VII.1, "Operator Actions," states that "The safety analysis reviews determined that the existing required operator actions are not affected by the power uprate." The NRC staff requested the licensee to verify that: (1) the MUR power uprate will not require any change in procedures and resources necessary for systems required to achieve the nuclear safety performance criteria and are adequate for the MUR power uprate, and (2) any effects from additional heat in the plant environment from the increased power will not interfere with existing operator actions (referred to as recovery actions per NFPA 805 licensing basis) being performed at their designated time and place as identified in the HNP fire protection program.

In its response dated August 3, 2011, the licensee stated that a review of the impact of the power uprate determined that there will be no changes in procedures and resources necessary for systems required to achieve and maintain the nuclear safety performance criteria. The licensee clarified that the existing procedures and resources are determined to be adequate for the MUR power uprate as currently written. The only nuclear safety performance resource identified as being impacted and requiring change was a recalculation of K_{eff} during cooldown. The licensee concluded on the August 3, 2011, letter that the impact of the MUR on K_{eff} shows that while the calculation required revision, the nuclear safety performance criteria are still met. The licensee referenced calculation HNP-F/NFSA-0171, "HNP Reactor Coolant System Cooldown Without Boration," for this analysis. The need for revision to this analysis for the MUR was addressed, and the calculation has already been revised as stated in the August 3, 2011, letter.

Further, the licensee stated that the effects of the additional heat in the plant environment will have no impact on existing recovery actions. The fire areas where the credited recovery actions to achieve and maintain the NFPA 805 nuclear safety performance criteria are all in the reactor auxiliary building where at least one of two safety-related chillers have been analyzed to be available for all postulated fires. For areas where the fire is postulated, actions will not be taken until the area can be accessed (i.e., after the postulated fire is extinguished and the smoke has been ventilated).

The licensee's response satisfactorily addresses the staff's concerns, since the licensee verified that the procedures and resources necessary for systems required to achieve and maintain the nuclear safety performance criteria are not affected by the MUR power uprate. For the MUR power uprate condition, the licensee reviewed plant areas where recovery actions are being performed for achieving and maintaining the nuclear safety performance criteria following a fire to determine if additional heat due to MUR power uprate conditions could adversely impact those defined recovery actions. The licensee indicated that the MUR power uprate does not affect the existing recovery actions due to additional heat in the plant environment. Note that this SE does not approve any new or existing recovery actions concerning HNP to achieve the nuclear safety performance criteria.

The NRC staff also noted that some plants credit aspects of their fire protection system for other than fire protection activities (e.g., utilizing the fire water pumps and water supply as backup cooling or inventory for non-primary reactor systems). The NRC staff requested the licensee to verify whether this is applicable to the HNP fire protection system. If so, the NRC staff requested that the licensee identify the specific situations applicable to HNP and discuss to what extent, if any, the MUR power uprate effects these "non-fire protection" aspects of the plant fire protection system.

In a letter to NRC dated August 3, 2011, the licensee responded that the water from the fire protection system is not normally used in the plant for non-fire protection-related functions; however, its use as a potential alternate source of water has been identified in specific off-normal conditions. Procedural guidance for the alternate uses is provided in HNP Operating Procedure, OP-149, "Fire Protection," Section 4.0. HNP, AOP-041, "Spent Fuel Pool Events," has a contingency for cooling the spent fuel pool (SFP) heat exchangers with fire protection water, in the event that normal cooling has failed and the SFP temperature is rising. Fire water is identified in the, "Severe Accident Mitigation Strategies," as a makeup source that can be used if needed for such activities as injecting into the steam generators (SG). The licensee stated that fire water is also identified as an alternate SFP makeup and cooling water source in the incident stabilization guidelines used for implementation of HNP B.5.b mitigation strategies. A B.5.b event is a beyond design basis loss of a large area of a reactor plant due to fires or explosions initiated by a terrorist threat. The licensee determined that the MUR power uprate has no impact on the probability of occurrence or the severity of these incidents and therefore, the MUR power uprate has no adverse impact on these "non-fire protection" aspects of the fire protection system.

The NRC staff finds the licensee's response addressed the staff's concerns, since the licensee verified that HNP does not credit the fire water supply for non-fire protection. The licensee also identified the following other uses of the fire protection system for non-fire protection functions: provide water to the SFP heat exchanger; inject water to the SG, and use of fire water for HNP B.5.b mitigation strategies. The staff finds the licensee's response to the request for additional information (RAI) acceptable because: (1) HNP does not credit the fire protection system to support the design basis for non-fire protection functions, and (2) any non-fire protection uses of the system would not adversely impact the fire protection systems.

Based on the licensee's nuclear safety performance criteria assessment and responses to the RAIs, the NRC staff finds this aspect of the capability of the associated fire protection SSCs to perform their design basis functions at an increased core power level of 2948 MWt acceptable with respect to fire protection.

Conclusion

The NRC staff has reviewed the licensee's fire-related safe-shutdown assessment and concludes that the licensee has adequately accounted for the effects of the increased decay heat on the ability of the required systems to achieve and maintain post-fire safe-shutdown conditions. The NRC staff further concludes that the proposed MUR power uprate will not have a significant impact on the fire protection program or nuclear safety performance criteria of NFPA 805.

3.4 Steam Generator Tube Integrity and Chemical Engineering

3.4.1 Chemical and Volume Control System (CVCS)

Regulatory Evaluation

The CVCS provides a means for 1) maintaining water inventory and quality in the RCS, 2) supplying seal-water flow to the RCPs and pressurizer auxiliary spray, 3) controlling the boron neutron absorber concentration in the reactor coolant, 4) controlling the primary-water chemistry and reducing coolant radioactivity level, and 5) supplying recycled coolant for demineralized water makeup for normal operation and high-pressure injection flow to the ECCS in the event of postulated accidents. The NRC staff has reviewed the existing AOR to determine whether continued operation at the proposed uprate power level is acceptable without the need for reanalysis.

The NRC's acceptance criteria for the CVCS is based on (1) GDC 14, "Reactor Coolant Pressure Boundary" (RCPB), of Appendix A to 10 CFR Part 50, as it requires that the RCPB be designed to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture; and (2) GDC 29, "Protection Against Anticipated Operational Occurrences," as it requires that the reactivity control systems be designed to assure an extremely high probability of accomplishing their functions in the event of condenser in-leakage or primary-to-secondary leakage. Specific review criteria are contained in SRP Section 9.3.4, "Chemical and Volume Control System [Pressurized-Water Reactor] (PWR)."

Technical Evaluation

The primary function of the CVCS is to maintain RCS water inventory, boron concentration, and water chemistry. In addition, the CVCS provides for boric acid addition and removal, chemical additions for corrosion control, reactor coolant cleanup and degasification, reactor coolant makeup, and processing of reactor coolant letdown.

During plant operation, reactor coolant letdown is taken from the cold leg on the suction side of the RCP, through the tube side of the regenerative heat exchanger and then through letdown control valves. The regenerative heat exchanger reduces the temperature of the reactor coolant and the control valves reduce the pressure. The letdown is cooled further in the tube side of the letdown heat exchanger and subsequently passes through the purification filter. Flow continues through the purification ion exchangers, where ionic impurities are removed, and enters the volume control tank (VCT). The charging pumps take suction from the VCT and return the coolant through the shell side of the regenerative heat exchanger to the RCS in the cold leg, downstream of the RCP.

The licensee stated, in its LAR dated April 28, 2011, that accidents, transients and other FSAR analyses were reviewed to determine the impact of the MUR on the CVCS. The licensee reported that the hot-leg and cold-leg temperatures of the RCS is predicted to increase and decrease by 0.6 degrees Fahrenheit (°F) to 623.8 °F and 553.8 °F, respectively. The RCS pressure and average temperature are indicated to stay the same at 2250 pounds per square inch atmosphere (psia) and 588.8 °F. The licensee evaluated the effects of the MUR on the CVCS and determined that the CVCS will continue to satisfy the design basis requirements when considering the temperature, pressure and flow rate effects resulting from the power uprate.

The licensee stated that an evaluation was performed on the CVCS malfunction that results in a decrease in boron concentration in the reactor coolant. This event is analyzed primarily to assess the challenge to the time-to-criticality criteria. The licensee stated that this accident event is bounded by the uncontrolled RCCA bank withdrawal at power event with respect to departure from nucleate boiling ratio and fuel centerline melt criteria, found in FSAR Section 15.4.2. The AOR for the RCCA bank withdrawal event assume a reactor power level of 2958 MWt. This power level is 102 and 100.3 percent of 2900 and 2948 MWt, which are the rated thermal power level and the proposed rated thermal power level for HNP, respectively. The AOR for the uncontrolled RCCA bank withdrawal at power event are bounding of MUR power uprate conditions.

In addition, the licensee stated that an evaluation was performed for the CVCS malfunction that increases reactor coolant inventory event. The licensee indicated that this event is bounded by the CVCS malfunction that results in a decrease in the boron concentration and by the inadvertent operation of the ECCS during power operation events, which are found in FSAR Section 15.4.6 "Chemical and Volume Control System Malfunction that Results in a Decrease in the boron Concentration in the Reactor Coolant" and Section 15.5.1 "Inadvertent Operation of Emergency Core Cooling System During Power Operation," respectively. The licensee stated that the AOR regarding this second accident event are bounding for uprate conditions.

The licensee stated that the rupture of the CVCS letdown line accident event was also evaluated. The licensee indicated that this event is the most severe radioactivity release from a failed line that carries primary coolant outside of containment. The licensee further indicated that since the AOR for this event assumed a core power of 2958 MWt or 102 percent of 2900 MWt, it is bounding for the MUR power uprate.

The NRC staff has reviewed the licensee's evaluation and has confirmed that applicable acceptance criteria were reviewed and found to be acceptable. The licensee has demonstrated that the CVCS will continue to maintain RCS inventory and water chemistry. The NRC staff finds that the CVCS will continue to meet system design requirements and that no new design transients will be created at MUR power uprate conditions.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed MUR on the CVCS and concludes that the licensee has adequately addressed changes to the reactor coolant and its effects on the CVCS. The NRC staff further concludes that the licensee has demonstrated that the AOR for the CVCS will continue to be acceptable and meet the requirements of GDC 14 and GDC 29 following implementation of the proposed MUR. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the CVCS.

3.4.2 Steam Generator (SG) Program

Regulatory Evaluation

SG tubes constitute a significant part of the RCPB. SG tube inservice inspection provides a means for assessing the structural and leak-tight integrity of SG tubes through periodic inspection and testing of critical areas and features of the tubes. The NRC staff reviewed the effects of changes in differential pressure, temperature, and flow rates resulting from the proposed power uprate on the design and operation of SGs. Specifically, the NRC staff evaluated whether changes to these parameters continue to be bounded by those considered in the plant design and licensing basis (i.e., the TS SG tube plugging limits).

Technical Evaluation

HNP has three Westinghouse model Delta 75 SGs. Each SG has 6,307 thermally treated Alloy 690 tubes. The tubes have an outside diameter of 0.688 inches and a nominal wall thickness of 0.040 inches. The tubes are hydraulically expanded at each end for the full depth of the tubesheet and are supported by Type 405 stainless steel tube support plates, which have trefoil shaped holes. The first eight tube rows were heat treated after bending to relieve stresses.

The licensee indicated that the current structural limit is not changed by the power uprate. RG 1.121 "Bases for Plugging Degraded PWR Steam Generator Tubes," describes an acceptable method for establishing the limiting safe tube degradation beyond which tubes found defective by inservice inspection must be removed from service. The acceptable degradation level is called the repair limit. The licensee stated that the RG 1.121 evaluation defines the structural limit for an assumed uniform thinning mode of degradation in both axial and circumferential directions. Furthermore, it was indicated that the existing analysis (i.e., WCAP-15678, Rev. 1, "Regulatory Guide 1.121, Analysis for the Shearon Harris Replacement Steam Generator," June 2001) is applicable to power uprate conditions.

Design criteria were established for maintaining tube structural integrity under the postulated DBA condition loadings in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, 1971 minimum strength properties. The licensee stated that the existing tube repair limit is unaffected by the MUR power uprate and remains valid.

The licensee reported that a thermal-hydraulic evaluation was performed at HNP and it focused on changes to secondary-side operating characteristics. The licensee evaluated performance characteristics such as steam pressure and flow, circulation ratio, bundle mix flow, heat flux, secondary-side pressure drop, moisture carryover, hydrodynamic stability, secondary-side mass and other parameters which are affected by the proposed increase in power level. The licensee stated that the evaluation concluded that the SG thermal-hydraulic operating characteristics remain acceptable for the MUR power uprate.

The licensee stated that the SGs have exhibited no indications of corrosion-related tube degradation. It was also stated that no active systematic corrosion mechanisms have been identified, and that only foreign object or loose parts wear is identified as an existing SG tube degradation mechanism. The licensee stated that SG condition monitoring and operational assessment evaluations are prepared after each SG inspection, which is performed periodically and is consistent with the HNP TSs.

The licensee indicated that potential tube degradation mechanisms such as antivibration bar wear and outside diameter stress corrosion cracking were absent in the SGs, but continue to be mechanisms included in the inspection planning. It was indicated that after the power uprate, the potential tube degradation mechanisms resulting from hypothetical localized chemistry changes at the tube surface are the various modes of outside diameter stress corrosion cracking (ODSCC). The licensee stated that based on laboratory and operating experience, and current operating and maintenance practices, the power uprate will not produce excessive degradation due to ODSCC. The licensee further stated that on the basis of temperature increase alone (RCS hot-leg temperature increases to 623.8 °F) the mechanical wear processes are unlikely to be significantly changed. In addition, it was reported that primary water stress corrosion cracking (PWSCC) has not been identified in the SGs and the incidence of it is not expected to be affected by the power uprate, since thermally-treated Alloy 690 tube material is highly resistant to PWSCC.

The licensee performed an analysis to evaluate SG tube wear (i.e., fretting) on current design basis analysis and consideration of SG secondary-side thermal-hydraulic changes resulting from the power uprate. The analysis considered fluid-elastic effects in the U-bend region and turbulence induced displacement effects in the straight leg tube region. The fluid-elastic stability ratio is reported to increase by as much 3.4 percent to a ratio of 0.42, and the vibration amplitude due to turbulence is reported to increase by 6.9 percent to amplitude of 17.4 mils (a unit of measurement in the English system that is measured in thousandths of an inch), under the proposed MUR power uprate. The projected stability ratio will remain less than the 1.0 allowable and the vibration amplitude increase of 17.4 mils will remain less than one-half the distance separating the tubes (146 mils). The NRC staff finds the licensee's analysis acceptable.

The licensee reported that the power uprate increases the tube wear by 6.9 percent over the calculated original design power level. The licensee indicated that the post-uprate SG tube wear is predicted to increase to 4.9 mils over the projected 60-year plant life. Although the tube wear increases by 6.9 percent, the licensee reported that the value is below the tube plugging limit of 16 mils (40 percent wear depth). The NRC staff finds this acceptable because the predicted tube wear will remain below the tube plugging limit, which is consistent with RG 1.121.

The licensee also reviewed tube stress and fatigue. The tube stress resulting from flow-induced vibration (FIV) concerns, under power uprate conditions, is reported to be less than 2.0 kilopound per square inch, and corresponds to a fatigue usage of 0.024, which is less than the 1.0 allowable. The licensee concluded that the tube stress is acceptable under MUR conditions.

The licensee stated that the increase in tube wear will not significantly affect tube integrity. In addition, tube stresses and the FIV loading fatigue usage factor are stated to be acceptable and have negligible effects under MUR power uprate conditions. The licensee also stated that fatigue degradation from flow induced vibration is not anticipated. The NRC staff has reviewed the FIV licensee's evaluation and finds it acceptable.

The NRC staff has reviewed the licensee's evaluation and has confirmed that the applicable regulatory guidance was followed. The NRC staff, therefore, finds that the proposed MUR power uprate will introduce only insignificant changes as it relates to tube wear due to potential tube degradation mechanisms, which will not affect satisfactory performance in maintaining SG tube integrity.

Conclusion

The NRC staff reviewed the licensee's evaluation of the effect of the proposed MUR power uprate on SG tube integrity and concludes that the licensee has adequately assessed the continued acceptability of the plant's TSs. Specifically, the licensee has an ongoing inspection program, including periodic degradation assessments, that will identify any increased degradation and any need for additional or enhanced inspections prior to tube integrity being challenged.

The NRC staff concludes that the power uprate is acceptable because the licensee's evaluation of the thermal-hydraulic performance, their structural evaluation, and their FIV analysis have shown that the MUR power uprate is expected to introduce only negligible changes in the SG parameters, which will not significantly affect the performance of the SGs, and it will continue to operate within its design limits under uprate conditions. Therefore, the staff finds the proposed MUR power uprate acceptable with respect to the SG program.

3.4.3 Flow Accelerated Corrosion (FAC)

Regulatory Evaluation

FAC is a corrosion mechanism occurring in carbon steel components exposed to single-phase or two-phase water flow. Components made from stainless steel are immune to FAC, and FAC is significantly reduced in components containing even small amounts of chromium or molybdenum. The rates of material loss due to FAC depend on flow velocity, component geometry, fluid temperature, steam quality, oxygen content, and potential of hydrogen (pH). During plant operation, it is not normally possible to maintain all of these parameters in a regime that minimizes FAC; therefore, loss of material by FAC can occur. The NRC staff reviewed the effects of the proposed MUR power uprate on FAC and the adequacy of the licensee's FAC program to predict the rate of material loss so that repair or replacement of damaged components could be made before reaching a critical thickness.

The licensee's FAC program is established and maintained per Generic Letter (GL) 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," dated May 2, 1989 (ADAMS Accession No. ML031200731). The licensee performs inspections and analyses to determine the number of refueling or operating cycles remaining before components reach minimum allowable wall thickness. If an analysis indicates that an area will reach the minimum allowed wall thickness before the next scheduled outage, corrective action should be considered. The NRC's acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

Technical Evaluation

The licensee stated that the HNP FAC program is based on the guidelines found in the Electric Power Research Institute's (EPRI) Report NSAC-202L-R3, "Recommendation for an Effective Flow-accelerated Corrosion Program," dated May 2006 (available from EPRI). The program provides a standardized method of identifying, inspecting, and evaluating piping systems susceptible to FAC. The licensee stated that HNP uses the CHECWORKS Steam/Feedwater Application (SFA) FAC monitoring computer code to model the thermal dynamic conditions in the secondary side high energy piping systems to predict and track FAC susceptible components.

It was indicated that the CHECWORKS SFA model was updated to incorporate the changes associated with the MUR power uprate. An analysis was performed to calculate the change in CHECWORKS predicted wear. The licensee stated that some FAC susceptible lines (i.e., systems and components) are predicted to have a decrease in wear. The licensee provided in the application Table IV-3, "Wear Rate Analysis," which provides the system lines that are anticipated to have increased wear. According to Table IV-3 the line that is predicted to have the maximum increase in wear of 8.2 percent is located in the reheater drain to heater 5. The NRC staff finds the corrosion rate increases reasonable for the corresponding changes in operating conditions. The NRC staff finds that the CHECWORKS SFA model provides reasonable assurance that the program will continue to be an acceptable predictive model after the implementation of the power uprate.

It was reported that no additional secondary-side piping has been identified as requiring monitoring under the existing FAC program. Furthermore, the licensee stated that the modeling and analysis demonstrated that the MUR power uprate impact on FAC wear rates is not significant and that the predicted remaining service life is essentially unchanged. The licensee indicated that the existing FAC program bounds all FAC susceptible piping, and changes to current inspection scope and frequency are not necessary.

The NRC staff has reviewed the licensee's evaluation and has determined that the applicable regulatory guidance was followed. The licensee has demonstrated that the FAC program is adequate for managing the potential effects on the piping components susceptible to FAC. The NRC staff therefore, finds that the FAC program is adequate in predicting the rate of material loss.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the proposed MUR power uprate on the FAC analysis for HNP and concludes that the licensee has adequately addressed the impact of changes in plant operating conditions. Additionally, the NRC staff concludes that the licensee has demonstrated the existing analyses will predict the loss of material by FAC and will ensure timely repair or replacement of degraded components following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the FAC program.

3.4.4 Steam Generator Blowdown System (SGBS)

Regulatory Evaluation

Control of secondary-side water chemistry is important for preventing degradation of SG tubes. The SGBS provides a means for removing SG secondary-side impurities and, thus, assists in maintaining acceptable secondary-side water chemistry in the SGs. The design basis of the SGBS includes consideration of expected and design flows for all modes of operation. The NRC staff reviewed the existing AOR to determine whether continued operation at the proposed uprate power level is acceptable without the need for re analysis. The NRC's acceptance criteria for the SGBS is based on 10 CFR Part 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," as it requires that the RCPB be designed so as to have an extremely low probability of abnormal leakage, rapidly propagating fracture, and gross rupture. Specific review criteria are contained in SRP Section 10.4.8, "Steam Generator Blowdown System (PWR)."

Technical Evaluation

The SGBS is designed to extract water containing particulates and dissolved solids from the secondary side of the SGs, as a means of controlling SG water chemistry. The water collected from the SG is piped to the blowdown tank, which is vented to the atmosphere and drains to the service water (SW) system. The SGBS also provides a means for sampling the secondary side water in the SG. These samples are used for monitoring water chemistry and for detecting the amount of radioactive primary coolant leakage through the SG tubes. Proper control of SG secondary-side chemistry reduces the probability of secondary-side-initiated SG tube degradation.

The licensee stated that the required SGBS flow rates during plant operation are based on chemical control and tubesheet sweep necessary to control solids buildup. It was indicated that the volumetric flow rate will not increase at power uprate conditions. As such, associated system flow velocities will not increase. Although no increase in flow rate is expected, the licensee stated that SGBS operating temperatures and pressures will decrease and remain bounded by the existing design parameters. The licensee stated that the decrease in pressure, approximately 7 pounds per square inch (psi), may cause SGBS flow control valves to open slightly to accommodate the same flow rate into the flash tank. The licensee also stated that since the system velocities are not increasing, the wear rate due to FAC does not increase. The licensee concludes that the SGBS will continue to meet system design requirements at MUR power uprate conditions. The NRC staff finds the licensee's analysis acceptable because the current AOR is bound power uprate conditions.

The NRC staff has reviewed the licensee's evaluation and has confirmed that the applicable regulatory guidance was followed. The licensee has demonstrated that the SGBS is adequate for maintaining secondary-side water chemistry within industry guidelines to maintain and control corrosion rates in secondary system components. The NRC staff concurs that the SGBS will continue to meet system design requirements at MUR power uprate conditions.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed MUR power uprate on the SGBS and concludes that the licensee has adequately addressed the affects of the MUR power uprate on system flow and impurity levels. The NRC staff further concludes that the licensee has demonstrated that the SGBS will continue to be acceptable and will continue to meet the requirements of 10 CFR Part 50, GDC 14, following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the SGBS.

3.4.5 Protective Coating Systems

Regulatory Evaluation

Protective coatings (paints) inside containment are used to protect equipment and structures from corrosion and contamination from radionuclides and also provide wear protection during plant operation and maintenance activities. The NRC's staff acceptance criteria for coatings are subject to 10 CFR Part 50, Appendix B "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The NRC staff reviewed whether the pressure, radiation and temperature conditions under the MUR power uprate continue to be bounded by the conditions to which the coatings were qualified.

Technical Evaluation

The licensee conducted a review to determine the power uprate effect on the protective coatings (Service Level 1 coating) used inside containment and suitability and stability under design basis LOCA (DBLOCA) conditions. Failure of the coatings under DBALOCA conditions may result in unaccounted for debris that could challenge the ECCS suction strainers. The review considered containment pressure and temperature (P-T), radiation levels, and boric acid concentrations. The licensee stated that the containment analyses are unaffected and the analyzed post-LOCA DBA containment peak P-T transients remain valid and bounding at MUR uprate conditions for coatings qualification. In addition, it was indicated that the current containment dose estimates remain applicable. As such, the radiation level analysis conclusions remain valid for the containment coatings. The licensee also stated that the power uprate does not affect the existing containment spray/sump water pH; therefore, the uprate conditions are bounded by the currently analyzed pH range for Service Level 1 coatings.

The licensee stated that the post-LOCA containment P-T, integrated radiation dose, and pH range values are bounded by the data used to qualify the Service Level 1 containment coatings. The licensee concluded that the Service Level 1 containment coatings remain qualified under MUR power uprate conditions. The NRC staff finds this acceptable.

The NRC staff has reviewed the licensee's evaluation and finds that the applicable regulatory guidance was followed. The NRC staff concurs that the coatings will not be adversely impacted by the MUR power uprate and that temperature, pressure, and radiation limits under power uprate conditions continue to be bounded by the conditions to which the coatings were qualified.

Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed MUR power uprate on protective coating systems and concludes that the licensee has appropriately addressed the impact of changes in conditions following a DBLOCA and their effects on the protective coatings. The NRC staff further concludes that the licensee has demonstrated that the current protective coatings will continue to be acceptable following implementation of the proposed MUR power uprate and will meet the requirements of 10 CFR Part 50, Appendix B. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to protective coatings systems.

3.4.6 Steam Generator Tube Integrity and Chemical Engineering Conclusion

In the areas of SGs and chemical engineering, the NRC staff concludes that the licensee has adequately addressed (1) the changes to the reactor coolant and their effect on the CVCS, (2) the changes in the SG operating parameters, the effects on the SGs and the determination that the SG tube integrity will continue to be maintained, (3) the changes in the plant operating conditions for the FAC program, (4) the changes in the system flow and impurity levels, and their effects on the SGBS and the (5) the effects on protective coatings.

3.5 Safety-Related Valves, and Inservice Test Program

3.5.1 Safety-Related Valves

Regulatory Evaluation

The NRC staff reviewed the licensee's safety-related valves analysis for HNP. The NRC staff examined the overall design change and included plant-specific evaluations of GL 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," and GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions."

Technical Evaluation

In Section IV.1.Aix, "Safety-Related Valves" of the submittal, the licensee described the impact of the proposed MUR power uprate on the existing safety-related valves DBA. The licensee stated that no changes in RCS flow, design, or operating pressure were made as part of the power uprate. Evaluations concluded that the temperature changes due to the power uprate are bounded by those used in the existing analyses. As a result, none of the safety-related valves required a change to their design or operation as a result of the MUR power uprate. The analyses also confirmed that the existing main steam (MS) safety valves' capacity is adequate for overpressure protection at MUR power uprate conditions and that the existing lift setpoints are unchanged. Due to the insignificant changes in temperature and operating pressure, none of the safety-related valves required a change to their design or operation as a result of the MUR power uprate.

The licensee also evaluated in Section VII.6.E.i "GL 89-10 and GL 96-05 Motor Operated Valve Program" of the submittal the impact of the proposed MUR power uprate on the current air-operated valve (AOV) program, GL 89-10 and GL 96-05 motor-operated valve (MOV) program, and GL 95-07 pressure locking/thermal binding (PLTB) program. The overall system evaluations concluded that valve function, valve design, operational conditions, thrust, and torque requirements are unaffected by the MUR power uprate and all valves remain capable of performing their design basis functions. Therefore, no changes are required to the existing AOV, MOV, and PLTB programs. Based on the NRC staff review of the licensee's evaluations and analyses, the NRC staff concluded that the performance of existing safety-related valves is acceptable with respect to the MUR power uprate.

Conclusion

Based on the above, the NRC staff concludes that with respect to the safety-related valves, the current plant design is considered adequate and would require no modifications to the design or operation for the MUR power uprate conditions.

3.5.2 Inservice Test (IST) Program

Regulatory Evaluation

For the IST program, for safety-related pumps and valves, the HNP code of record is the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) 2001 Edition through the 2003 Addenda.

Technical Evaluation

The licensee described in Section IV.1E.i "Inservice Testing Program" of the submittal the IST program for safety-related pumps and valves at HNP during the MUR power uprate operation. The IST program at HNP assesses the operational readiness of pumps and valves within the scope of the ASME OM Code. There were no significant changes to operating conditions or the design basis requirements that would affect component performance, test acceptance criteria, or reference values. Therefore, the existing IST program will not be impacted by the MUR power uprate. Based on the licensee's evaluation, the NRC staff determines that the existing IST program will be acceptable at the MUR power uprate conditions.

Conclusion

Based on the above discussions, the NRC staff concludes that the current IST program is acceptable for the current plant design and would require no modifications to the design or operation for the MUR power uprate conditions.

3.6 Reactor Systems LEFM [Leading Edge Flow Meter] Analysis

Regulatory Evaluation

As described in the background section of this SE, MUR power uprates may be authorized by the NRC staff based on the current wording of 10 CFR 50.46, and Appendix K to 10 CFR

Part 50, provided that the licensee has demonstrated that the proposed instrumentation adequately accounts for instrument uncertainties. In this case, the licensee has referred to NRC-approved Cameron Topical Reports ER-80P[5] Revision 0 and ER-157P[6] Revision 8 to provide this justification. The NRC staff reviewed thermal-hydraulic aspects of the LEFM CheckPlus system installation, including its laboratory calibration, and the effects of system changes such as transducer replacement.

Technical Evaluation

3.6.1 Background

The originally installed instruments for measuring FW flow rate in existing nuclear power plants were usually a venturi or a flow nozzle, each of which generates a differential pressure proportional to the FW velocity in the pipe. Of the two, the venturi was the most widely used because of relatively low head loss. However, error in the determination of flow rate is introduced due to venturi fouling and, to a lesser extent, flow nozzle fouling, the transmitter, and the analog-to-digital converter. "Venturi" will generally be used in the remainder of this document to reference both venturis and flow nozzles.

Because of the desire to reduce flow instrumentation uncertainty to enable operation of the plant at a higher power while remaining within the licensed rating, the industry assessed alternate flow rate measurement techniques and found that UFMs are a viable alternative. UFMs are based on computer-controlled electronic transducers that do not have differential pressure elements that are susceptible to fouling. Caldon, Inc., which is part of Cameron Measurement Systems, developed a UFM called a "leading edge flow meter" and named it the LEFM Check system. It was followed with the LEFM CheckPlus System, which provides a more accurate FW flow measurement than the Check system. Both of these UFMs have demonstrated better measurement accuracies than the differential pressure type instruments and provide on-line verification to ensure that the UFM is operating within its uncertainty bounds.

Caldon submitted an engineering report, ER-80P, in March 1997 that describes the LEFM, includes calculations of power measurement uncertainty using a Check system in a typical two-loop PWR or a two-FW-line boiling-water reactor, and provides guidance for determining plant-specific power calorimetric uncertainties. The NRC staff approved this report that allowed for an uncertainty less than 2 percent for the requirements of Appendix K to 10 CFR Part 50 and approved a 1 percent power uprate for using the LEFM.[7] Following publication of the amendment to Appendix K that allowed for an uncertainty less than 2 percent, Caldon submitted a supplement to ER-80P, ER-160P[8] that the NRC staff approved on January 19, 2001,[9] for up to

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5 Caldon Engineering Report ER-80P, Revision 0, Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System, March 1997.

6 Cameron Engineering Report ER-157P, Revision 8, Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System, Caldon Inc., May 2008.

7 Letter from Project Directorate IV-1, Division of Licensing Project Management, Office of Nuclear Reactor Regulation, to C.L. Terry (TU Electric), Comanche Peak Steam Electric Station, Units 1 and 2 - Review of Caldon Engineering Topical Report 80P, Improving Thermal Power Accuracy and Plant Safety While Increasing Power Level Using the LEFM System (Accession Number ML9903190065 legacy library), March 8, 1999.

8 Caldon Inc., "Supplement to Engineering Report ER-80P: Basis for a Power Uprate with the LEFM√™ System," ER-160P, Revision 0, May 2000.

a 1.4 percent power uprate. Subsequently, the NRC staff approved ER-157P, Revision 5,[10] for up to a 1.7 percent power uprate using the CheckPlus System[11] and recently approved ER-157P, Revision 8.[12] Revision 8 corrects minor errors in Revision 5, provides clarifying text, and incorporates revised analyses of coherent noise, non-fluid delays, and transducer replacement.

3.6.2 Feedwater Flow Measurement Device

HNP, was originally designed with FW flow and temperature instrumentation consisting of venturis, differential pressure transmitters, and thermocouples. Modifications required for the MUR power uprate include installation of the CheckPlus system. Existing FW flow and temperature instrumentation will be retained and used for comparison monitoring of the LEFM system and as a backup FW flow measurement when needed.

The FW flow measurement system to be permanently installed in HNP is a Cameron LEFM CheckPlus ultrasonic 8-path transit time flowmeter. The LEFM CheckPlus System provides on-line main FW flow and temperature measurement to determine reactor thermal power. The system uses acoustic energy pulses to determine the main FW mass flow rate and temperature. The LEFM consists of a measuring section containing 16 ultrasonic multi-path transit time transducers divided into two planes of eight, one dual resistance temperature detector (RTD) and two pressure transmitters installed in each of the three main FW lines, and an electronic signal processing cabinet. The electronic cabinet is located in the Turbine Building. The measurement spool pieces are installed in each of the three main SG FW flow header lines. Spool pieces in the A and B FW lines are installed well downstream of the existing venturis. The spool piece in the C FW line is installed upstream of the venturi.

The calibration and accuracy assessment testing of the LEFM CheckPlus System was performed at Alden Research Laboratory (ARL) using current plant configuration and variations of the plant configuration. The calibration testing determined the meter calibration constant, or meter factor. The FW piping configurations are explicitly modeled as part of the CheckPlus meter factor.

3.6.3 Transducer Installation

The CheckPlus provides an array of 16 ultrasonic transducers installed in a spool piece to determine average velocity in 8 paths. The transducers are arranged in fixtures such that they

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9 Martin, Robert, "Staff Acceptance of TS Changes, Power Uprate Request, and Caldon Engineering Report ER-160P," NRC letter to J.A. Scalice, Tennessee Valley Authority, January 19, 2001.

10 Caldon Inc., "Supplement to Engineering Report ER-80P: Basis for a Power Uprate with the LEFM√™ or LEFM CheckPlus™ System," ER-157P, Revision 5, October 2001.

11 Richards, Stuart A., "Review of Caldon, Inc., Engineering Report ER-157P," NRC letter to Michael A. Krupa, Entergy, December 20, 2001.

12 Letter from Thomas B. Blount (USNRC) to Ernest Hauser (Cameron), Final Safety Evaluation For Cameron Measurement Systems Engineering Report ER-157P, Revision 8, Caldon Ultrasonics Engineering Report ER-15 7P, Supplement to Topical Report ER-80P: Basis For A Power Uprate With The LEFM Check Or CheckPlus System, ML102160663, August 16, 2010.

form parallel and precisely defined acoustic paths. The chordal placement is intended to provide an accurate numerical integration of the axial flow velocity along the chordal paths. The planned installation location of each CheckPlus as stated in the submittal conforms to the applicable requirements in Cameron's Installation and Commissioning Manual and Cameron engineering reports ER-80P, ER-160P and ER-157P, which have been approved by the NRC.

The licensee stated in the application that the effect for uncertainty in the transducer installation of the CheckPlus system is identified in the Caldon Customer Information Bulletin CIB-125, Revision 0. The uncertainty calculations for the transducer installation are documented in the Cameron Engineering Report ER-69713 Rev. 2, which is a Cameron proprietary document. These system uncertainties incorporate an additional transducer variability uncertainty in both the profile factor uncertainty and in the installation uncertainty.

In Footnote Reference 14 the licensee showed that LEFM commissioning will include verification of ultrasonic signal quality and evaluation of actual plant hydraulic flow profiles as compared to those documented during the ARL testing. These parameters were incorporated as required during the LEFM commissioning and calibration process completed in November 2010. Since the uncertainty of the LEFM CheckPlus installation is incorporated in the uncertainty calculations, the NRC finds that the transducer installation variability has been acceptably addressed.

3.6.4 CheckPlus Calibration

The bounding calibration factor acceptability for the spool pieces was established by tests at the ARL and is addressed in ER-720 Rev. 2.[15] ER-720 provides test configuration drawings and NRC staff audited HNP piping and instrumentation diagrams that show the CheckPlus installation locations. Distances between the exit of the CheckPlus spool pieces and the downstream elbows in the tests need to be greater than 6-½ feet. As discussed below, this separation distance is large enough that there will be no effect on UFM calibration.

Loop A was tested with an upstream distance from the UFM to the venturi that is slightly larger than installed in the plant. Loop B was tested with an upstream distance from the UFM to the venturi that is shorter than that of its installation in the plant by a few feet. Loop C was tested with an upstream distance from the UFM to the elbow centerline that is a few inches shorter than installed in the plant. All loops were tested with greater than 8 feet of straight pipe upstream of the UFM. Although it is desirable to have close correspondence between test and plant geometries, these upstream differences are not anticipated to significantly affect the calibrations because they will be bracketed by the effect of geometry variations introduced during the tests. Further, the CheckPlus capability to address changes in flow profile will provide any needed calibration correction as a result of in-plant testing.

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13 Cameron Ultrasonics Engineering Report ER-697, Revision 2, "Bounding Uncertainty Analysis for Thermal Power Determination at Harris Unit 1 Using the LEFM CheckPlus System," January 2011.

14 Markowski, David, "Field Commissioning Data Package for Progress Energy Service Company, LLC, Shearon Harris Unit 1 Nuclear power Station," December 2010.

15 Augenstein, Don, "Meter Factor Calculation and Accuracy Assessment for Harris Nuclear Plant," Proprietary, Cameron Measurement Systems, ER-720, Revision 2, contained in Attachment 6 to ML11124A180, January 2011.

3.6.5 Evaluation of the Effect of Downstream Piping Configurations on Calibration

Turbulent flow regimes that exist when the plant is near full power result in limited upstream flow profile perturbation from downstream piping. Consequently, the effects of downstream equipment need not be considered for normal CheckPlus operation provided changes in downstream piping, such as the entrance to an elbow, are located greater than two pipe diameters downstream of the chordal paths. However, if the CheckPlus is operated with one or more transducers out of service, the acceptable separation distance is a function of the transducer to elbow orientation. In such cases, if separation distance is less than five pipe diameters, it should be addressed.

As discussed above, separation from downstream components is needed so that CheckPlus operation will not be affected. In HNP, the separation is greater than 9 feet and downstream piping components such as elbows and venturis will not affect the CheckPlus operation.

3.6.6 Evaluation of Upstream Flow Straighteners on CheckPlus Calibration

Operation with an upstream flow straightener is known to affect CheckPlus calibration to a greater extent than most other upstream hardware. A previously undocumented effect of upstream tubular flow straighteners on CheckPlus calibration was discovered during ARL testing that did not appear to apply to any previous CheckPlus installations. As followup, additional tests were conducted with several flow straighteners and two different pipe and spool piece diameters to enhance the statistical data basis and to develop an understanding of the interaction between flow straighteners and the CheckPlus. The results are provided in the proprietary version of ER-790 Rev. 1.[16]

Cameron concluded that two additional meter factor uncertainty elements are necessary if a CheckPlus is installed downstream of a tubular flow straightener and provided uncertainty values are derived from the test results. The data also provide insights into the unique flow profile characteristics downstream of tubular flow straighteners and a qualitative understanding of why the flow profile perturbations may affect the CheckPlus calibration.

Cameron determined that the two uncertainty elements are uncorrelated and, therefore, combined them as the root sum squared to provide a quantitative uncertainty. The Cameron approach is judged to be valid, but there is concern that the characteristics of existing tubular flow straighteners in power plants may not be adequately represented by samples tested in the laboratory. Any applicant that requests an MUR with the configuration discussed above should provide justification for claimed CheckPlus uncertainty that extends the justification provided in ER-790 Rev. 1. No flow straighteners are installed in the applicant's FW lines and flow straightener effects are not a concern.

3.6.7 In-Plant Operation

Many of the calibration aspects associated with transfer from a test facility to the plant apply

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16 Estrada, Herb, Engineering Report: ER-790 Rev. 1, "An Evaluation of the Impact of 55 Tube Permutit Flow Conditioners on the Meter Factor of an LEFM CheckPlus," Proprietary, Cameron Measurement Systems, ML100840026, March 2010. The results do not apply to the Check UFM. Consequently, the findings do not apply to a Check that is installed downstream of a tubular flow straightener.

during operation as valve positions change, different pumps are operated, and physical changes occur in the plant. The latter include such items as temperature changes, preheater alignment and characteristics changes, pipe erosion, pump wear, crud buildup and loss, and valve wear. Further, potential UFM changes, such as transducer degradation or failure, may also occur and the UFM should be capable of responding to such behavior. Either the UFM must remain within calibration and traceability must continue to exist during such changes or the UFM must clearly identify that calibration and traceability are no longer within acceptable parameters. Experience is that the CheckPlus is capable of handling these operational aspects. Further, as stated above, UFM operation should be cross-checked with other plant parameters that are related to FW flow rate and the UFM must be considered inoperable if its calibration is no longer established to be within acceptable limits.

Section I.1 of the LAR provides coverage of training, calibration, maintenance, procedures, entry into the corrective action program, and procedures to ensure compliance with the requirements of 10 CFR Appendix B. Therefore, the NRC staff finds the licensee's evaluation acceptable.

3.6.8 CheckPlus Operation with a Failed Component

ER-157P-A Rev. 8 states that "The redundancy inherent in the two measurement planes of an LEFM CheckPlus also makes this system more resistant to component failures" when compared to the LEFM Check system. "For any single component failure, continued operation at a power greater than that prior to the uprate can be justified with a CheckPlus system ... since the system with the failure is no less than an LEFM Check." This is acceptable subject to two qualifications:

(1) Continued operation at the pre-failure power level for a pre-determined time and the decrease in power that must occur following that time are plant specific and must be acceptably justified.

(2) The only mechanical difference that potentially affects the quoted statement is that the CheckPlus has 16 transducer housings interfacing with the flowing water whereas the LEFM Check has 8. Consequently, a CheckPlus operating with a single failure is not identical to an LEFM Check. Although the effect on hydraulic behavior is expected to be negligible, this must be acceptably quantified if an applicant wishes to operate as stated. An acceptable quantification method is to establish the effect in an acceptable test configuration such as can be accomplished at ARL.

Sections I.1.G and I.1.H of the application addresses allowed outage time (AOT), monitoring of CheckPlus status, and operational processes associated with a degraded or non-operational CheckPlus. The difference between a degraded CheckPlus and a Check is covered by the ARL test results.

To operate above the CLTP of 2900 MWt, the licensee proposes to use the Cameron LEFM CheckPlus System in the normal and the maintenance modes. If the UFM is not functional, the input for the calorimetric will not use the Cameron LEFM data. Instead, the input will revert to the original source from the venturis.

In the normal mode of operation, both planes of transducers are in service and system operations are processed by both central processing units (CPUs). The LEFM reading will be used as input for the FW flow in the calorimetric.

The maintenance mode of operation is defined as a time when there is a failure involving a transducer, failure of one plane of operation or if a CPU related malfunction occurs, in this case, the system reverts to the LEFM Check system.

In the nonfunctional mode, when neither normal mode nor maintenance mode exist, the licensee will have 72-hours to return the system to normal status or maintenance mode status any time the plant is operating above 2900 MWt. Further, if a power transient occurs during the 72-hour period such that the plant experiences a power decrease below 2900 MWt, the maximum permitted power level will be 2900 MWt. Stated differently, in non-functional mode the plant will be operated as though the CheckPlus was never installed and the power uprate was not in effect. These actions are to be covered in the LEFM operability requirements contained in the HNP Relocated TS and Design Basis Requirements limiting condition of operation. The NRC staff finds that operation with an inoperable (non-functional) CheckPlus and a planned operation with a with a failed CheckPlus component have been acceptably addressed.

The FW LEFM calorimetric requirements will be contained in the HNP procedure PLP-114 "Relocated Technical Specifications and Design Basis Requirements." The licensee made a commitment in the April 28, 2011, submittal to revise the PLP-114 procedure to include the LEFM calorimetric requirements.

3.6.9.1 Accident Analyses Bounded by Current Analysis of Record

Although the licensee generally concluded that existing analyses were bounding of uprated plant operation with reduced uncertainty, the analyses were shown to be bounding in one of three different ways:

- For analyses that assume steady-state plant operation with a core power of 2900 MWt, the licensee evaluated accident or transient, and reanalyzed as necessary.
- Zero-power transients were not reanalyzed.
- The licensing basis transients and accidents are summarized in Table 1 "Accident and Transient Analysis."

For analyses that assume steady-state plant operation with a core power of 2958 MWt, there is a 2 percent margin for power measurement uncertainty at the CLTP, 2900 MWt. These analyses are bounding also of plant operation at the MUR RTP of 2948 MWt, with an operating margin of 0.339 percent, which is greater than the stated 0.336 percent calorimetric power measurement uncertainty.

RIS 2002-03, states that:

> *In areas (e.g., accident/transient analyses, components, systems) for which the existing analyses of record do bound plant operation at the proposed uprated power level, the staff will not conduct a detailed review.*

The NRC staff, therefore, finds the licensee's analyses that were performed at 102 percent of the CLTP level acceptable without detailed review. In the sections after Table 1, discussion is provided for those analyses that were not performed at 102 percent of the original licensed thermal power. The licensee found that there are no accidents or transients where the existing AOR do not bound plant operation at the proposed uprated power level.

Table 1: Accident and Transient Analyses

Transient/Accident	Analytic Power Level (% CLTP)	Review Comments
Excessive Increase in Secondary Steam Flow	102	Acceptable
FW System Malfunctions that Results in FW Increase	102	Acceptable
Steam System Piping Failure	0 and 100	See Section 3.6.9.2
Turbine Trip	102	Acceptable
Loss of Non-Emergency AC Power to the Station Auxiliaries	102	Acceptable
Loss of Normal FW Flow	102	Acceptable
FW System Pipe Break	102	Acceptable
Complete Loss of Forced Reactor Coolant Flow	102	Acceptable
RCP Shaft Seizure (LR)	102	Acceptable
Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition	0	See Section 3.6.9.3
Uncontrolled RCCA Bank Withdrawal at Power	102	Acceptable
Dropped Full Length RCCA or RCCA Bank	102	Acceptable
Withdrawal of a Single Full Length RCCA	102	Acceptable
Statically Misaligned RCCA or Bank	102	Acceptable
Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	102	Acceptable
Spectrum of RCCA Ejection Accidents	102	Acceptable
Inadvertent Operation of the ECCS During Power Operation	102	Acceptable
Inadvertent Opening of a Pressurizer Safety or Power Operated Relief Valve	102	Acceptable
Break in Instrument or Other Line From Reactor Coolant Pressure Boundary that Penetrate Containment	102	Acceptable
Steam Generator Tube Rupture	102	Acceptable
Large Break Loss of Coolant Accident	102	Acceptable
Small Break Loss of Coolant Accident	102	Acceptable
Natural Circulation Cooldown	102	Acceptable
Long Term LOCA Mass and Energy Release	102	Acceptable
Short Term LOCA Mass and Energy Release	102	Acceptable
MS Line Break Mass and Energy Release	102	Acceptable
Station Blackout	102	Acceptable

3.6.9.2 Rupture of a Main Steam (MS) Pipe

This event is evaluated to assess the challenge to radiological dose criterion from fuel failure due to exceeding DNBR and/or fuel centerline melt limits and is analyzed at both hot full power (HFP) and hot zero power (HZP) conditions. HZP conditions were not necessary to reanalyze for purposes of the MUR because the system response for the HZP cases is unaffected.

HFP cases were initiated at the current licensed nominal power of 2900 MWt. The licensee stated that HFP cases are driven by the maximum rate of positive moderator reactivity insertion, which is predominantly a function of the largest break flow rate, which is a function of break size and the most negative moderator temperature coefficient. The maximum break size and most negative moderator temperature coefficient are unchanged by the power uprate. The licensee also stated that the initial vessel average temperature also remains unchanged for the power uprate. Also, there will be no significant change in the positive Doppler reactivity feedback at MUR power uprate conditions, and the Doppler reactivity feedback is less significant than the large positive moderator reactivity feedback. The licensee stated that since there is no significant change in the system response for the HFP cases at MUR conditions and the HZP cases are unaffected, the AOR is bounding for the MUR power uprate.

The NRC staff reviewed the disposition of the rupture of a MS line pipe analysis and concludes that there would not be a significant change in system response at HFP conditions at the uprated power and the current analysis is bounding for the MUR power uprate.

3.6.9.3 Uncontrolled Control-Rod Assembly Withdrawal from Subcritical Condition

The control rod withdrawal from subcritical is analyzed at zero-power conditions, its analysis is unaffected by the proposed power uprate. Therefore, the licensee did not reanalyze this transient. The NRC staff concludes that the zero-power transient analysis is acceptable at the proposed power uprate because it is unaffected by core power level.

3.6.10 Reactor Systems Conclusion

The NRC staff reviewed the reactor systems and thermal-hydraulic aspects of the proposed LAR in support of implementation of a MUR power uprate. Based on the considerations discussed above, the NRC staff determined that the results of the licensee's analyses related to these areas continue to meet applicable acceptance criteria following implementation of the MUR. The AOR assumed a core power of 2958 MWt or 102 percent of 2900 MWt. The proposed amendment is based on the use of a Cameron LEFM Check Plus system that would decrease the uncertainty in the FW flow, thereby decreasing the power level measurement uncertainty from 2.0 percent to 0.336 percent. In these cases, the proposed MUR rated thermal power of 2948 MWt is bounded by the current AOR.

3.7 Reactor Pressure Vessel Integrity

The NRC staff's review in the area of reactor vessel (RV) integrity focuses on the impact of the proposed MUR power uprate on the RV surveillance capsule withdrawal schedules, RV P-T limits, upper shelf energy (USE) evaluations, and pressurized thermal shock (PTS) calculations. This review was conducted, consistent with the guidance contained in RIS 2002-03, to verify that the results of licensee analyses related to these areas continue to meet the requirements of 10 CFR 50.60 and 50.61, and 10 CFR Part 50, Appendices G "Fracture Toughness Requirements" and H "Reactor Vessel Material Surveillance Program Requirements" following the implementation of the proposed MUR power uprate.

3.7.1 RV Material Surveillance Program

Regulatory Evaluation

Appendix H to 10 CFR Part 50 requires licensees to monitor changes in the fracture toughness properties of ferritic materials in the RV beltline region of light-water nuclear power reactors. The surveillance program requirements in Appendix H to 10 CFR Part 50 were established to monitor the radiation-induced changes in the mechanical and impact properties of the RV materials.

Appendix G to 10 CFR Part 50 provides requirements on the USE values used for assessing the safety margins of the reactor pressure vessel (RPV) materials against ductile tearing. Appendix G to 10 CFR Part 50 states:

> *Reactor vessel beltline materials must have Charpy upper-shelf energy in the transverse direction for base material and along the weld for weld material according to the ASME Code, of no less than 75 ft-lb initially and must maintain Charpy upper-shelf energy throughout the life of the vessel of no less than 50 ft-lb, unless it is demonstrated in a manner approved by the Director, Office of Nuclear Reactor Regulation, that lower values of Charpy upper-shelf energy will provide margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.*

Appendix H to 10 CFR Part 50 states that the design of the surveillance program and the withdrawal schedule must meet the requirements of the edition of American Standard for Testing of Materials (ASTM) E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," that is current on the issue date of the ASME Code to which the RPV was purchased. Later editions of ASTM E-185 may be used including those editions through 1982 (i.e., ASTM E-185-82). NUREG-1801 (ADAMS Accession No. ML012060521) "Generic Aging Lessons Learned Report," provides additional guidance for the surveillance program for the 60-year extended period of operation.

Technical Evaluation

The NRC approved RV surveillance capsule withdrawal schedule for HNP is based on ASTM E-185-82. Per ASTM E-185-82, the withdrawal of a capsule is to be scheduled at the nearest vessel RFO to the calculated effective full-power years (EFPY) established for the particular surveillance capsule withdrawal. The licensee also stated in the application that the surveillance capsule withdrawal schedule for HNP meets the surveillance program requirements in Appendix H of 10 CFR Part 50, which were established to monitor the radiation-induced changes in the mechanical and impact properties of the RV materials.

The licensee uses the requirements of ASTM E-185-82 as its basis for meeting the RV materials surveillance capsule withdrawal requirements of Appendix H to 10 CFR Part 50. Table 1 of ASTM E-185-82 requires that either a minimum of three, four, or five surveillance capsules be removed from each of the vessels, as based on the projected nil ductility reference temperature shift (ΔRT_{NDT}) of the limiting material at the clad-vessel interface location of the RV at the end-of-[licensed plant] life (EOL). In accordance with ASTM E-185-82, at a minimum, the number of RV materials surveillance capsules required for HNP shall consist of at least five capsules. HNP has six RV materials surveillance capsules, which are detailed in Table 2 "Summary of Surveillance Capsule Withdrawal at Harris" below.

By letter dated August 16, 2011 (ADAMS Accession No. ML11235A7303), the licensee submitted a request for revising the withdrawal schedule for the RV surveillance capsules for HNP. The purpose of the licensee's submittal was to better align the withdrawal schedule with the projection of neutron fluence at the end of life extended (EOLE) while satisfying the requirements of Appendix H to 10 CFR Part 50. Section III (B)(3) of Appendix H to 10 CFR Part 50 requires that proposed withdrawal schedules must be submitted and approved by the NRC staff prior to implementation.

The MUR power uprate conditions EOL peak base metal fluence at 55 EFPY is projected as 6.88×10^{19} n/cm^2. The projected 55 EFPY fluence resulting from the power uprate is higher than the value projected in the AOR of 6.80×10^{19} n/cm^2. The projected neutron fluence for Capsule W is 6.89×10^{19} n/cm^2 (E > 1.0 megaelectron volt (MeV)). The licensee removed the fourth capsule (Capsule W) during the fall 2010 outage when the neutron fluence on Capsule W was expected to be roughly equal to the maximum neutron fluence on the clad-vessel interface at EOL, 55 EFPY. Capsule W is currently in storage, held ready for testing or reconstitution and reinsertion into the vessel in accordance with the Harris renewed license NPF-63, Condition 2.K. The licensee will use the results of the Capsule W analysis to optimize the neutron exposure and withdrawal schedule for one of the remaining two surveillance capsules (Capsule Y and Capsule Z), so that the capsule neutron fluence will not exceed twice the maximum vessel neutron fluence per ASTM E-185-82.

The NRC staff reviewed the August 16, 2011, submittal and in the SE dated October 21, 2011 (ADAMS Accession No. ML11293A076), the NRC staff concluded that the proposed changes were consistent with applicable regulations and guidance found in Appendix H to 10 CFR Part 50; as well as ASTM Standard E185-82, and NUREG-1801, Revision 2. Table 2 contains the approved surveillance withdrawal schedule.

Table 2: Summary of Surveillance Capsule Withdrawal at Harris

ID	Withdrawal EFPY	Withdrawal Neutron Fluence (E > 1.0 MeV)
U	1	0.55×10^{19} n/cm^2
V	3	1.32×10^{19} n/cm^2
X	9	3.25×10^{19} n/cm^2
W	18	$6.8^{B} \times 10^{19}$ n/cm^2
Y	27.2^{A}	$9.39^{B} \times 10^{19}$ n/cm^2
Z	27.2^{A}	$9.39^{B} \times 10^{19}$ n/cm^2

^{A} Proposed withdrawal date for either Y or Z, but not both

^{B} Estimated fluence value

The NRC staff concludes that the licensee's current surveillance capsule withdrawal schedule for HNP MUR is acceptable with the understanding that any revised RV materials surveillance capsule withdrawal schedules must be incorporated in the appropriate sections of the HNP UFSAR in accordance with the requirements of 10 CFR 50.71(e).

3.7.2 Pressure-Temperature (P-T) Limits and Use

Regulatory Evaluation

Appendix G to 10 CFR Part 50 provides fracture toughness requirements for ferritic (low alloy steel or carbon steel) materials in the RCPB, including requirements on the USE values used for assessing the safety margins of the RV materials against ductile tearing and for calculating P-T limits for the plant. These P-T limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests.

Technical Evaluation

The NRC staff's review of the USE assessments covered the impact of the MUR power uprate on the neutron fluence values for the RV beltline materials and the USE values for the RV materials through the end of the current licensed operating period. The NRC staff's P-T limits review covered the P-T limits methodology and the calculations for the number of EFPY specified for the proposed MUR power uprate considering neutron embrittlement effects.

The current P-T Limits and low-temperature overpressurization protection (LTOP) system setpoints for HNP are approved through 36 EFPY and were based on ¼ thickness (T) and ¾ T adjusted reference temperature (ART) values of 191 °F and 179 °F, respectively at 36 EFPY for the limiting material, HNP intermediate shell plate B4197-2.

In the MUR power uprate application dated April 28, 2011, ART calculations were performed for the HNP RV materials based on updated 36 EFPY neutron fluence values. The most limiting ¼T and ¾T ART values for HNP were determined to be 190.4 °F and 177.8 °F, respectively, and were based on the evaluation of the HNP intermediate shell plate B4197-2. The NRC staff independently checked these ART values and confirmed the licensee's results. Since the ¼ T and ¾ T ART values through 36 EFPY for the HNP limiting material (intermediate-shell plate B4197-2), after consideration of the effects of the MUR power uprate remained bounded by the ¼ T and ¾ T ART values used to construct the current HNP P-T limits and to establish the current HNP LTOP system setpoint, the current P-T limits curves remain valid through their currently approved period of 36 EFPY.

The licensee reviewed the LTOP system design basis mass input and heat input transients. The critical analysis input parameters for the LTOP system study do not change for the MUR power uprate. Since the MUR power uprate neutron fluence for the HNP is bounded by the current neutron fluence at 36 EFPY, the MUR power uprate has no impact on the current P-T limit curves and LTOP system setpoint. Therefore, the NRC staff concludes that the HNP P-T limits and LTOP system setpoint would continue to meet the requirements of 10 CFR Part 50, Appendix G under the MUR power uprate condition through 36 EFPY. The licensee will, however, need to modify the period of applicability of their current P-T limits, or submit revised P-T limits, before the current curves expire.

The NRC staff's review of the USE assessment included the impact of the MUR power uprate on the neutron fluence values and USE values of the RV beltline materials through the end of the currently licensed operating period (55 EFPY). The NRC staff independently verified the licensee calculations of USE values using RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials" (ADAMS Accession No. ML003740284). The USE values calculated by the staff for the HNP RV beltline materials found that all USE values remained greater than 50 ft-lbs through 55 EFPY. The limiting USE value at 55 EFPY was 53 ft-lbs for intermediate shell plate B4179-2. Since all USE values are projected to be greater than 50 ft-lbs through 55 EFPY, the NRC staff finds that the HNP RV beltline materials will continue to satisfy the USE requirements of 10 CFR Part 50, Appendix G after consideration of the effects of the MUR power uprate.

3.7.3 Pressurized Thermal Shock (PTS)

Regulatory Evaluation

The PTS evaluation provides a means for assessing the susceptibility of pressurized PWR RV beltline materials to failure during a PTS event to assure that adequate fracture toughness exists during reactor operation. The NRC staff's requirements, methods of evaluation, and safety criteria for PTS assessments are given in 10 CFR 50.61, "Fracture toughness requirements for protection against pressure thermal shock events." The NRC staff's review covered the PTS methodology and the calculations for the reference temperature for PTS (RT_{PTS}) at the expiration of the license, considering neutron embrittlement effects.

Technical Evaluation

The PTS calculations were performed for HNP using the procedures specified in 10 CFR 50.61. Updated neutron fluence projections, corresponding to the EOL (55 EFPY) conditions, were used in the PTS analyses. RT_{PTS} calculations were performed in accordance with RG 1.99, Revision 2. For HNP, the limiting material is the intermediate shell plate B4197-2, with a projected RT_{PTS} value of 209.7°F at 55 EFPY. This is lower than the PTS screening criterion of 270°F for plates. Therefore the NRC staff concludes that the HNP RV will remain within the limits for PTS after the MUR power uprate and the RV materials will continue to meet the PTS screening criteria requirements of 10 CFR 50.61 through 55 EFPY.

3.7.4 RV Internals (RVIs) and Core Support Materials

Regulatory Evaluation

The RVIs and core support structures include SSCs that perform safety functions or whose failure could affect safety functions performed by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the RCPB). The NRC's acceptance criteria for RVIs and core support materials are based on GDC 1 "Quality Standards and Records" and 10 CFR 50.55a "Codes and standards" for material specifications, controls on welding, and inspection of RV internals and core supports. Matrix 1 of NRC RS-001, Revision 0, "Review Standard for Extended Power Uprates," provides references to the NRC's approval of the recommended guidelines for RVIs in Westinghouse Commercial Atomic Power (WCAP) Topical Reports WCAP-14577, Revision 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals" (ADAMS Accession No. ML010290348, Non-Public), and BAW-2248-A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" (ADAMS Accession No. ML003708443). These reports have been superseded by the EPRI report, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (Materials Reliability Program (MRP)-227 Rev. 0)" approved by the NRC staff on June 22, 2011 (ADAMS Accession No. ML111600498).

Technical Evaluation

The RVIs of PWR-designed light-water reactors may be susceptible to the following aging effects:

- cracking induced by thermal cycling (fatigue-induced cracking), stress corrosion cracking (SCC), or irradiation assisted stress corrosion cracking (IASCC);
- loss of fracture toughness properties induced by radiation exposure for all stainless steel grades, or the synergistic effects of radiation exposure and thermal aging for cast austenitic stainless steel (CASS) grades;
- stress relaxation in bolted, fastened, keyed or pinned RVI components induced by irradiation exposure and/or exposure to elevated temperatures; and
- void swelling (induced by radiation exposure).

Matrix 1 of NRC RS-001, Revision 0 provides the NRC staff's basis for evaluating the potential for extended power uprates to induce these aging effects. In Note 1 to Matrix 1, the NRC staff stated that guidance on the neutron irradiation-related threshold for IASCC for PWR RV internals are given in BAW-2248-A and WCAP-14577, Revision 1-A. Note 1 to Matrix 1 further stated that for thermal and neutron embrittlement of CASS, SCC, and void swelling, licensees will need to provide plant-specific degradation management programs or participate in industry programs that investigate degradation effects and determine appropriate management programs. The BAW-2248-A and WCAP-14577 reports have been superseded by the MRP-227 report.

The MRP-227 Report guidelines consider various aging factors, including neutron fluence exposure, temperature history, and representative stress levels for determining relative susceptibility of PWR internals to postulated aging mechanisms that include SCC, IASCC, wear, fatigue, thermal aging embrittlement, irradiation embrittlement, irradiation-enhanced stress relaxation and creep, and void swelling. In an RAI, the NRC staff asked the licensee to provide a description of plant-specific degradation management programs or participation in industry programs to investigate degradation effects and determine appropriate management programs. In an RAI response dated October 28, 2011, the licensee stated that "Participation is planned in the industry's initiatives on age-related degradation of pressurized water reactor vessel internals, including submittal of a plant-specific program consistent with the MRP-227 report guidelines."

The ASME Code Section XI examination programs for internals will continue to apply to other components not requiring augmented examinations. Since the HNP MUR power uprate results in very small changes to aging parameters such as temperature and neutron flux, the current aging management program for the reactor internals is acceptable.

3.7.5 Reactor Vessel Integrity Conclusion

The NRC staff has reviewed the licensee's proposed LAR to increase the rated core thermal power by 1.66 percent and has evaluated the impact that the MUR power uprate conditions will have on the structural integrity assessments for the RV and RVIs. The NRC staff has determined that the changes identified in the proposed LAR will not impact the remaining safety margins required for the following structural integrity assessments: RV surveillance program; RV USE assessment; P-T limits; PTS assessment; and RVIs and core support structures, and the information provided in the LAR and the response to the staff RAI is adequate to support the requested MUR power uprate.

3.8 Mechanical and Civil Engineering

Regulatory Evaluation

The NRC staff's review in the areas of mechanical and civil engineering focused on verifying that the licensee has provided reasonable assurance that the structural and pressure boundary integrity of SSCs at HNP will continue to be adequately maintained following the implementation of the MUR power uprate under normal, upset, emergency and faulted loading conditions, as applicable. The NRC staff's assessment of the proposed MUR power uprate in the areas of mechanical and civil engineering considered 10 CFR 50.55a and GDC 1, "Quality Standards and Records," GDC 2 "Design Bases for Protection Against Natural Phenomena," GDC 4, "Environmental and Dynamic Effects Design Bases," GDC 14 "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design," which are located in 10 CFR Part 50, Appendix A.

The acceptance criteria are based on continued conformance with the requirements of the following regulations: (1) 10 CFR 50.55a, and GDC 1 as they relate to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed; (2) GDC 2 as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC 4 as it relates to structures and components important to safety being designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal and accident conditions and these structures and components being appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids; (4) GDC 14 as it relates to the reactor coolant pressure boundary being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; and (5) GDC 15 as it relates to the RCS being designed with sufficient margin to ensure that the design conditions are not exceeded.

Section IV "Mechanical/Structural/Material Component Integrity and Design," of RIS 2002-03, provides information to licensees on the scope and detail of the information that should be submitted to the NRC staff regarding the impact an MUR power uprate has on the structural and pressure boundary integrity of the aforementioned SSCs.

Technical Evaluation

The NRC staff's review covers the structural and pressure boundary integrity of the piping, components and supports that make up the nuclear steam safety supply system (NSSS) and the balance-of-plant (BOP) systems. The scope also includes an evaluation of other new or existing SSCs that are affected by the implementation of the proposed MUR power uprate. Specifically, this review focuses on the impact of the proposed MUR power uprate on the structural integrity of the HNP pressure-retaining components and their supports and the RVIs. The NRC staff's review also focused on the impact of the proposed MUR power uprate on postulated high-energy line break (HELB) locations and corresponding dynamic effects resulting from the postulated HELBs, including pipe whipping and jet impingement. The NRC staff's review focused on verifying that the licensee provided reasonable assurance of the structural and pressure boundary integrity of the aforementioned piping systems, components, component internals and their supports under normal and transient loadings, including those due postulated accidents and natural phenomena, such as earthquakes.

The proposed 1.66 percent power uprate will increase the RTP level from 2900 MWt to 2948 MWt at HNP. The power uprate will be achieved by implementing the Cameron LEFM ChekPlus system. The LEFM provides a more accurate measurement FW flow and reduces the uncertainty in the FW flow measurement for HNP.

3.8.1 Power Uprate Evaluation Parameters

Table 4 "Revised NSSS Design Parameters for Harris Nuclear Plant MUR Uprating" of Enclosure 1 of the LAR shows the pertinent temperatures, pressures, and flow rates for the current and uprated conditions. The licensee evaluated the effects of the proposed MUR power uprate for the NSSS at a bounding power level of 2970.4 MWt. This power level corresponds to the proposed level following the uprate (2948 MWt) plus the revised uncertainty of 0.34 percent (2958 MWt), and an additional 12.4 MWt to account for RCP heat input. The licensee evaluated four bounding cases of NSSS parameters to ensure that the most limiting conditions at the proposed power level were captured. Cases 1 and 2 represent an average vessel temperature of 572 °F, with Case 2 assuming that 10 percent of SG tubes are plugged. Cases 3 and 4 represent an average vessel temperature of 588.8 °F, with Case 4 also representing 10 percent of SG tubes being plugged. As indicated in Enclosure 1 of the LAR, analyses supporting the implementation of the proposed MUR power uprate considered the most bounding parameter values of any of the four cases presented in the LAR submittal. Subsequently, the evaluations performed by the licensee to demonstrate continued structural and pressure boundary integrity of the aforementioned SSCs at the uprated conditions consider the most limiting values of the parameters stipulated in the four cases, depending on which parameters are used in the AOR for the SSCs.

As shown in Table 4, there is no change in the RCS operating pressure (2250 psia) for any of the four cases. Additionally, the thermal design flow of 92,600 gallons per minute (gpm) per loop remains unchanged due to MUR power uprate implementation. At full power, Cases 1 and 2 yield a vessel outlet temperature of 608 °F (from the current design condition of 623 °F) and a vessel inlet temperature of 536.0 °F (from the current design condition of 554.4 °F). Cases 3 and 4 depict that the vessel outlet temperature increases from 623.2 °F to 623.8 °F while the vessel inlet temperature decreases from 554.4 °F to 553.8 °F. The most limiting MS pressure increase is represented by Case 3, which shows an increase of 6 psia in the MS pressure at the uprated conditions; all other cases show a decrease in MS pressure. Under the most limiting conditions, the MS flow increases from 12.84 million pounds per hour (Mlbm/hr) to 13.10 Mlbm/hr at the uprated conditions. The licensee notes in Table 4 of Enclosure 1 of the LAR that analyses that are limited by high steam pressure assume a maximum steam pressure of 997 psia, MS temperature of 544.2 °F and 13.12 Mlbm/hr to ensure that optimal SG performance is accounted for, should this condition exist. Additionally, Table 4 in Enclosure 1 of the LAR indicates that the zero load MS temperature of 557 °F remains unchanged and bounds all other MS temperature changes at uprated conditions. Therefore, the HNP currently analyzed FW temperatures do not change as a result of MUR power uprate implementation.

The loading combinations used to structurally qualify Class 1, Class 2, and Class 3 piping, components and their supports for normal, upset, emergency, and faulted loading conditions are documented in Section 3.9 of the HNP FSAR. The design parameters for the RCS components at HNP, including the RPV, RCS loop piping, RCPs, SGs (primary and secondary side components) and the pressurizer are found in Chapter 5 of the HNP FSAR. The RCS components are designed to 650 °F (except the pressurizer and pressurizer surge line, which are designed to 680 °F) and 2,485 pounds per square inch gauge (psig). Chapter 10 of the HNP FSAR provides the design basis information for the secondary side systems, including the MS system and the FW and condensate system. In comparing the design values for these SSCs to the parameters in Table 4 of Enclosure 1 of the LAR, it is evident that the proposed MUR power uprate at HNP does not affect the design values for the SSCs affected by the uprate.

3.8.2 Pressure-Retaining Components and Component Supports

As stated in Section IV.1.A "Mechanical/Structural/Material Component Integrity and Design" of RIS 2002-03, certain SSCs must be evaluated to support the implementation of an MUR power uprate to determine whether the SSCs are able to support the implementation of the proposed power uprate. The evaluations discussed in Section IV of RIS 2002-03 focus on determining what impact the MUR power uprate would have on the AOR for a particular SSC in order to determine whether the AOR for a particular SSC needs to be revised as a result of the power uprate. If the AOR for a particular SSC was performed at conditions that bound those that will be present at the MUR power level, no further evaluation is required. The licensee confirmed throughout Section IV "Mechanical/Structural/Material Component Integrity and Design" of Enclosure 2 of its April 28, 2011, submittal that pressure-retaining components and corresponding supports were evaluated in accordance with their original design codes of record to support MUR implementation. The design codes of record are documented in Table IV-2 "Codes of Record" of the licensee's April 28, 2011, LAR submittal.

The pressure-retaining components that must be evaluated in support of an MUR power uprate include the following: the RPV, including the RPV shell, RPV nozzles and supports; the pressure-retaining portions of the control rod drive mechanism (CRDM); NSSS piping, pipe supports and branch nozzles associated with the RCS; BOP piping and supports; SGs, including their supports, the SG shells, secondary side internal support structures and nozzles;

the pressure retaining portions of the RCPs; the pressurizer, including the pressurizer shell, nozzles and the surge line; and safety-related valves. Furthermore, Section IV.1.B of RIS 2002-03 indicates that for those SSCs whose AOR are affected by implementation of an MUR power uprate, the licensee should address the following, as they relate to the impact of the uprate on the AOR: stresses, cumulative usage factors (CUF) (i.e., fatigue), FIV, and changes in temperature, pressure and flow rates resulting from the power uprate.

In reviewing the licensee's evaluation of pressure-retaining components and their supports, the NRC staff focused on those components and supports whose AOR was not bounded at MUR conditions (i.e., affected SSCs). The licensee was able to disposition a number of components and their associated supports as unaffected by the proposed MUR power uprate based on whether the plant parameter changes resulting from the MUR implementation affect the loads included in the AOR for the component and its supports. The licensee dispositioned other SSCs as unaffected based on whether the magnitude of the change in an operating parameter or load used in the AOR for particular SSCs was within the analytical tolerances of the AOR supporting the structural qualification of the corresponding SSC.

The licensee stated that the AOR related to the structural and mechanical evaluations of the following SSCs are unaffected by the proposed MUR power uprate at HNP: primary equipment supports; the RPV and RPV nozzles; SG nozzles; the pressure-retaining portions of the CRDMs; all BOP piping except for portions of the condensate system, portions of the extraction steam system, portions of the FW system, and portions of the heater vents and drains; RCS piping, including the RCS loop piping, the pressurizer surge line, attached Class 1 auxiliary piping, and RCS loop branch nozzles; the pressure-retaining portions of the RCPs; and the pressure-retaining portions of safety-related valves. For structural and mechanical components whose CUF are affected by environmental effects, the licensee confirmed in its submittal that the environmentally assisted fatigue evaluations performed in support of the HNP operating license renewal remain valid at the MUR power level.

The NRC staff considers these dispositions acceptable based on the following considerations: 1) the magnitude of plant parameter changes, as documented in Table 4 of Enclosure 1 to the licensee's April 28, 2011, submittal, are minimal and that the structural integrity of most plant SSCs would not be affected by the proposed power uprate; and 2) the licensee's approach to disposition certain SSCs as unaffected by the proposed power uprate is consistent with RIS 2002-03. Based on these considerations, the NRC staff concludes that there is reasonable assurance that the structural integrity of the aforementioned SSCs will be adequately maintained following the implementation of the proposed MUR power uprate.

3.8.3 Affected Pressure-Retaining Components and Supports

The licensee identified a number of pressure-retaining components and supports whose AOR did not envelope the conditions that accompany MUR implementation and, as such, the NRC staff's review focused on these components and their supports. For these components where the AOR is not bounding, with respect to MUR conditions, the licensee evaluated the AOR to determine what effects MUR implementation has on the stresses and fatigue usage factors (if applicable) associated with the component and its supports. These evaluations were performed to determine whether the component and its supports will continue to satisfy their design basis acceptance criteria following the implementation of the proposed MUR power uprate. The following pressure-retaining components are affected by the MUR and required further structural evaluations to qualify these components for the proposed MUR power level: BOP piping systems identified above, the pressurizer, and the SGs.

3.8.4 Balance of Plant (BOP) Piping and Supports

For those BOP piping systems that the licensee identified as unbounded by their AORs at the proposed MUR power level, the licensee performed further evaluations to demonstrate the structural and pressure boundary integrity of the BOP piping and supports at the proposed power level and summarized these evaluations in Section IV.1.A.v "Balance-of-Plant Piping (NSSS Interface Systems, Safety-Related Cooling Water Systems and Containment Systems) and Supports" of Enclosure 2 in its April 28, 2011, submittal. The licensee's method for evaluating BOP piping systems is described in its submittal and relies upon the development of change factors based on the flow, temperature, and pressure changes within a particular BOP piping system. The change factors are a direct comparison between the pressure, temperature, and flow in each system before and after the proposed power uprate. If a change factor is found to be less than or equal to 1.0, the licensee stated that the current AOR for the system bounds the conditions that the system would be exposed to at the proposed power level. If a change factor was greater than 1.0, the licensee performed a re-evaluation of the affected piping system to determine whether the allowable stresses prescribed by the design code of record would continue to be met following implementation of the proposed power uprate.

Following its evaluation, the licensee determined that portions of the condensate system, portions of the extraction steam system, portions of the FW system and portions of the heater vents and drains exhibited change factors greater than 1.0; all of the change factors that exceeded 1.0 were the result of temperature increases. The results of the evaluations performed for the portions of these systems demonstrated that the acceptance criteria associated with the design codes of record will continue to be satisfied following MUR implementation. The licensee also noted that no pipe support modifications of the BOP piping systems are required to support the proposed MUR power uprate.

The NRC staff considers the licensee's evaluation of the BOP piping and supports under MUR conditions acceptable. This acceptance is based on the fact that the licensee was able to demonstrate that the design basis requirements associated with the BOP piping systems (i.e., the systems design codes of record) will continue to be satisfied following the implementation of the MUR power uprate. As such, by maintaining compliance with the criteria stipulated in the design codes of record for the piping system, the NRC staff concludes that there is reasonable assurance that the structural integrity of the affected BOP piping and supports will be adequately maintained following the implementation of the proposed MUR power uprate.

3.8.5 Pressurizer

In its evaluation of the pressurizer at the proposed MUR power level, summarized in Section IV.1.a.viii "Pressurizer" of its April 28, 2011, submittal, the licensee stated that most of the HNP pressurizer components were bounded by the existing AOR at the uprated conditions and did not require further evaluation to support the proposed MUR power uprate. As previously indicated, the AOR for the pressurizer surge line piping are unaffected by the proposed power uprate, including the pressurizer surge line stratification evaluations included in the AOR. The licensee stated that the pressurizer spray nozzle and the surge nozzle were the only two components that were re-evaluated as part of the MUR power uprate implementation; this re-evaluation included the full, preemptive structural weld overlays which have been applied to the safe ends of both nozzles. The pressurizer spray nozzle evaluation was performed to support the increase in the maximum pressurizer spray flow rate, which will accompany the power uprate. The licensee stated that the AOR for the spray nozzle and the spray nozzle safe end weld overlay were reviewed to determine the impact of the increased spray flow rate on the analyses. Based on the revised conditions, the licensee stated that the ASME B&PV Code design basis acceptance criteria for the nozzle and the weld overlay will continue to be satisfied at the revised conditions resulting from the MUR power uprate.

With respect to the pressurizer surge nozzle, the licensee noted that its evaluation of the surge nozzle and its corresponding structural weld overlay at the MUR conditions resulted in an increase in fatigue usage at the surge nozzle knuckle location. However, the licensee confirmed that the applicable ASME B&PV Code requirements related to the stresses and fatigue usage in the surge nozzle and the safe end weld overlay will continue to be satisfied following the implementation of the proposed MUR power uprate. Of note, the licensee also evaluated the pressurizer safety/relief valve structural weld overlay. However, the AOR for this weld overlay bounds the conditions at which the weld will be in service at the revised MUR power level and, as such, its AOR remains unaffected and its design basis acceptance criteria remain satisfied at the MUR power level.

The NRC staff considers the licensee's assessment of the structural integrity of the pressurizer at MUR conditions acceptable. This acceptability is based on the fact that the licensee's evaluations demonstrated that the design code requirements related to the pressurizer, including all components and nozzles, will continue to be satisfied following the implementation of the MUR power uprate. As such, given that the design basis acceptance criteria remain satisfied at MUR conditions, the NRC staff concludes that there is reasonable assurance that the structural integrity of the pressurizer will continue to be maintained following implementation of the MUR power uprate at HNP.

3.8.6 Steam Generators

The licensee's evaluation of the structural integrity of the HNP SGs at the proposed MUR power level focused on the most limiting components within the primary and secondary sides. The licensee stated that by focusing on the most structurally limiting components in the SGs, the remainder of the SG components could be dispositioned as acceptable for operation at the MUR power level. This rationale was based on the licensee's assertion that if the most limiting components were deemed acceptable, as demonstrated by satisfying their design basis acceptance criteria, then the less limiting SG components were also acceptable as they are enveloped by the evaluation of the limiting components. The NRC staff considers this approach acceptable, given that the licensee's reconciliation of the stresses and fatigue usage factors in the most limiting SG components provides a sufficient means to characterize the structural integrity of the non-limiting SG components.

With respect to the SG primary side, the licensee identified the channel head-to-tubesheet junction as the most limiting area while the minor shell taps were identified as the most limiting secondary side component within the HNP SGs. Based on plant parameter changes from the current power level to the MUR power level, the licensee developed scaling factors to extrapolate the stresses and CUFs in the current AOR for the limiting components to the stresses and CUFs applicable to these components at the MUR power level. Following the licensee's extrapolation, the licensee confirmed that the SG components are acceptable for operation at the proposed MUR power level. The licensee also confirmed that the ASME B&PV Code limits on the primary-to-secondary differential pressure allowable values remain satisfied for normal operating and upset condition transients at the MUR power level. Based on the fact that the licensee has demonstrated that the design basis acceptance criteria associated with the most limiting SG components will remain satisfied following the implementation of the MUR power uprate, the NRC staff concludes that there is reasonable assurance that the structural integrity of the HNP SGs will be adequately maintained following implementation of the power uprate.

3.8.7 Reactor Vessel Internals

In accordance with Section IV.1.A.ii "Reactor Core Support Structures and Vessels Internals" of RIS 2002-03, the licensee evaluated the effects of the proposed MUR power uprate on the HNP RVIs. As previously stated, Section IV.1.B of RIS 2002-03 indicates that for those SSCs, including RVIs, whose AOR are affected by implementation of an MUR power uprate, the licensee should address the following, as they relate to the impact of the uprate on the AOR: stresses, cumulative usage factors (i.e., fatigue), FIV, and changes in temperature, pressure and flow rates resulting from the power uprate. The licensee summarized its evaluation of the effects of the proposed power uprate on the structural integrity of the RVIs in Section IV.1.A.ii "Reactor Vessels Internals" of Enclosure 2 of the LAR. Section 3.9.5 "Reactor Pressure Vessels Internals" of the HNP FSAR details the design criteria associated with the HNP RVIs. Mechanical and structural evaluations were performed by the licensee to determine any effects on the RVIs due to the conditions that would be present following the implementation of the proposed MUR power uprate. The licensee's mechanical evaluation of the RVIs focused on the impact of the proposed power uprate on the design basis loads due to a seismic event, LOCAs and FIV. The licensee confirmed in the April 28, 2011, LAR submittal that the MUR power uprate has no impact on the design basis seismic, LOCA and FIV loads used in the mechanical evaluation of the RVIs. The structural evaluations performed by the licensee for the RVIs at MUR conditions are summarized in Section IV.1.A.ii.5 "Structural Evaluation" of the licensee's April 28, 2011, LAR submittal.

The licensee focused on the effects of the higher heat generation effects on the AOR for the RVIs. By letter dated September 6, 2011, in response to an NRC staff RAI regarding the effect of the proposed MUR power uprate on the loads used in the AORs for the RVIs, the licensee confirmed that the MUR power uprate only affects the design parameters and loads associated with heat generation and all other design parameters and loads associated with the design of the RVIs are unaffected by the proposed power uprate. Further, with respect to the structural qualification of the RVIs, the licensee stated that the normal and upset loading conditions (Level A and Level B service limits, respectively) are the only loading conditions affected by loads involving heat generation effects. Based on these considerations, the NRC staff focused its review on those RVIs whose AORs are affected by loads resulting from heat generation effects.

In response to an NRC staff RAI regarding the design code of record used to qualify the RVIs for operation at the MUR power level, the licensee stated that the HNP FSAR references the stress limits found in the 1973 draft of Subsection NG, "Core Support Structures," of the ASME B&PV Code, Section III, as the design basis acceptance criteria for the RVIs. With respect to the evaluations performed for the RVIs to support the proposed MUR power uprate, the licensee stated that the stress limits from the 2004 Edition of the ASME B&PV Code, Section III, Subsection NG were used to qualify the RVIs. However, the licensee confirmed that the stress limits cited in the HNP FSAR for the RVIs will remain satisfied following MUR implementation. The NRC staff has reviewed the licensee's response, regarding the reconciliation of the acceptance criteria used to qualify the RVIs for operation at the MUR power level and the design criteria cited in the HNP FSAR, and finds the licensee's response acceptable. This acceptance is based on the fact that the stress limits found in Figure NG-3221-1 "Stress Categories and Limits of Stress intensities for service Levels A and B" of the original design criteria and the 2004 Edition of the ASME B&PV code are identical with respect to the requirements regarding Level A and Level B service limits. As such, the licensee was able to confirm that the original stress limits cited in the HNP FSAR remain satisfied.

The NRC staff also notes that the 2004 Edition of the ASME B&PV Code, Section III, Division 1, Subsection NG is incorporated by reference in 10 CFR 50.55a. As such, the use of these criteria provides an acceptable means to demonstrate continued compliance with the regulatory requirements associated with the use of codes and standards. However, the NRC staff notes that the use of design criteria other than those stipulated in the design bases is reviewed on a case-by-case basis, when these with respect to the criteria used to structurally qualify SSCs in support of a proposed license amendment. As such, the NRC staff's acceptance of the use of the 2004 Edition of Subsection NG of the ASME B&PV Code discussed here within is specific to the MUR power uprate LAR at HNP.

The RVIs that were identified as those being affected by higher heat generation effects are the upper core plate, lower core plate, core baffle plates, former plates, core barrel, thermal shield, baffle-former bolts, and barrel-former bolts. The licensee concluded that the AOR for most of the RVIs remains bounding and that no additional evaluations were required for these RVIs to accommodate the effects of higher heat generation at MUR power uprate conditions. However, the AORs for the upper core plate, lower core plate and baffle-former bolts were not bounding at MUR conditions and, as such, these RVIs required further evaluations to demonstrate that they will remain structurally qualified at MUR conditions. For the upper and lower core plates, the licensee stated that the higher heat generation rates were used to evaluate the stresses and fatigue usage factors in these components at MUR conditions against the applicable design criteria. Based on the results, the licensee stated that the stresses and fatigue usage factors for both components are acceptable.

The NRC staff considers the licensee's evaluation of the upper and lower core plates acceptable. This acceptance is based on the fact that the licensee evaluated these components in accordance with the existing design basis criteria and demonstrated that these criteria will remain satisfied following the implementation of the MUR power uprate. As such, by demonstrating that these original design criteria will continue to be satisfied at MUR conditions, the NRC staff concludes that there is reasonable assurance that the structural integrity of these components will be adequately maintained following implementation of the proposed power uprate. The NRC staff's review of the licensee's assessment of the baffle-former bolts is discussed below.

3.8.8 Qualification of Baffle-Former Bolts

The licensee indicated in its April 28, 2011, LAR submittal that the baffle-former bolts at HNP were qualified for operation at the proposed MUR power level based on an evaluation of the baffle-former bolts which have been previously qualified for operation at a facility similar to HNP, Almaraz Unit 2 (nuclear power plant in Spain). In response to an NRC staff RAI regarding the correlation between HNP and Almaraz Unit 2, the licensee indicated in its October 21, 2011, RAI response that it had compared the drawings, operating parameters, geometric parameters, design transients and heat generation rates of the two units to ensure that an adequate comparison of the baffle-former bolts could be deduced. The HNP operating parameters considered in the comparison are those that will be present at the MUR power level. The licensee provided the results of each of these comparisons in its October 21, 2011, RAI response, which confirmed that the pertinent features of Almaraz Unit 2 are identical or bound those features associated with the HNP baffle-former bolts. The NRC staff has reviewed the licensee's comparison efforts and concludes that the licensee's use of structural qualification results of the Almaraz Unit 2 baffle-former bolts is reasonable and acceptable, based on the similarity between the baffle-former bolt construction data and facility operating parameters of the two units.

With respect to the qualification of the Almaraz Unit 2 baffle-former bolts, the licensee's October 21, 2011, RAI response indicates that the structural qualification of the Almaraz bolts was performed in accordance with the testing provisions of the ASME B&PV Code, Section III, Subsection NG, in lieu of analytical evaluations of the bolts. As indicated above, the licensee focused on demonstrating that the Level A and Level B service limits applicable to the RVIs remain satisfied as a result of the increased heat generation rates. As previously stated, the stress intensity limits applicable to these service limits are stipulated on Figure NG-3221-1 of the ASME B&PV Code.

The licensee stated that the testing provisions detailed in NG-3228.4, "Tests for Level A and Level B Service Limits," were satisfied for the Almaraz Unit 2 baffle-former bolts to demonstrate compliance with requirements on average and maximum stresses for threaded structural fasteners stipulated in NG-3232.1 and NG-3232.2, respectively. With respect to the design requirements related to the fatigue of structural fasteners, the licensee stated that testing performed for the Almaraz Unit 2 baffle-former bolts demonstrated that the criteria of NG-3232.3 were also satisfied based on the results of the cyclic testing performed for the Almaraz Unit 2 bolts. Based on the testing results of the Almaraz Unit 2 baffle-former bolts, which were shown to satisfy all applicable stress intensity Level A and Level B service limits stipulated in Figure NG-3221-1, and the demonstrated applicability of these results to HNP, the licensee concluded that the HNP baffle-former bolts are also qualified for operation at the MUR power level.

The NRC staff reviewed the results of the licensee's RAI response regarding the testing performed to structurally qualify the Almaraz Unit 2 baffle-former bolts against the criteria of Subsection NG of the ASME B&PV Code and finds the licensee's assessment acceptable and applicable to HNP. This acceptance is based on the fact that the testing performed for the Almaraz Unit 2 baffle-former bolts was carried out in strict accordance with the criteria of Subsection NG, the intent of which was designated as the design basis criteria for the HNP RVIs. As such, given that the licensee has adequately demonstrated that the Almaraz Unit 2 baffle-former bolts satisfy the Level A and Level B service limits stipulated in Subsection NG, the NRC staff concludes that there is reasonable assurance that the baffle-former bolts at HNP are enveloped by this evaluation and, therefore, will remain structurally adequate at the MUR power level. This applicability is further substantiated by the fact that the baffle-former bolts in HNP are of the same construction as Almaraz Unit 2 and operate under identical operating conditions or operating conditions which are bounded by those of Almaraz Unit 2.

3.8.9 High Energy Line Break (HELB) and Associated Dynamic Effects

The licensee evaluated the effects of the proposed MUR power uprate on systems classified as high energy to determine whether any changes to the HELB AOR will result from the implementation of the power uprate. This assessment is summarized in Section IV.1.B.vii "High Energy Line Break Locations" of Enclosure 2 to the licensee's letter dated April 28, 2011. As indicated in the summary of the licensee's assessment, the current AOR for HELBs was reviewed to compare the temperatures, pressures, and flow rates in high energy piping at the uprated conditions with those in the current AOR. Based on this comparison, the licensee determined that the input parameters used in the current AOR bound those at the uprated conditions. As such, the licensee stated that the proposed MUR power uprate does not result in any new or revised pipe break locations. Subsequently, the licensee also concluded that the dynamic effects evaluations associated with the HELBs postulated in the current AOR, including those due to jet impingement and pipe whipping, remain valid at the uprated conditions.

The NRC staff has reviewed the licensee's evaluations related to determinations of HELB rupture locations and their corresponding dynamic effects. For the reasons set forth above, which demonstrate that the HELB analyses of record will remain bounding under the proposed MUR power level, the NRC staff concludes that the licensee has adequately addressed the effects of the proposed MUR on these analyses. The NRC staff further concludes that the licensee has demonstrated that all of the regulatory requirements applicable to the HELB analyses will continue to be met following implementation of the proposed MUR. Therefore, the NRC staff finds the proposed MUR acceptable with respect to the determination of rupture locations and dynamic effects associated with the postulated rupture of piping.

3.8.10 Mechanical and Civil Engineering Conclusion

The NRC staff has reviewed the licensee's assessment of the impact of the proposed MUR power uprate on the structural and pressure boundary integrity of pressure-retaining components and supports and RVIs. Additionally, the NRC staff reviewed the licensee's assessment of the effects on the HNP HELB AOR, including associated dynamic effects. Based on the reviews delineated above, the NRC staff finds the MUR power uprate acceptable with respect to the structural integrity of the aforementioned SSCs affected by the power uprate. This acceptance is based on the licensee's demonstration that the regulatory requirements related to the civil and mechanical engineering purview, will continue to be satisfied following implementation of the MUR power uprate.

3.9 Electrical Systems

Regulatory Evaluation

The regulatory requirements which the staff applied in its review of the application include:

- GDC 17, "Electric Power Systems," of 10 CFR Part 50, Appendix A requires, in part, that an onsite power system and an offsite electrical power system be provided with sufficient capacity and capability to permit functioning of structures, systems, and components important to safety. Conformance to GDC 17 is discussed in the HNP FSAR, Section 3.1 "Conformance with NRC Design Criteria."

- 10 CFR 50.63, "Loss of all alternating current [ac] power," requires, in part, that all nuclear plants have the capability to withstand a loss of all ac power (station blackout, (SBO)) for an established period of time, and to recover there from.

- 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," requires, in part, that licensees establish programs to qualify electric equipment important to safety, located in harsh environment.

Technical Evaluation

The licensee developed the LAR consistent with the guidelines in RIS 2002-03. The electrical equipment design information is provided in Enclosure 2, Section V "Electrical equipment Design" of the LAR.

The NRC staff reviewed the licensee's evaluation of the impact of the MUR power uprate on the following electrical systems/components: AC distribution system, power block equipment (main generator, transformers, isolated-phase bus duct), direct current (DC) system, emergency diesel generators (EDGs), switchyard, grid stability, SBO, and equipment qualification (EQ) program.

3.9.1 Alternate Current (AC) Distribution System

The AC distribution system is the source of power for the nonsafety-related buses and for the safety-related emergency buses. The system consists of the 6.9 kiloVolt (kV), 480 volt (V), and 120 V systems. The licensee stated that the proposed MUR power uprate will only affect the 6.9 kV buses while the 480 V buses will not see a load increase and 120 V system loads are independent of the MUR power uprate. The 6.9 kV loads that will be affected are the main FW pumps, condensate pumps, condensate booster pumps, heater drain pumps, and RCPs. The licensee stated that the increase in brake horsepower (hp) is small, in the range of 100 to 200 hp and the new brake hp is within the rated hp of the affected motors. The affected 6.9 kV motors will remain the same size, with no additional motors added, and the available fault current from these sources will not change. The power uprate conditions will not result in unacceptable steady-state voltages, overloads or short circuit ratings exceeding the design limit.

In response to an RAI on the degraded voltage relay setting under the worst case steady-state voltage at the 6.9 kV buses, the licensee, in letter dated August 31, 2011, provided information regarding the pre- and post-MUR steady-state voltage profiles on the buses due to the increased loads. The NRC staff reviewed the information and determined that the motor load changes due to MUR power uprate are all within the motor rated hp and that the post-MUR voltage changes at the emergency 6.9 kV buses 1A-SA and 1B-SB are insignificant and will not affect the degraded voltage relay nominal setpoints.

Furthermore, in response to a staff question on the effect of the Cameron International LEFM CheckPlus system on loading at the low voltage buses and other impacts on the power distribution system, the licensee stated in letter dated August 31, 2011, that the LEFM CheckPlus system is not required to function during a postulated accident or event and, therefore it uses 120 V AC nonsafety power sources. Therefore, the safety-related power supplies and the EDG loading are unaffected due to the LEFM CheckPlus system addition.

The NRC staff reviewed the LAR and licensee's responses to the staff's RAIs and finds that the minor load changes at the 6.9 kV system will not adversely impact the loadings and voltages of safety-related buses, and that the AC distribution system has adequate capacity to support the plant loading for the MUR power uprate condition.

3.9.2 Power Block Equipment

As a result of the power uprate, the rated thermal power will increase to 2948 MWt from the previously analyzed core power level of 2900 MWt. In the LAR, the licensee stated that the gross electrical output for the power uprate case is 1021.8 megawatt electric (MWe), which is 29.8 MW more than the gross electrical output of 992 MWe at pre-uprate conditions. The 29.8 MWe increase includes approximately 19 MWe associated with the MUR power uprate and the balance from the other upgrades. In response to a staff question regarding the other upgrades, the licensee explained in letter dated August 31, 2011, that the other upgrades consist of a cooling tower fill replacement and a high pressure turbine modification.

In the LAR, the licensee stated that the replacement generator has a nameplate rating of 1155 MegaVolt Amperes (MVA), at 75 psig hydrogen pressure, 0.94 power factor (PF) lagging (0.95 PF leading), and rated voltage of 22 kV. In response to a staff question on the capability of the replacement generator, the licensee in letter dated August 31, 2011, explained that the required generator capability is determined based on an 80 MWe (66 MWe corresponding to grid stability study plus some margin) generation increase from the pre-uprate condition (which includes postulated future power increases) and the corresponding MegaVolts Amperes Reactive (MVAR) increase in agreement with PEC Transmission Planning. Based on the explanation provided by licensee and the NRC staff's audit of the licensee's generator calculated capability curve, the NRC staff concludes that the main generator maximum capability supports the MUR uprate operation.

In the LAR, the licensee stated that according to main generator capability curve, at 1021.8 MWe, the main generator is capable of exporting approximately 430 MVAR, and importing approximately 410 MVAR. However, the MVAR export is administratively set at a limit of 175 MVAR, and the MVAR import is set at 150 MVAR based on the main generator voltage regulator minimum excitation limiter setting. In response to an RAI, the licensee explained in letter dated August 31, 2011, that the MVAR requirements for the generation additions is based on agreement between HNP and PEC Transmission Planning.

In the LAR, the licensee stated that with the 1021.8 MWe generator output associated with the MUR, the ratings of the existing three single-phase main step-up transformers, rated at a total 1008 MVA, will be exceeded, when the auxiliary loads are on the startup transformers (SUTs). The licensee will be procuring replacement main step-up transformers with a total rated capacity of 1275 MVA (three single phase transformers, each 425 MVA, at a temperature rise of

65 degrees Centigrade (°C)). The NRC staff verified that this capacity will carry the full main generator output under all required operating conditions. The licensee further stated that until the replacement transformers are installed, the uprated plant output will be limited to within the capacity of the existing main transformers, and this is supported by a regulatory commitment in the LAR.

In response to an RAI on the protective relays and setpoint changes due to installation of the replacement generator and main transformers, the licensee in letter dated August 31, 2011, stated that requisite changes have been completed to main generator protective relays due to the increase in the main generator rating. The requisite changes will also be made to the main transformer relaying as part of the transformer replacement project.

In the LAR, the licensee stated that the isolated phase bus (IPB) segment that connects the main generator output to the primary windings of the main transformers has a continuous rating for 28,830 Amperes (A) and is forced air cooled. This rating adequately bounds the 29.8 MWe increase (which includes 19 MWe due to power uprate) with the reactive power at the maximum HNP administrative limit. In response to a NRC staff question on the IPB rating corresponding to the 66 MWe increase for which the grid study has been performed, the licensee, in the letter dated August 31, 2011, stated that the existing IPB will require increased cooling to accommodate the increase in MWe generation and that the plant will implement a modification to the IPB during the planned spring 2012 refueling to increase the bus rating.

In response to an RAI, the licensee in letter dated October 20, 2011, stated that the proposed IPB rating with increased cooling will be 30,310 A, based on the nameplate rating of the replacement generator (1155 MVA), which will bound 80 MWe increase (66 MWe plus some margin). Therefore, the NRC staff finds that the proposed IPB rating is adequate for the MUR power uprate conditions. The licensee provided a regulatory commitment, see Section 4.0: "Regulatory Commitments," of this SE to limit plant output to capacity of the exiting IPB, prior to upgrading the IPB.

In the LAR, the licensee stated that there are two unit auxiliary transformers (UATs), 3-winding type, each rated at 60/67.2 MVA, with a 55°C/65°C temperature rise, connected to the 22 kV IPB from the main generator, which normally supply power to the auxiliaries of the unit by stepping down from 22 kV to 6.9 kV. During normal operating conditions, the UATs power the 6.9 kV switchgear, 480 V load centers, and motor control centers. As a result of the MUR power uprate, the affected BOP electric loads increase the loading on the UATs. However, the net increase will be in the range of 100-200 hp, or approximately 0.1-0.2 MVA of each UAT. The licensee calculated and demonstrated that unit transformer x-winding and y-winding loading increase leaves adequate margin in the transformer capacity. Therefore, the NRC staff finds that the existing UATs will not be overloaded by the increased load, and are capable of operation at MUR power uprate conditions.

In the LAR, the licensee stated that two SUTs, 3-winding type, each rated 36/48/60 MVA with a 55 °C temperature rise and 67.2 MVA at 65 °C temperature rise, connect the 230 kV switchyard to the 6.9 kV system. The 6.9 kV buses are powered from the SUTs during station start-up and shutdown conditions. The BOP electrical loads due to power uprate for the SUTs are the same loads for the UATs, therefore the loading on the SUTs will increase. The licensee stated that the same margins exist for SUTs as for UATs when they are supplying the plant auxiliary loads.

Therefore, the NRC staff finds that the SUTs will not be overloaded by the increase in the loading, and are capable of operation at MUR power uprate conditions.

3.9.3 Direct Current (DC) Distribution System

In the LAR, the licensee stated that the DC distribution system consists of 250 V and 125 V systems. A review of the DC distribution system performed by the licensee determined that there is no increase in electrical loading on the DC power system under MUR power uprate conditions. The existing DC power system margins will continue to be maintained. Since there is no DC load increase, the NRC staff finds that the existing dc system will continue to perform its design function under MUR power uprate conditions.

3.9.4 Emergency Diesel Generators

The HNP has two EDGs, one for each division, dedicated to the safety-related, redundant electrical buses. The EDG system provides a safety-related source of AC power to sequentially energize and restart loads necessary for safe shutdown of the reactor, and to maintain the reactor in a safe shutdown condition.

In the LAR, the licensee stated that the electrical loads that changed as a result of the MUR power uprate are not fed from the EDG system. There is no increase to the emergency bus loads supported by the EDGs, and the existing 703 kW EDG loading margin, as provided in the HNP FSAR, will be maintained. Based on the above, the NRC staff finds that the EDG system will continue to have adequate capacity and capability to power the safety-related loads under MUR power uprate conditions.

3.9.5 Switchyard

According to the FSAR Section 8.2 "Offsite Power System," the offsite (preferred) power system for the HNP includes the licensee's transmission network (grid), the 230 kV switchyard, and two 230/6.9 kV start-up transformer circuits. The switchyard accepts the electrical output of the unit and supplies the transmission system via seven 230 kV transmission lines. The switchyard is comprised of two 230 kV buses connected to the transmission system and SUTs, through breaker-and-a-half and double-breaker schemes, and with the main transformer through a double-breaker scheme. The switchyard provides plant electrical power through two SUTs during unit startup and shutdown conditions.

In the LAR, the licensee provided an evaluation of the switchyard components at MUR power uprate conditions and determined that adequate positive margin exists between the maximum worst case steady-state load and equipment ratings at power uprate conditions. The licensee stated that the maximum apparent power through the switchyard components at power uprate conditions will be approximately 1108.6 MVA. The switchyard component ratings for the 230 kV tie-line, breakers, disconnects and buses, exceed this value. The NRC staff reviewed the information and determined that the existing switchyard equipment ratings have adequate margins and the switchyard system is capable of supporting the MUR power uprate conditions.

3.9.6 Grid Stability

In the LAR, the licensee stated that the PEC Transmission Planning department performed a generator interconnection study corresponding to MUR power uprate conditions. The study assessed the impact of the power uprate on local transmission area power flows and voltages, transmission equipment short circuit withstand and interrupting capability, and included the transient and dynamic stability of the HNP and other nearby generation. Based on the study, the licensee determined that there is no power flow, short circuit, stability, or interconnection impacts on the grid transmission system, and therefore, no additions or modifications are required to accommodate the proposed MUR power uprate.

In response to an RAI regarding the above mentioned interconnection impact study, the licensee in its letter dated August 31, 2011, provided the following additional information:

- The power flow analysis was performed for the summer peak load conditions and extrapolated for future years' load forecast projections (an incremental increase of 66 MWe). Since the HNP is in a summer peaking utility, forecast transmission system loads for a given year are higher than the winter. The study results indicated that no transmission line overloads or unacceptable voltages would be expected as a result of proposed power uprate.

- The minimum required switchyard voltage and the plant post-trip auxiliary loading are provided by HNP Engineering to PEC Transmission Planning as required by North American Electric Reliability Corporation Reliability Standard NUC-001-2 "Nuclear Plant Interface Coordination." The PEC Transmission Planning uses this as input in determining how the transmission grid must be operated to ensure adequate switchyard voltage is available from offsite power. Since neither of these loading values was changed, the transmission system operating procedures do not require any change for power uprate conditions.

- The short circuit analysis was performed by PEC Transmission and Control Engineering and considered the impact of the new 1155 MVA rated generator as well as a proposed new transmission line to be terminated in the HNP switchyard in 2014. The analysis showed that the maximum available switchyard fault current would be less than 45,000 A. This value is less than the short circuit interrupting capability of the 63,000 A breakers in the switchyard. There is one exception (a 50,000 A breaker) and this breaker is scheduled to be replaced with a 63,000 A breaker in 2011. However, all the breakers meet the short circuit requirement.

The NRC staff reviewed the information provided in the LAR, and licensee's supplemental information in letter dated August 31, 2011, and finds that the proposed MWe increase would not adversely impact the transmission system and the stability of the grid.

3.9.7 Station Blackout (SBO)

Section 50.63 to 10 CFR Part 50 requires, in part, that each light water cooled nuclear power plant be able to withstand and recover from a loss of all AC power, referred to as an SBO. The HNP's SBO coping duration is 4 hours. This is based on the evaluation of the offsite power design characteristics, emergency AC power system configuration, and EDG reliability, in accordance with the evaluation procedure outlined in NUMARC 87-00 "Guidelines and

Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," and RG 1.155 "Station Blackout." The offsite power design characteristics include the expected frequency of a grid-related loss of offsite power, the estimated frequency of LOOP from severe and extremely severe weather, and the independence of offsite power.

In the LAR, the licensee evaluated the impact of power uprate on the alternate AC power source, emergency condensate storage tank inventory, Class 1E battery capacity, ventilation, compressed air, containment isolation and RCS inventory. The licensee stated that the HNP is an AC-independent plant, and it relies on the DC system for the necessary coping power and decay heat generated steam to operate the auxiliary FW system to cool the RCS. The HNP has two Class 1E battery systems with sufficient capacity, including 10 percent margin to power SBO loads for 4 hours. In response to an RAI, the licensee, in letter dated August 31, 2011, stated that equipment used during SBO remain unchanged under power uprate conditions, and therefore battery capacity margin is not impacted.

In the LAR, the licensee stated that the condensate storage tank has adequate inventory to maintain the plant in hot standby for 6 hours, followed by a 6-hour cooldown to residual heat removal (RHR) conditions. The storage tank inventory analysis based on 2958 MWt remains valid for MUR power uprate conditions and bound the SBO requirements.

In the LAR, the licensee stated that the ventilation for areas containing SBO equipment, the operation of containment isolation valves, and the compressed air requirements are unaffected by the MUR power uprate. The licensee also stated that there are no proposed plant modifications to the RCS system corresponding to the power uprate that can impact the RCS inventory during an SBO event.

Based on above information, the NRC staff finds that the MUR power uprate will not impact the requirements to meet HNP's SBO coping duration of 4 hours. The licensee will continue to meet the requirements of 10 CFR 50.63 under MUR power uprate conditions.

3.9.8 Equipment Qualification (EQ) Program

In the LAR, the licensee stated that based on the evaluation conducted to determine the effects of the power uprate on the EQ of electrical equipment, the power uprate will not impact the EQ. The revised operating conditions are bounded by equipment design limits and will not adversely diminish the capability of safety-related equipment in performing their intended safety function. The licensee further stated that the electrical equipment will continue to meet the requirements of 10 CFR 50.49 following the implementation of the MUR power uprate.

In response to an RAI to demonstrate that the increased P-T, and radiation levels considered for EQ of electrical equipment remain bounding for both normal and accident conditions for the power uprate conditions, the licensee in the letter dated August 31, 2011, stated that the licensee performed evaluations for the following conditions:

- LOCA and MSLB,
- MSLB/main FW line break (MFWLB) pressure/temperature outside containment,
- Other HELBs pressure/temperature outside containment,
- Radiation – accident and normal.

The licensee stated that the pressure/temperature profile calculations for both LOCA and MSLB Pressure/Temperature inside containment were performed using a power level of 102 percent of rated power. Also, FSAR Section 3.11.C, "Supplemental Environmental Qualification of Inside Containment Safety Related Electrical Equipment," indicates that the Mass and Energy (M&E) release calculations were performed assuming a 102 percent power level. Similarly, for outside containment, the MSLB and MFWLB analyses were performed using a power level of 102 percent. FSAR Section 3.11.E, "Supplemental Environmental Qualification of Safety Related Electrical Equipment Inside the Main Steam Tunnel," indicates that M&E release calculations were performed assuming a 102 percent power level. The licensee determined that the FSAR conclusions related to the long-term and short-term M&E releases remain valid for the HNP MUR power uprate, and therefore P-T profiles considered for EQ of electrical equipment are not impacted inside or outside the containment.

Regarding the radiation environment for both accident and normal operation under power uprate conditions, the licensee stated in the letter dated August 31, 2011, that "the HNP MUR power uprate operation will have no effect on the accident dose estimates [currently considered in the supporting records] inside and outside containment, and will continue to be conservative for the normal operation dose estimates in the Reactor Auxiliary Building." The normal operation dose inside containment is primarily dependent on the nitrogen (N-16) concentration in the RCS. Subsequent to the HNP MUR power uprate, the normal operation dose rate inside containment areas influenced by N-16 will increase by 1.66 percent. However, the actual power history would be: 14 years at core power level of 2,775 MWt, 10 years at a core power level of 2900 MWt, and 36 years at a core power level of 2948 MWt. For the above operation history, the effective power level averaged over the 60-year period is approximately 2,900 MWt. Therefore, considering the average power level operation, the dose estimates currently considered at 2900 MWt for inside-containment locations will remain valid for the MUR power uprate conditions.

In a letter dated August 31, 2011, the licensee also stated that Westinghouse found an error in the M&E software code, EPITOME, which could result in an increase in the containment P-T for the DEPS (double-ended pump suction) LOCA. In a letter dated October 20, 2011, the licensee provided containment composite LOCA/MSLB temperature and pressure profiles after addressing the EPITOME error. The licensee stated that based on its EQ process evaluation, the exiting peak temperature and pressure remain applicable, and the EQ peak temperature and pressure for the DEPS LOCA scenario have not changed. The HNP EQ process evaluates margins for peak temperature and pressure, and an overall margin identified as equipment's post-accident operability period for each component in accordance with the Institute of Electrical and Electronics Engineers (IEEE) Standard 323-1974 "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," acceptance criteria.

Based on review of information provided in the LAR, and the licensee's letters dated August 31 and October 20, 2011, the staff finds that the MUR power uprate will have no adverse impact on the licensee's ability to continue meet the EQ requirements of 10 CFR 50.49.

3.9.9 Electrical Systems Conclusion

Based on the technical evaluation provided above, the NRC staff finds that HNP will continue to meet GDC Criterion 17 of 10 CFR Part 50, Appendix A, the requirements of 10 CFR 50.63, and 10 CFR 50.49. Therefore, the NRC staff finds the proposed MUR power uprate acceptable.

3.10 Instrumentation and Controls

The HNP LAR referenced Cameron/Caldon Topical Report ER-80P and its supplement, Topical Report ER-157P, Rev. 8. These topical reports, which are generically applicable to nuclear power plants, document the ability of the Cameron International LEFM Check and CheckPlus systems to increase the accuracy of flow measurement. Together, these two reports and their respective SEs provide a generic basis for an MUR power uprate. The licensee's submittal also provides several attachments (proprietary) that describe the plant-specific bases for the proposed MUR uprate at HNP, Unit 1, Cameron Engineering Report ER-720, Rev. 2 and Cameron Engineering Report ER-697, Rev. 3.

Regulatory Evaluation

The NRC regulatory requirements related to the content of the TSs are set forth in 10 CFR 50.36, "Technical specifications," which requires that the TSs include limiting safety system settings (LSSS). This regulation requires, in part, that "where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective action will correct the abnormal situation before a safety limit is exceeded." Accordingly, the limits for instrument channels that initiate protective functions must be included in the TSs.

In accordance with GDC 20, "Protection System Functions," of Appendix A to 10 CFR Part 50, the protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety. Adherence to acceptable fuel design limits is called for in GDC 10, "Reactor Design"; these limits are specified in each plant's core operating limits report and maintained as a part of the Administrative TSs.

RG 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation"[17] describes a method acceptable to the NRC staff for complying with the NRC regulations for assuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits.

RIS 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, 'Technical Specifications,' Regarding Limiting Safety System Settings during Periodic Testing and Calibration of Instrument Channels,"[18] discusses issues that could occur during testing of LSSS and which, therefore, may have an adverse effect on equipment operability.

17 Regulatory Guide 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," U.S. Nuclear Regulatory Commission, December 1999 (ADAMS Accession No. ML993560062).

18 RIS 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, 'Technical Specifications,' Regarding Limiting Safety System Settings during Periodic Testing and Calibration of Instrument Channels," U.S. Nuclear Regulatory Commission, August 24, 2006 (ADAMS Accession No. ML051810077).

Technical Evaluation

Neutron flux instrumentation is calibrated to the core thermal power, which is determined by an automatic or manual calculation of the energy balance around the plant NSSS. The accuracy of this calculation depends primarily on the accuracy of FW flow and FW net enthalpy measurements. FW flow is the most significant contributor to the core thermal power uncertainty. A more accurate measurement of this parameter will result in a more accurate determination of core thermal power.

FW flow rate is typically measured using a venturi. This device generates a differential pressure proportional to the FW velocity in the pipe. Due to the need to improve flow instrumentation measurement uncertainty, the industry evaluated other flow measurement techniques and found the Cameron International Check and LEFM CheckPlus ultrasonic flow meters to be viable alternative.

The NRC staff's review in the area of instrumentation and controls (I&C) covers the proposed plant-specific implementation of the FW flow measurement technique and the power increase gained as a result of implementing this technique, in accordance with the guidelines (A through H) provided in Section I "Feedwater flow measurement technique and power measurement uncertainty" of Attachment 1 to RIS 2002-03, which relates to 10 CFR Part 50, Appendix K. The NRC staff conducted its review to confirm that the licensee's implementation of the proposed FW flow measurement device is consistent with staff-approved Topical Reports ER-80P and ER-157P Rev. 8. The NRC staff also reviewed the power measurement uncertainty calculations to ensure that (1) the conservatively proposed uncertainty value of 0.34 percent correctly accounts for all uncertainties associated with power level instrumentation errors and (2) the uncertainty calculations meet the relevant requirements of 10 CFR Part 50, Appendix K.

3.10.1 Leading Edge Flow Meter Technology and Measurement

The Cameron International LEFM CheckPlus systems uses transit time methodology to measure fluid velocity. The basis of the transit time methodology for measuring fluid velocity and temperature is that ultrasonic pulses transmitted through a fluid stream travel faster in the direction of the fluid flow than through the opposite flow. The difference in the upstream and downstream traversing times of the ultrasonic pulse is proportional to the fluid velocity in the pipe. The temperature is determined using a correlation between the mean propagation velocity of the ultrasound pulses in the fluid and the fluid pressure.

The LEFM CheckPlus system uses multiple diagonal acoustic paths instead of a single diagonal path, allowing velocities measured along each path to be numerically integrated over the pipe cross-section to determine the average fluid velocity in the pipe. This fluid velocity is multiplied by a velocity profile correction factor, the pipe cross-section area, and the fluid density to determine the FW mass flow rate in the piping. The mean fluid density may be obtained using the measured pressure and the derived mean fluid temperature as an input to a table of thermodynamic properties of water. The velocity profile correction factor is derived from calibration testing of the LEFM in a plant-specific piping model at a calibration laboratory.

The Cameron International LEFM Check system consists of a spool piece with eight transducers, two on each of the four acoustic paths in a single plane of the spool piece. The velocity measured by any one of the four acoustic paths is the vector sum of the axial and the transverse components of fluid velocity as projected onto the path. The Cameron International LEFM CheckPlus system uses 16 transducers, 8 each in two orthogonal planes of the spool piece. In the Cameron International LEFM CheckPlus system, when the fluid velocity measured by an acoustic path in one plane is averaged with the fluid velocity measured by its companion path in the second plane, the transverse components of the two velocities are canceled and the result reflects only the axial velocity of the fluid. This makes the numerical integration of four pairs of averaged axial velocities and the computation of volumetric flow inherently more accurate than a result obtained using four acoustic paths in a single plane. Also, since there are twice as many acoustic paths and there are two independent clocks to measure the transit times, errors associated with uncertainties in path length and transit time measurements are reduced.

The licensee provided the following information regarding the Cameron International LEFM CheckPlus system FW flow measurement technique and its implementation at HNP. The LEFM CheckPlus system consists of an electronic cabinet installed in the secondary sampling equipment enclosure located in the Turbine Building, and measurement spool pieces installed in each of the three main FW flow lines. Spool pieces in the A and B FW lines are installed well downstream of the existing venturis, and will have no impact on venturi performance. The spool piece in the C FW line is installed upstream of the venturi and will have no impact on that venturi's performance. The LEFMs were calibrated at the ARL facility using the current plant piping configuration and variations of the plant configuration. The calibration test determines the meter calibration constant, or meter factor. The meter factor provides a small correction to the numerical integration to account for fluid velocity profile specifics and any dimensional measurement errors. Parametric tests are performed to determine meter factor sensitivity to upstream hydraulics.

Each measurement section consists of 16 ultrasonic, multi-path, transit time transducers divided into two planes of eight (plane A and plane B), one dual resistance RTD, and two pressure transmitters. Both metering planes exist on each of the three FW lines, therefore, there are a total of 12 A metering plane paths and 12 B metering plane paths. The licensee can identify the different flow planes based upon plane path labeling using the HNP plant process computer. Each transducer may be removed at full power conditions without disturbing the pressure boundary. These flow elements conform to the installation location requirements specified in Topical Reports ER-80P and ER-157P Rev. 8.

The electronic cabinet controls magnitude and sequences transducer operation, makes time measurements; and calculates volumetric flow, temperature and mass flow. The system software employs the ultrasonic transit time method to measure velocities at precise locations. The system numerically integrates the measured velocities. System software has been developed and maintained under a verification and validation (V&V) program. The FW mass flow rate and temperature are displayed on the electronic cabinet and transmitted to the plant process computer for use in the calorimetric measurement (secondary plant energy balance) of reactor thermal output. The system utilizes continuous calorimetric power results transmitted by direct, redundant links to the plant computer, and incorporates self-verification features. These features ensure that system performance is consistent with the design basis.

A new calorimetric calculation was added to the plant computer for the LEFM indications of FW mass flow and temperature. It runs in parallel with the venturi-based flow and RTD temperature inputs currently used in the plant calorimetric measurement calculations. The plant computer system calorimetric programs will add a correction factor to the venturi calculation, to normalize the flow and temperature to the LEFM values. Both plant computer system calorimetric programs will receive data from SG blowdown flow to calculate calorimetric power. The existing venturi-based flow and RTD temperature will continue to be used for FW control and other functions, and may be used for plant calorimetric measurement when the LEFM is inoperable.

The Cameron International LEFM CheckPlus system has two operating modes ("normal" and "maintenance") and a "fail" mode. In the "normal" mode: normal operation is also known as CheckPlus mode. In this mode, both planes of transducers are in service and system operations are processed by both CPUs. Calculated power level uncertainty associated with the LEFM flow measuring system in this condition is 0.34 percent. In maintenance mode, when a plane of operation is lost, the system alerts the control room operators through the annunciator window for "computer alarm reactor," and shifts from normal operation to maintenance mode, also known as "check" mode.

In a response to an RAI dated August 25, 2011, the licensee stated that the power uncertainty associated with the system in "maintenance" mode was calculated assuming a loss of a single flow plane on all three FW lines. For example, although the failure of the A metering plane on one FW line would not result in the failure of the A metering planes on all three FW lines, the uncertainty associated with "maintenance" mode assumes that the A metering planes are lost. Likewise, although the failure of the B metering plane on one FW line would not result in the failure of the B metering planes on the other two FW lines, the maintenance mode uncertainty would assume that all three B metering planes are lost.

The calculated power level uncertainty associated with the LEFM flow measuring system in "maintenance" mode is 0.48 percent. As stated in the Cameron/Caldon Topical Report, ER-157P, Rev. 8, LEFM systems (Check or CheckPlus) provide accurate flow and temperature indications from synchronization to full power.

If the system suffers a loss of AC power or other total failure, the system also alerts the operators through the annunciator window "computer alarm reactor." Operations personnel are alerted to system trouble through the aforementioned annunciator if the electronic cabinet internal temperature is high or when other trouble conditions occur as determined by the plant computer.

The licensee stated in the application that an "alert" alarm is caused by:

1. Loss of a single process input:

    a. Loss of a single flow plane (loss of one or more flow transducers in a flow plane) on any FW line
    b. Loss of a single flow plane (loss of one or more flow transducers in a flow plane) in multiple FW lines
    c. Loss of a single redundant spool piece RTD on any line
    d. Loss of a single redundant FW header pressure input

2. Loss of a single electronics unit redundant component. The electronics unit includes two redundant systems, each one includes a separate power supply with 5 volt, +12 volt, -12 volt, 24 volt and 180 volt outputs; four acoustical signal processing units that transmit and receive ultrasonic flow signals; and a CPU that performs flow and temperature calculations, system self checks, and system verifications. The loss of any one of these components would produce an "alert" alarm.

3. Process input or output is calculated outside a pre-determined allowable range.

4. Internal self-check indicates system parameters that exceed pre-established limits and affect a single plane in one or more loops; for example, problems could be identified with the global synchronization signal board, signal rejects, signal transit time, path high gain, or speed of sound.

A "fail" alarm indicates a loss of function and the power level uncertainty reverts to the 2.0 percent error associated with the venturi flow meters, subject to the proposed AOT discussed in items G and H in Section 1 of Attachment 1 of RIS 2002-03.

The licensee stated in the application that a "fail" alarm is caused by:

1. Loss of both redundant process inputs:

   a. Loss of both planes (A & B) on a single FW line or multiple FW lines
   b. Loss of both redundant spool piece RTDs on a single loop
   c. Loss of both FW header pressure inputs

2. Failure of both redundant components in the electronics unit, such as both 180 V power supplies.

3. A process input or output is calculated outside a pre-determined allowable range by both CPU units.

4. Loss of the data link between the CheckPlus system and the plant computer.
5. Internal self-check indicates system parameters that exceed predetermined limits and affect multiple planes in one or both loops; for example, problems could be identified with the global synchronization signal board, signal rejects, signal transit time, path high gain, or speed of sound.

The licensee stated that with flow plane A inoperable in one FW line and flow plane B inoperable on a different FW line, the LEFM would automatically enter "fail" mode and would simultaneously generate a LEFM CheckPlus system "fail" alarm in the control room.

3.10.2 Conformance with RIS 2002-03, Attachment 1, Section I, Items A-H

Items A through C

Items A, B, and C in Section I of Attachment 1 to RIS 2002-03 guide licensees in identifying the approved Topical Reports, providing references to the NRC's approval of the measurement technique, and discussing the plant-specific implementation of the guidelines in the topical report and the NRC staff's approval of the FW flow measurement technique, respectively. In its LAR, the licensee identified Topical Reports ER-80P and ER-157P, Revision 8, as applicable to the Cameron International LEFM CheckPlus system. The licensee also referenced NRC SEs dated March 8, 1999, for Topical Report ER-80P, and dated August 16, 2010, for Topical Report ER-157P. In its response to item C, the licensee stated that the LEFM CheckPlus system is permanently installed in HNP, according to the requirements specified in Topical Reports, ER-80P and ER-157P Rev.8.

Based on its review of the licensee's submittals as discussed above, the staff finds that the licensee has sufficiently addressed the plant-specific implementation of the Cameron International LEFM CheckPlus system using proper Topical Report guidelines. Therefore, the licensee's description of the FW flow measurement technique and implementation of the power uprate using this technique follows the guidance in Items A through C of Section I of Attachment 1 to RIS 2002-03 and meets the regulatory requirements of 10 CFR Part 50, Appendix K.

Item D

Item D in Section I of Attachment 1 to RIS 2002-03 guides licensees in addressing four criteria, that the NRC staff stated in its SEs on Topical Reports ER-80P and ER-157P Rev. 8, when implementing the FW flow measurement uncertainty technique. The staff's SEs on Topical Reports ER-80P and ER-157P Rev. 8 both include these four plant-specific criteria to be addressed by a licensee referencing these topical reports for power uprate. The licensee's submittals address each of the four criteria as follows:

(1) The licensee should discuss the maintenance and calibration procedures that will be implemented with the incorporation of the LEFM. These procedures should include processes and contingencies for an inoperable LEFM and the effect on thermal power measurement and plant operation.

The licensee stated that implementation of the MUR power uprate will include developing the necessary procedures and documents required for operation, maintenance, calibration, testing, and training at the uprated power level using the LEFM CheckPlus system. A preventive maintenance program will be developed for the LEFM based on the vendor's maintenance and troubleshooting manual. Work on the LEFM will be performed by site I&C personnel qualified per the HNP I&C Training Program. The HNP Nuclear Information Technology group will assist when computer hardware or software maintenance is required. The preventive maintenance activities include:

- General terminal and cleanliness inspection
- Power supply inspection
- CPU inspection
- Acoustic processor unit checks
- Analog input/output checks
- Alarm relay checks
- Watchdog timer checks that ensure the software is running

- Communication checks
- Transducer checks
- Calibration checks on each FW pressure transmitter

Items G and H of this section of the SE discuss contingency plans for plant operation with an inoperable LEFM.

Based on its review of the licensee submittals, the NRC staff concludes that the licensee adequately addressed Criterion 1.

(2) For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installation and confirm that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Topical Report ER-80P.

The licensee stated that the LEFMs were installed in HNP during the fall 2010 RFO, with commissioning and calibration completed in November 2010. Since then, the LEFM FW flow and temperature data has been monitored and compared to the venturi FW flow and FW RTD output. The licensee stated that this data comparison demonstrated that the LEFM is consistent with the venturi FW flow and RTD FW temperature and that the LEFMs are functioning as designed. The licensee also stated that there have been no maintenance related activities since the LEFMs were installed.

Based on the above, the NRC staff finds that the licensee provided information comparing the venturi and LEFM in each of the three loops. The data provided showed that the LEFM is consistent with the venturi data. The NRC staff concludes that the licensee adequately addressed Criterion 2.

(3) The licensee should confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current FW instrumentation is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative approach is used, the application should be justified and applied to both venturi and the LEFM for comparison.

The licensee stated that the methodology used to calculate the LEFM uncertainty is based on accepted plant setpoint methodology and that an alternate methodology for calculating LEFM uncertainty was not used. The licensee stated that it uses a core thermal power uncertainty calculation approach consistent with Instrument Society of America (ISA) RP67.04.02-2000,[19] which is consistent with the guidelines in RG 1.105,[20] Revision 3, "Setpoints for Safety-Related Instrumentation," issued December 1999; and Topical Report ER-80P, as supplemented by ER-157P Rev. 8.

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19 ISA Standard ISA-RP67.04.02, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation," January, 2000.

20 Regulatory Guide 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," U.S. Nuclear Regulatory Commission, December 1999 (ADAMS Accession No. ML993560062).

The NRC staff finds that 1) HNP uses a methodology consistent to the approved methodologies in the Cameron Topical Reports to determine the uncertainty of the LEFM, 2) HNP uses a statistical approach to determine uncertainty in the setpoint methodology, 3) dependent parameters are arithmetically combined to form statistically independent groups and then combined using the square root of the sum of squares (SRSS) to determine the overall uncertainty, and the same fundamental approach was used to determine the UFM based power calorimetric uncertainty. Therefore, the NRC staff concludes that the licensee adequately addressed Criterion 3.

(4) For plant installation where the ultrasonic meter (including LEFM) was not installed with flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors are not representative of the plant-specific installation), licensees should provide additional justification for its use. The justification should show that the meter installation is either independent of the plant-specific flow profile for the stated accuracy, or that the installation can be shown to be equivalent to known calibrations and plant configurations for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, licensees should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

The licensee stated that the LEFMs were installed in HNP during the fall 2010 RFO with final commissioning being completed on November 2010. A LEFM bounding uncertainty has been provided for use in the uncertainty calculation described in Cameron Engineering Report ER-697, Rev. 2. The bounding calibration factor acceptability for the spool pieces was established by tests at the ARL as described in ER-720, Revision 2. These tests included a full-scale model of the HNP hydraulic geometry and a straight pipe. Test results were evaluated and documented in an ARL test data report and Cameron engineering report. The calibration factor used for the LEFM is based on these reports. The spool piece calibration factor uncertainty is based on the Cameron engineering reports. The site-specific uncertainty analysis documents these analyses and will be maintained as part of the HNP technical basis for the power uprate. A Cameron installation and setup review confirmed that the HNP LEFM CheckPlus system meets the requirements specified in ER-697, Rev. 2.

Based on the information given above and the NRC staff's review of the licensee's submitted calibration data in Cameron Engineering Reports ER-697 Rev. 2 and ER-720 Rev. 2, the NRC staff concludes that the licensee adequately addressed Criterion 4.

Based on its review of the licensee's submittals as discussed above, the NRC staff finds that the licensee has sufficiently addressed the plant-specific criteria stated in the SEs for Topical Reports ER-80P and ER-157P Rev. 8, and therefore follows the guidance in Item D of Section I of Attachment 1 to RIS 2002-03 and meets the regulatory requirements of 10 CFR 50, Appendix K.

Item E

Item E in Section I of Attachment 1 to RIS 2002-03 guides licensees in the submittal of a plant-specific total power measurement uncertainty calculation, explicitly identifying all parameters and their individual contributions to the power uncertainty.

To address Item E of RIS 2002-03, the licensee provided Cameron Engineering Report ER-697, Rev. 2. In addition, the licensee listed each parameter's contribution and the values for the overall thermal power calorimetric uncertainty in Table I-1 of Enclosure 2 to the LAR. The uncertainties documented in this table are based on Cameron Engineering Reports ER-720 and ER-697 Rev. 2. The NRC staff reviewed these reports and determined that the licensee properly identified all the parameters associated with the thermal power measurement uncertainty, provided individual measurement uncertainties, and calculated the overall thermal power uncertainty.

The licensee's fundamental approach used in the setpoint methodology is to statistically combine inputs to determine the overall uncertainty. Channel statistical allowances are calculated for the instrument channels. Dependent parameters are arithmetically combined to form statistically independent groups, which are then combined using the SRSS approach to determine the overall uncertainty. This methodology is consistent with the vendor's determination of the uncertainty of the Cameron International LEFM CheckPlus system, as described in the referenced Topical Reports, and is consistent with the guidelines in RG 1.105, Rev. 3.

As a result, the NRC staff finds that the licensee has provided calculations of the total power measurement uncertainty at the plant, explicitly identifying all parameters and their individual contributions to the overall thermal power uncertainty. Therefore, the licensee has adequately addressed the guidance in Item E of Section I of Attachment 1 to RIS 2002-03 and has met the regulatory requirements of 10 CFR Part 50, Appendix K.

Item F

Item F in Section I of Attachment 1 to RIS 2002-03 guides licensees in providing information to address the specified aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric.

In the LAR, the licensee addressed each of the five aspects of the calibration and maintenance procedures listed in Item F of RIS 2002-03, as follows:

(1) Maintaining Calibration

The licensee stated that LEFM hardware and instrumentation calibration and maintenance will be performed using procedures based on the appropriate Cameron International LEFM CheckPlus system technical manuals, thus ensuring that the LEFM remains bounded by the Topical Report ER-80P analysis and assumptions. The other calorimetric process instrumentation and computer points are maintained and periodically calibrated using approved procedures. Preventive maintenance tasks are periodically performed on the plant computer system and support systems to ensure continued reliability. Work is planned and executed in accordance with established HNP work control processes and procedures. Routine LEFM preventive maintenance activities will include, but not limited to, those activities specified in item D.1.

(2) Controlling Hardware and Software Configuration

The licensee stated that the Cameron International LEFM CheckPlus system is designed and manufactured per Cameron's 10 CFR Part 50, Appendix B, Quality Assurance Program and V&V Program. The licensee stated that after installation, the LEFM software configuration will be maintained using existing procedures and processes, which include V&V of software configuration changes. LEFM hardware and the calorimetric process instrumentation will be maintained per the HNP configuration control processes.

(3) Performing Corrective Actions

The licensee stated that plant instrumentation that affects the power calorimetric, including the LEFM inputs, will be monitored by HNP personnel. Problems detected are documented per the HNP corrective action program, with necessary follow-up actions planned and implemented.

(4) Reporting Deficiencies to the Manufacturer

The licensee stated that conditions found to be adverse to quality (as defined in 10 CFR Part 50, Appendix B) will be documented per the HNP corrective action program and reported to the vendor, as needed, to support corrective action.

(5) Receiving and Addressing Manufacturer Deficiency Reports

The licensee stated HNP has existing processes for addressing manufacturer's deficiency reports. Such deficiencies will be documented in the HNP corrective action program and actions will be controlled by the HNP work control process.

The NRC staff's review of the above information found that the licensee addressed the calibration and maintenance aspects of the Cameron International LEFM CheckPlus system and all other instruments affecting the power calorimetric. Therefore, the NRC staff finds that the licensee has met the guidance in Item F of Section I of Attachment 1 to RIS 2002-03 and the regulatory requirements of 10 CFR Part 50, Appendix K.

Items G and H

Items G and H in Section I of Attachment 1 to RIS 2002-03 guide licensees to provide a proposed AOT for the instrument and to propose actions to reduce power if the AOT is exceeded.

The licensee proposed a 72-hour AOT for operation at any power level above the current licensed power of 2900 MWt with the LEFM in "fail" mode, provided steady-state conditions persist throughout the AOT. The licensee's basis for the proposed 72-hour AOT is as follows:

- Operations procedures will direct the use of the back-up calorimetric in the event of LEFM failure ("fail" mode). This algorithm receives input from alternate plant instruments (FW venturis and RTDs) for FW flow rate calculation. During normal LEFM operations, the FW flow from the three venturis will be normalized to the LEFM FW flow rate, so that the alternate calorimetric matches the primary LEFM based calorimetric. Also, the FW RTD temperature measurements will be normalized to the more accurate data from the LEFM.

- HNP has performed a drift study of the FW flow transmitters. As found/as left calibration data from May 6, 2006, through November 8, 2010, was obtained for all six FW flow transmitters used by the emergency response facility information system calorimetric calculation. These transmitters are calibrated on a refueling interval basis, so the study included four complete calibrations for each transmitter and 27 drift data points per transmitter. The results indicate the worst case transmitter drift over the 18-month calibration interval is 0.45 percent. Conservatively assuming that all six FW flow transmitters (two per loop) drifted by this magnitude in the same direction, the impact on thermal power measurement over the proposed 72-hour completion time has been calculated as less than 0.1 MWt. This assumes a linear drift behavior over the 18-month interval. Based on the calculated average drift value for all FW transmitters over the 18-month interval, the impact on thermal power measurement over the 72-hour period would be negligible.

- One FW flow venturi is visually inspected each RFO. No venturi fouling has been observed to date. Based on these inspection results, it is very unlikely that venturi fouling or defouling would occur during the proposed 72-hour completion time.

- LEFM repairs are expected to be completed within an 8-hour shift. A completion time of 72 hours provides plant personnel sufficient time to diagnose, plan and package work orders, complete repairs, and verify normal system operation within original uncertainty bounds.

The 72-hour completion time begins when the annunciator alarm is received in the main control room. A control room alarm response procedure will be developed providing guidance to the operators for initial alarm diagnosis. Methods to determine LEFM CheckPlus system status and the cause of alarms are described in Cameron documentation. Cameron documentation will be used to develop specific procedures for operators and maintenance response actions.

A plant computer loss is treated as a loss of both the LEFM and the ability to obtain corrected calorimetric power using the alternate plant instrumentation. Operation with a plant computer loss at the uprated power level may continue until the next required nuclear instrumentation heat balance, which could be up to 24 hours. A plant computer failure will require reducing core thermal power to 2900 MWt as needed to support a manual calorimetric power calculation. These requirements ensure that an operable low uncertainty (less than 2.0 percent) input is used whenever core power is greater than 2900 MWt. With the LEFM in "fail" mode, if the plant experiences a power decrease below 2900 MWt (98.4 percent of RTP) during the 72-hour AOT, the maximum permitted power level will be the current licensed core power level of 2900 MWt until the LEFM is restored to either "normal" or "maintenance" mode operation. The operators will be provided with procedural guidance in the PLP-114 procedure for those occasions when the LEFM is inoperable.

A single path or plane malfunction ("maintenance" mode) results in an uncertainty change from 0.34 percent to 0.48 percent (0.14 percent difference). In the event of a failure of one path or plane that cannot be restored to full functionality ("normal" mode) within 72 hours, power will be reduced to approximately 99.86 percent RTP (2943 MWt, rounded down). The plant can operate at this power level indefinitely with a single plane of LEFM system. The operators will be provided with procedural guidance in the PLP-114 for those occasions when the LEFM is in the "maintenance" mode.

Table 3: "Maximum Allowable Power Levels" below, shows the proposed maximum allowable power levels for each LEFM mode:

Table 3: Maximum Allowable Power Levels		
LEFM Operating Mode	**Total Power Uncertainty (%)**	**Maximum (MWt)**
Normal	0.34%	2948
Maintenance	0.48%	2943
Fail	2.00%	2900

Based on the above discussion and the NRC staff's review of the licensee's LAR, RAI responses, and Cameron engineering reports, the NRC staff finds that the licensee provided sufficient justifications for the proposed AOT and the proposed power reduction actions if the AOT is exceeded. Therefore, the NRC staff concludes that licensee has followed the guidance in Items G and H of Section I of Attachment 1 to RIS 2002-03 and has met the regulatory requirements of 10 CFR Part 50, Appendix K.

3.10.3 Technical Specifications, Protection System and Emergency System Settings

Section VIII of Attachment 1 to RIS 2002-03 guides licensees in providing information to address the changes to the plant's TSs, protection system settings, and/or emergency system settings needed to support the power uprate.

Items A through C

Items A, B, and C in Section VIII "Changes to technical specifications, protection system settings, and emergency system settings" of Attachment 1 to RIS 2002-03 guide licensees in providing a description of the change, identification of analyses affected by and/or supporting the change, and the justification for the change for any analyses that support and/or are affected by the change. The NRC staff's evaluation of the identified instrumentation for new power level is based on the analytical limits documented by the licensee in the submitted application.

In its LAR, the licensee proposed to make changes to TSs Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoint," which describes the reactor protection system (RPS) functions. HNP employs a five-column format for this table. The columns are: total allowance (TA), Z (a factor that accounts for statistical summation of errors), sensor error, trip setpoint, and allowable value (AV). The TA term represents the difference between the safety analysis limit (SAL) and the Trip Setpoint. The licensee has proposed to make the following changes to TSs Table 2.2-1 to support the power uprate as shown in Table 4 "Changes to Technical Specifications":

Table 4 : Changes to Technical Specifications			
Functional Unit		Current Value	MUR Value
Power Range Neutron Flux – High Setpoint	Total Allowance	7.5	5.83
	Trip Setpoint	≤ 109% of RTP	≤ 108% of RTP
	Allowable Value	≤ 111.1% of RTP	≤ 109.5% of RTP
Power Range Neutron Flux – Low Setpoint	Total Allowance	8.3	7.83
	Allowable Value	≤ 27.1% of RTP	≤ 26.8% of RTP
Power Range Neutron Flux – High Positive Rate	Total Allowance	2.5	2.33
Power Range Neutron Flux – High Negative Rate	Total Allowance	2.5	2.33

The licensee also proposed to reduce the SAL for the Power Range Neutron Flux High Setpoint from 118 percent of current rated thermal power to 115 percent of the uprated power level. The licensee's basis for the proposed changes is as follows:

> The safety analysis trip setpoints in terms of absolute power are unchanged for the Power Range Neutron Flux Low, Power Range Neutron Flux High Positive Rate and Power Range Neutron Flux High Negative Rate reactor trips. However, these trips are listed in terms of percent RTP; therefore the safety analysis trip setpoints are re-designated based on the ratio of the current RTP to the uprated RTP (2900/2948). The TA for these three trips is recalculated to reflect the 1.66 percent increase in RTP.
>
> The power range, neutron flux high setpoint has additional impact beyond the TA change. The evaluation of non-LOCA events at power uprate conditions concluded that the results are acceptable and no trip setpoint changes were required. However, the margins for events that use statistical departure from DNBR methodology are fractionally affected in the negative (nonconservative) direction by the reduction in power uncertainty from 2.0 percent to 0.34 percent. Operating at a known slightly higher power level with less uncertainty resulted in a minimum DNBR with less margin to the safety analysis limit for the limiting event (uncontrolled rod withdrawal at power). To retain design margin for future cycle variability, the SAL for the Power Range, Neutron Flux High was reduced from 118 percent to 117 percent in terms of the current RTP. Considering the uprated power level, the 117 percent SAL was further reduced to 115 percent (117 percent x (2900/2948)). This change increases the DNB margin for limiting events to satisfy HNP objectives for design margin. The SAL change impacts the Power Range Neutron Flux High TA, Trip Setpoint, and AV.

In response to an RAI, the licensee provided the staff with portions of HNP calculation HNP-I/INST-1010, Revision 4, "Evaluation of [Reactor Trip System/Engineered Safety Features Actuation System] RTS/ESFAS TS Related Setpoints, Allowable Values and Uncertainties," which contains summary calculations for the proposed TS changes and a description of the methodology used to make the calculations. A previous revision of this document is referenced in Section 7.2 "Reactor Trip System" of the HNP FSAR.

The NRC staff reviewed the calculations provided by the licensee, including the SAL, TA, the channel statistical allowance (CSA), trip setpoint, margin, as-found tolerance (AFT), and as-left tolerance (ALT).

The licensee used the following criterion to set the ALT and AFT:

ALT = RCA

AFT = $\pm [(ALT)^2 + (RD)^2 + (RMTE)^2]^{0.5}$

Where,

RCA = rack calibration accuracy
RD = rack drift
RMTE = rack measurement and test equipment error

The results of the provided calculations are shown in Table 5: "Results of Power Range Neutron Flux Calculations":

Table 5: Results of Power Range Neutron Flux Calculations

	Power Range Neutron Flux - High	Power Range Neutron Flux - Low	Power Range Neutron Flux - High Negative Rate	Power Range Neutron Flux - High Positive Rate
SAL	115% RTP	34.4% RTP	7.8% RTP	7.8% RTP
TA	5.83% Span	7.83% Span	2.33% Span	2.33% Span
Trip Setpoint	108.0% RTP	25.0% RTP	5.0% RTP	5.0% RTP
CSA	4.72% Span	4.72% Span	1.45% Span	1.45% Span
Margin	1.11% Span	3.11% Span	0.88% Span	0.88% Span
RCA	0.50% Span	0.50% Span	0.50% Span	0.50% Span
ALT	0.50% Span	0.50% Span	0.50% Span	0.50% Span
AFT	1.12% Span	1.12% Span	1.12% Span	1.12% Span

The NRC staff reviewed the HNP calculations for each of the proposed TS changes and found that the licensee maintains positive margin and operates with proper bounds of AFT and ALT. Thus, the licensee provides adequate assurance that the control and monitoring of this setpoint are established and maintained in a manner consistent with plant safety function requirements. Therefore, the NRC staff finds that the licensee's proposed TS changes comply with 10 CFR 50.36 requirements and are acceptable.

The licensee also proposed to partially implement the Technical Specifications Task Force (TSTF)-493, "Clarify Application of Setpoint Methodology for LSSS Functions," Revision 4,[21]

21 Technical Specifications Task Force Traveler (TSTF)-493, "Clarify Application of Setpoint Methodology for LSSS Functions," Rev. 4, January 5, 2010 (ADAMS Accession No. ML100060064), and an errata sheet, "Transmittal of TSTF-493, Rev. 4, Errata," April 23, 2010 (ADAMS Accession No. ML101160026).

recommendations to the Nuclear Instrumentation setpoints being impacted by the MUR by adding the following two notes to TS Table 2.2-1:

> Note 7: If the as-found channel setpoint is outside its predefined as-found tolerance, the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.
>
> Note 8: The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Trip Setpoint in Table 2.2-1 (Nominal Trip Setpoint (NTSP)) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTSP are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine NTSPs and the as-found and the as-left tolerances are specified in EGR-NGGC-0153, "Engineering Instrument Setpoints." The as-found and as-left tolerances are specified in PLP-106, "Technical Specification Equipment List Program and Core Operating Limits Report."

The above notes correspond to TSTF-493 Option A, Notes 1 and 2, respectively. The licensee proposed to add these notes for the following functions in TS Table 2.2-1:

- Power Range Neutron Flux High Setpoint
- Power Range Neutron Flux Low Setpoint
- Power Range Neutron Flux High Positive Rate
- Power Range Neutron Flux High Negative Rate

The NRC staff reviewed the two notes proposed by the licensee and found that they are consistent with the intent of the two notes in Option A of TSTF-493, Revision 4. Therefore, the staff finds the addition of Notes 7 and 8 acceptable.

The NRC staff reviewed the proposed changes to TS Figure 2.1-1: "Reactor Core Safety Limits – Three Loops in Operation with measured RCS Flow > [293,540 GPM X (1.0 + C_1)]." The changes request to lower the core safety limit lines using approved methods for the power uprate. The NRC staff verified that the methods used were approved and that conservative assumptions and methods were used. The lines are confirmed by a process the licensee used that tests a sample of cases that vary pressure, power and T_{ave}. The license stated in the March 26, 2012 letter that the statistical combination of uncertainties for RCS temperature, power and pressure were done in accordance with AREVA methodology EMF-92-082(P)(A) "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors." The NRC staff compared the current and proposed values as shown in the Table 6: "Technical Specification Figure 2.1-1" below. The NRC staff found that lowering the core safety limit lines as submitted is conservative and therefore the NRC staff finds the proposed change acceptable.

The NRC staff reviewed the proposed changes to the TS Table 3.7-1 "Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steam Line Safety Valves During 3 Loop Operation." The changes request to lower the maximum allowable power range neutron flux high setpoint as a percent of rated power with specified number of inoperable steam line safety valves. The NRC staff finds the change to be conservative as lowering the maximum allowable power range neutron flux high setpoint at MUR conditions will allow the steam line safety valves to relieve pressure without challenging the system with increased steam flow caused by the higher power MUR conditions. A 50 percent of rated thermal power at MUR conditions produces more steam flow than 50 percent rated thermal power at current power levels. Lowering the setpoint will, therefore, lower the steam flow needed to be relieved at MUR conditions. Therefore, the NRC staff finds the proposed change acceptable.

Table 6: Technical Specification Figure 2.1-1

Pressure (psig)	Fraction of Rated Power	RCS T_{ave} (°F)	
		Current	Post-Uprate
2375	0	654.75	654.75
	0.96	626.9	625
	1.2	604.59	599
2235	0	645.26	645.25
	0.96	614.41	613
	1.2	595.09	589
1960	0	599.16	627
	0.96	599.16	598
	1.2	578.84	575

Based on the above discussion and the NRC staff's review of the licensee's LAR and RAI responses, the NRC staff found that the licensee provided sufficient justifications for the proposed TS changes. The NRC staff considers that the licensee has followed the guidance in Items A through C in Section VIII of Attachment 1 to RIS 2002-03 and has, therefore, met the regulatory requirements of 10 CFR Part 50, Appendix K.

3.10.4 Instrumentation and Controls Conclusion

The NRC staff reviewed the licensee's proposed plant-specific implementation of the FW flow measurement device and the power uncertainty calculations. Based on its review of the licensee's LAR, RAI responses, uncertainty calculations, and referenced Topical Reports, the NRC staff finds that the licensee's proposed amendment is consistent with the approved Topical Report ER-80P and its supplement Topical Report ER-157P Rev. 8, as well as with the guidance of RIS 2002-03 specific to this section of the SE. Therefore, the licensee's proposed amendment for NHP meets the relevant requirements of 10 CFR 50, Appendix K.

The NRC staff also finds that the licensee adequately accounted for all instrumentation uncertainties in the total thermal power measurement uncertainty calculations and demonstrated that the calculations meet the relevant requirements of 10 CFR Part 50, Appendix K and the guidance of RIS 2002-03.

The NRC staff concludes that the proposed TS changes meet the requirements of 10 CFR 50.36, the guidance of RG 1.105, Rev. 3 and TSTF-493, Rev. 4. Therefore, the NRC staff concludes that the I&C aspect of the proposed MUR thermal power uprate of 1.66 percent is acceptable.

3.11 Containment and Heating, Ventilation, and Air Conditioning (HVAC) Systems

Regulatory Evaluation

The NRC staff reviewed the licensee's LAR for compliance with the following regulations given in 10 CFR Part 50, Appendix A:

GDC 16, "Containment Design," requires that the containment shall provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment.

GDC 19, "Control Room," requires that the control room must provide the operators with the capability to operate the nuclear power units safely under normal conditions and maintain the reactor in a safe condition under accident conditions including a LOCA.

GDC 38, "Containment Heat Removal," requires that the containment heat removal systems are capable of rapidly reducing the containment P-T following a LOCA and maintaining them at an acceptably low level.

GDC 50, "Containment Design Basis," requires that the containment accommodate the P-T conditions resulting from a LOCA without exceeding the design leakage rate.

GDC 60, "Control of Releases of Radioactive Materials to the Environment," requires that the nuclear power unit have means to control the release of radioactive materials in gaseous and liquid effluents during normal operation and anticipated operational occurrences.

Regulatory guidance for the containment systems (primary and secondary) is found in the SRP, Chapter 6 "Engineered Safety Features." The regulatory guidance for the habitability, filtration and ventilation systems is found in RG 1.52 "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," RG 1.78 "Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," SRP Chapter 6 and SRP Chapter 9 "Auxiliary Systems."

Technical Evaluation

3.11.1 Containment

The NRC staff reviewed the following areas of containment design and analysis for the proposed HNP MUR power uprate: Long term LOCA containment response analyses, short-term LOCA containment response analyses, containment response to a MSLB and FW line break, impact of the MUR power uprate on the response to GL 96-06 "Assurance of Equipment Operability and Containment Integrity During design Basis Accident Conditions, NRC Generic Letter 96-06," dated September 30, 1996, and the impact of the MUR power uprate on the containment leak rate testing program.

The method HNP used to calculate the LOCA M&E release for FSAR Section 6.2.1 "Containment Functional Design" is an NRC approved method described in WCAP-10325-P-A "Westinghouse LOCA Mass and Energy Release Model for Containment Design," dated March 1979. Westinghouse, who performed the HNP containment analyses, identified a generic issue with the computer code, EPITOME, as it related to HNP LOCA long-term M&E. Westinghouse identified this issue as a result of an inconsistency in calculations performed by the EPITOME computer code that impacts post-blowdown LOCA M&E calculations. This error only impacts the double ended pump suction (DEPS) break. As a result of this issue, the EPITOME computer code has been revised, and the modifications made to the code do not invalidate the staff's SER for WCAP 10325-P-A. The short-term LOCA M&E calculations used for subcompartment analyses and the steam line break M&E release calculations are not impacted by the identified generic issue.

In response to an RAI, the licensee, in a letter dated August 15, 2011, provided revised results after resolving the errors in the LOCA M&E calculation and incorporating the results into the containment response analysis. HNP remains double ended hot leg (DEHL) break limited with no change in the peak DEHL break pressure at 56.50 psia and peak containment atmosphere temperature at 270.2 °F. The DEPS break blowdown peak pressure remains the same at 54.80 psia, but the post-blowdown peak pressure increases by 2.74 psia to a new value of 55.74 psia. As a result, the DEPS break case is now limited by the post-blowdown pressure. Although the DEPS analysis changed, the DEHL analysis remains bounding and unchanged.

The long term M&E release analyses for HNP assumed a core power of 2958 MWt or 102 percent of 2900 MWt. Thus, these analyses remain bounding for the MUR power uprate. The short term LOCA M&E releases (containment subcompartment response) analyses are affected by reductions in RCS temperatures, due to the fluid density effect on the initial pressure pulse created when the pipe ruptures. Therefore, the mass flux into the subcompartments would increase for a cold leg break. However, since RCS piping breaks have been eliminated by the leak-before-break methodology, and previous assessments indicated that the RHR system and accumulator line breaks near the reactor cavity and in the SG subcompartments are bounded by the original analyses, the only breaks evaluated for the power uprate are those in the pressurizer subcompartment. Power uprate conditions do not affect the pressurizer surge line M&E releases, because the RCS hot leg temperature increases and the increase in the pressurizer spray line M&E release is small and not limiting for the pressurizer subcompartment design basis. Therefore, the current subcompartment analyses are unaffected by the MUR power uprate and remain bounding.

The analyses for the long term steam line breaks inside and outside containment assumed a core power of 102 percent of 2900 MWt with the addition of 12.4 MWt for RCP heat. Thus, the analyses of the MSLB accident inside and outside containment are conservative and bounding for the MUR power uprate which is the expected result. The licensee states that the only critical parameter for the short-term FW line break is the maximum SG pressure. The bounding value for the SG pressure remains valid at uprate conditions and therefore, the M&E releases remain appropriate. The NRC staff agrees with the licensee's analysis and finds it acceptable for the MUR power uprated conditions for HNP.

The licensee states that there is no increase in the possibility of over pressurization of isolated segments of safety-related piping inside containment, including penetrations, due to the power uprate, as it relates to GL 96-06. In addition, the licensee states that there are no modifications to containment penetrations resulting from the power uprate. As a result, the conclusions in HNP's GL 96-06 responses and the associated NRC SER remain valid at MUR power uprate conditions.

The licensee indicated that the current bounding accident inside containment with respect to pressure is the large break LOCA (LBLOCA). The staff reviewed the LBLOCA response analysis and confirmed that the analysis was performed at 102 percent of 2900 MWt and remains bounding, with a corresponding peak containment pressure of 41.8 psig. Since the LBLOCA peak pressure analysis is unaffected by the power uprate, the test pressure specified in TS 6.8.4.k remains valid. Therefore, TS 6.8.4.k and the applicable HNP Containment Leakage Rate Program procedures remain acceptable at MUR power uprate conditions.

3.11.2 HVAC Systems

The NRC staff reviewed the impact of the MUR power uprate on the containment ventilation system, the control room ventilation system, the engineered safety features (ESF) ventilation system and the fuel handling area ventilation system.

The licensee stated that the power uprate does not require modifications that would change containment air volume and therefore, the functions to maintain the containment at a slight vacuum, relieve excessive containment vacuum, and purge the containment atmosphere prior to personnel entry are not impacted by the power uprate. The NRC staff has reviewed the licensee's analysis of the airborne activity at uprate conditions and its conclusion that containment radiological loading does not increase. The NRC staff finds the licensee's analysis acceptable. Therefore, operation of the containment ventilation system at MUR power uprate conditions is acceptable.

The licensee evaluated the control room heat loads (electrical, lighting, personnel) at MUR power uprate conditions and concluded that the main control room, office area, relay and termination cabinet rooms, kitchen and sanitary facilities, and the component cooling water (CCW) surge tank room are not impacted by the MUR power uprate because the heat loads in these areas do not increase. The NRC staff reviewed the licensee's evaluation and finds the operation of the control room ventilation system at MUR power uprate conditions acceptable.

The licensee evaluated the current limiting ESF heat loads and found no significant increase. This conclusion was expected and the NRC staff finds operation of the ESF ventilation system at MUR conditions to be acceptable.

The licensee evaluated the fuel-handling building (FHB) ventilation system. Since the design basis heat loads are not impacted, calculated space temperatures remain bounding and the FHB operating floor air conditioning system is acceptable at uprate conditions. The operating floor emergency exhaust system is acceptable, because the air flow rate exhausted and amount/concentration of radioactive particles will not increase beyond design. The SFP heat loads remain bounded at power uprate conditions and therefore, the FHB below operating floor ventilation system is adequate at uprate conditions. The SFP pump room piping heat loads will not increase, thus the calculated heat loads and space temperatures remain applicable, resulting in the SFP pump room ventilation system being adequate at power uprate conditions. Therefore, the NRC staff finds that the operation of the FHB ventilation system at MUR power uprate conditions is acceptable.

3.11.3 Containment and HVAC Systems Conclusion

The effect of the MUR power uprate on containment safety analyses is bounded by the current containment safety analyses. Therefore, HNP remains in compliance with GDC 16, 38, and 50. The increase of heat loads due to the MUR power uprate in the containment, control room and on the ESF ventilation systems is insignificant. Therefore, the NRC staff concludes that HNP Unit 1 remains in compliance with GDC 19 and 60.

3.12 Piping and Non-Destructive Examination

Regulatory Evaluation

The NRC staff has revised GDC 4 "Environmental and Dynamic Effects Design Bases" in Appendix A to 10 CFR Part 50 to permit exclusion of dynamic effects of postulated pipe ruptures from the design basis if the probability of pipe failure is demonstrated by analysis (i.e., the leak-before-break (LBB) analysis) to be extremely low. Once the NRC staff approves the LBB analysis, a licensee may remove pipe whip restraints and jet impingement barriers. The LBB analysis includes (a) an evaluation of potential active pipe degradation mechanisms (e.g., water hammer, creep damage, erosion, corrosion, fatigue, and environmental conditions; (b) a deterministic fracture mechanics analysis, and (c) an evaluation of RCS leak detection capability. Specific review criteria are contained in NRC SRP Section 3.6.3, "Leak-Before-Break Evaluation Procedures."

Technical Evaluation

Section II.2.40.b, "Short-term LOCA Mass and Energy Release Analysis," of the HNP MUR power uprate, dated April 28, 2011, states that after a LOCA, containment subcompartments are subject to pressure transients and jet impingement forces caused by the M&E releases from postulated high energy pipe ruptures within their boundaries as discussed in the FSAR Section 6.2.1.2 "Containment Subcompartments." The licensee analyzed the structural integrity of subcompartment walls for the short pressure pulse accompanying a high energy line pipe rupture. Subcompartments where high energy ruptures are postulated include the reactor cavity, pressurizer subcompartment, and the three steam generator subcompartments.

The licensee stated that the original HNP design and licensing bases were not based on LBB methodology. This means that the dynamic effects of large RCS pipe breaks were included (not excluded per GDC 4) in the original structural design basis for the containment.

FSAR Section 6.2.1.2 considered the following short-term LOCA M&E releases:

- Case 1: 150 in^2 cold leg break (reactor cavity blowdown)
- Case 2: 150 in^2 hot leg break (reactor cavity blowdown)

- Case 3: Double-ended cold leg break
- Case 4: Double-ended hot leg break
- Case 5: Double-ended pump suction break
- Case 6: Double-ended pressurizer surge line break
- Case 7: Pressurizer spray line break

Based on the NRC-approved LBB application, the licensee eliminated Case 1 through 5 breaks and evaluated only breaks in the largest branch lines (Cases 6 and 7). The NRC staff notes that the dynamic effects of the LBB pipe break should still be considered in the containment design as stated in the Statement of Consideration of the final rule to modify GDC 4, dated April 11, 1986 (51 FR 12502). The NRC staff asked, in an RAI, the licensee to clarify why the dynamic effect for breaks in the RCS main loop in Cases 1 through 5 were not evaluated for the impact of the power uprate on the containment design.

By letter dated September 7, 2011, the licensee responded that Cases 1 through 5 are breaks that were analyzed for subcompartment analyses prior to the revision to GDC 4 to permit the LBB application. The licensee stated that large breaks in the RCS piping are still used to calculate the global pressure (53 FR 66) internal to the containment. However, the revision to GDC 4 has allowed application of LBB to breaks within a containment subcompartment if that subcompartment does not provide a containment related function as discussed in NRC Inspection Manual, Part 9900: "Definition of Leak-Before-Break Analysis and its Application to Plant Piping Systems," Change Notice 96-020, September 26, 1996.

The licensee stated that dry containments such as the HNP containment do not rely on the subcompartments for long-term containment cooling during post-LOCA. The licensee further stated that the application of LBB technology, that limits the break size within a subcompartment, has been accepted by the NRC and has been the industry practice subsequent to revising GDC 4. The licensee noted that nevertheless the subcompartments at HNP were designed prior to the revision to GDC 4 and, thus, were designed to accommodate breaks in the largest high energy piping within a given subcompartment, including the double ended severance of the main RCS piping. Application of LBB for the MUR continues to meet the recommended margins, which eliminates the need for full reanalysis of the subcompartments.

The NRC staff finds that the dynamic effects from breaks in the RCS main loop Cases 1 through 5 piping are acceptable to be eliminated for those subcompartments that are not required for long term cooling post-LOCA as permitted in accordance with GDC 4. The NRC staff also finds that the subcompartments are in compliance with GDC 4 and design basis.

Section IV.1.B.vii.2, "Leak-Before-Break Evaluation," of the MUR submittal dated April 28, 2011, states that the existing LBB analyses justified eliminating large primary loop pipe rupture from the structural design basis. The licensee used the applicable pipe loadings, normal operating pressure, and temperature parameters at power uprate conditions in the evaluation. The licensee stated that the LBB acceptance criteria in SRP Section 3.6.3 are satisfied for primary loop piping at power uprate conditions, the recommended margins in SRP Section 3.6.3 are satisfied, and the existing analyses conclusions remain valid.

The NRC staff asked the licensee to discuss in detail exactly how the acceptance criteria and the recommended margins are shown to be satisfied, and how the existing analyses conclusions remain valid for primary loop piping at power uprate conditions. By letter dated September 7, 2011, the licensee responded that and LBB evaluation for the HNP primary loop piping due to the MUR power uprate was performed using the recommendations and criteria proposed in SRP Section 3.6.3 "Leak-Before-Break Evaluation Procedures." The licensee used the applicable pipe loadings, normal operating pressure, and temperature parameters at MUR power uprate conditions to evaluate LBB piping. The evaluation results show that the LBB acceptance criteria (margin of 10 on leak rate, margin of 2.0 on flaw size and margin of 1.0 on loads, using absolute summation method for faulted load combination) are satisfied at MUR power uprate conditions. The licensee stated that the LBB acceptance criteria are satisfied and therefore, the existing analyses conclusion to eliminate the dynamic effects of RCS primary loop piping breaks from the structural design basis remain valid at MUR power uprate conditions. The licensee further stated that the structural design basis of RCS primary loop piping breaks at MUR power uprate conditions has been eliminated for all applicable components. The NRC staff finds that the licensee has evaluated LBB analysis with MUR power uprate conditions and that the LBB piping satisfies the safety margins in SRP Section 3.6.3.

PWR plants have experienced PWSCC in nickel-based Alloy 82/182 dissimilar metal (DM) butt welds. The NRC staff asked the licensee to provide information regarding nickel-based Alloy 82/182 DM welds in LBB piping. By letter dated September 7, 2011, the licensee responded that LBB only impacts the RCS primary loop piping. The licensee stated that Alloy 82/182 DM welds are present in the RCS primary loop piping at the RV hot leg and cold leg nozzle connections to the reactor coolant loop piping. During RFO-16 in the fall of 2010, the licensee mitigated the Alloy 82/182 DM welds at the three hot leg RV nozzles using the mechanical stress improvement process (MSIP).

RIS 2010-07 "Regulatory Requirements for Application of Weld Overlays and Other Mitigation techniques in Piping systems Approved for Leak-Before-Break" states that "licensees may install mechanical stress improvement without NRC authorization since it does not affect the Code design or inspection requirements...." RIS 2010-07 also states that "mechanical stress improvement and Alloy 52 inlays and onlays would not substantially change the weld geometry or the original design-basis assumptions of the weld and, therefore, likely would not invalidate the original LBB analyses submitted to the NRC for approval...." The licensee has updated the original HNP LBB evaluation to show that all LBB margins in SRP Section 3.6.3 are satisfied in the RCS primary loop, including the Alloy 82/182 weld locations after the MSIP application and that the LBB evaluation continues to be valid.

HNP mitigated the hot leg DM welds during RFO-16 to reduce the risk of a future unplanned outage due to PWSCC, which is prevalent in DM welds at hot leg temperatures. To date the PWR industry has not experienced PWSCC in the RV nozzle cold leg DM welds.

The licensee stated that the three cold leg nozzle DM welds are mandated to be inspected during RFO-17 (spring 2012). The licensee further stated that the Alloy 82/182 DM welds at the cold leg nozzles are more challenging to mitigate due to existing interferences in the reactor vessel gallery. However, the experience gained through hot leg mitigation during the RFO-16 will become useful if it is decided to mitigate the cold leg nozzle DM welds in the future. The NRC staff finds that the licensee has mitigated Alloy 82/182 DM welds in hot leg nozzles and

the licensee will continue to inspect the Alloy 82/182 DM welds at the cold leg nozzles. The NRC staff notes that 10 CFR 50.55a(g)(6)(ii)(F) requires the licensee to inspect the Alloy 82/182 DM welds in accordance with ASME Code Case N-770-1 "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated With UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities, Section XI, Division 1,"with conditions applied under 10 CFR 50.55a(g)(6)(ii)(F).
The NRC staff notes, however, that one aspect of the original LBB analysis, the existence of components and welds which are susceptible to PWSCC and which have not been mitigated, is contrary to guidance found in SRP Section 3.6.3. The NRC staff has established precedent for accepting LBB analyses for conditions in which non-mitigated, PWSCC susceptible, welds or components are present based on increased inspections performed under ASME Code N-770-1 and 10 CFR 50.55a(g)(6)(ii)(F). In this instance, the NRC staff chooses to remain consistent with this precedent and accepts the licensee's LBB analysis despite its deviation from SRP Section 3.6.3 with respect to unmitigated Alloy 82/182 similar metal welds. The NRC staff is, however, reviewing the PWSCC issue with respect to LBB evaluations generically. If necessary, changes in the staff policy on this issue will be generically addressed for all plants.

Conclusion

On the basis of information submitted, the NRC staff finds that the licensee has demonstrated that the existing LBB piping under the MUR power uprate condition satisfy the guidance of SRP Section 3.6.3. The licensee also clarified that the piping in the subcompartments that do not provide a containment related function such as long term cooling post LOCA. The NRC staff concludes that the licensee has adequately addressed changes in primary system P-T and their effects on the LBB analyses. The NRC staff further concludes that the licensee has demonstrated that the LBB analyses will continue to be valid following implementation of the proposed MUR and that piping for which the licensee credits the LBB technology will continue to meet the requirements of GDC 4. Therefore, the NRC staff finds the proposed MUR acceptable with respect to LBB.

3.13 Plant Systems

Regulatory Evaluation

The NRC staff's review in the area of plant systems covers the impact of the proposed MUR power uprate on the NSSS interface systems, containment systems, safety-related cooling water systems, SFP storage, and cooling, and radioactive waste systems. The NRC staff's review is based on the guidance in SRP Chapters 3 "Design of Structures, Components, Equipment, and Systems," Chapter 6 "Engineered Safety Features," Chapter 9 "Auxiliary Systems," Chapter 10 "Steam and Power Conversion System," and Chapter 11 "Radioactive Waste Management," and RIS 2002-03, Attachment 1, Sections II, III, and VI. The licensee evaluated the effect of the MUR on the plant systems in Enclosure (2) of the licensee's application dated April 28, 2011.

By letter dated February 9, 2012, the licensee requested a nontechnical change to the TS Table 2.2-1: "Reactor Trip System Instrumentation Trip Setpoints," TS Table 3.3-1: "Reactor Trip System Instrumentation," and TS Table 4.3-1 to revise the main turbine terminology from

"Turbine Impulse Pressure" and "Turbine Impulse Chamber Pressure" with "Turbine Inlet Pressure," to support the spring 2012 RFO and the HNP modifications in support of the MUR.

The NRC staff reviewed the proposed changes to TSs for compliance with 10 CFR 50.36, and other regulatory requirements, including conformance with applicable GDC, to determine whether or not the proposed changes maintain adequate safety.

According to 10 CFR 50.36, licensees may revise their TSs provided that a plant-specific review supports a finding of continued adequate safety because: (1) the change is editorial, administrative, or provides clarification (i.e., no requirements are materially altered); (2) the change is more restrictive than the licensee's current requirement; or (3) the change is less restrictive than the licensee's current requirement, but nonetheless still affords adequate assurance of safety when judged against current regulatory standards.

Technical Evaluation

3.13.1 Nuclear Steam Supply System (NSSS) Interface Systems

The NSSS interface systems include the MS system, steam dump, the main FW (MFW) and condensate systems, and the auxiliary FW system (AFW).

Main Steam (MS) System

The MS system provides isolation of the SGs after a steam line failure, provides overpressure relief and/or decay heat removal during accidents, and provides steam to the AFW system. The MS system contains the MS safety valves (MSSVs), the SG power operated relief valves (PORVs), the MS isolation valves (MSIVs), the moisture separator reheaters (MSRs), and the steam dump system.

There are five ASME B&PV Code MSSVs located on each MS line outside the containment building and upstream of the MSIVs. MS overpressure events (loss of external load and turbine trip) have been analyzed at 2958 MWt or 102 percent of 2900 MWt. The licensee stated that the safety analysis confirms that the MSSV capacity is adequate for overpressure protection at MUR power uprate conditions.

There are three SG PORVs, one on each MS line. The SG PORVs are located upstream of the MSIVs and adjacent to the MSSVs. The primary function of the SG PORVs is to remove NSSS heat when the MSIVs are closed or the condenser, the condenser circulating water pumps, or steam dump system are otherwise unavailable. There is no change in the SG PORV function associated with the power uprate. The installed capacity of the SG PORVs would continue to satisfy the minimum flow criteria for NSSS cooldown because it was based on 2958 MWt or 102 percent of 2900 MWt.

The MSIVs provide a means to isolate a SG in the event of a downstream steam line break. This prevents the uncontrolled blowdown of more than one SG and minimizes the associated RCS cool down and containment pressure to within acceptable limits following a MS line break. The MSIVs are required to close within 5 seconds of the receipt of a closure signal, against steam break flow conditions in either the forward or reverse direction. The limiting MS line break MSIV loading is controlled by the postulated break size, the MS flow restrictor size, the MSIV seat bore, and the no-load MS pressure. These parameters are not affected by the proposed power uprate, and, therefore, the uprate does not affect the MSIV's ability to close within the required time period.

There are two MSRs to increase the quality and enthalpy of the steam exiting the high pressure main turbine. MSR flow rates are not a limiting design parameter based on bounding MSR design pressures and temperatures. The licensee stated that the 0.24 percent increase in steam mass flow rate in the primary steam lines to the MSRs is not expected to impact the MSR valves in these lines. Thus, the increase in steam flow would not affect the MSR valves actuated by the turbine overspeed protection system.

For the MS system, the licensee stated that the MS system pressure, temperature and velocities were evaluated for the MUR power uprate conditions and they are bounded by design parameters. Therefore, the MS system will continue to operate within its design parameters.

Steam Dump System

The steam dump system provides an artificial load by dumping excess steam to the atmosphere via eight steam dump atmospheric valves, directly to the condenser via six condenser dump valves, or a combination of the two. HNP was originally designed to accommodate 100 percent electrical load rejection. Accordingly, the steam dump system was designed with a capacity of 70 percent rated full-load steam flow. This is no longer a design basis requirement, as specified in FSAR Section 10.4.4.1 "Steam Dump System Design Bases." The current analyzed design basis is a maximum electrical load rejection of 50 percent of plant rated electrical load. A 50 percent electrical load rejection without reactor trip requires a steam dump system with a capacity equal to 40 percent of rated full-load steam flow. The licensee stated that a steam dump system hydraulic analysis concluded that for the proposed range of NSSS design parameters, the minimum steam dump system capacity with one inoperable valve would be approximately 61 percent. This minimum capacity exceeds the minimum steam dump system sizing requirement for the 50 percent electrical load rejection. The response of the steam dump control system to loss of load and turbine trip transients was analyzed at 2970.6 MWt or 102 percent of 2912.4 MWt. The licensee analyses showed acceptable steam dump system stability for both transients. Therefore, the NRC staff finds that the steam dump system is acceptable for operation at MUR power uprate conditions.

Main Feedwater and Condensate System

The MFW and condensate systems provide FW to the SG from the condenser hotwell during normal operation. The MFW system isolates during accidents. The MUR power uprate results in approximately a 1.9 percent increase in both FW and condensate flow. The licensee performed hydraulic calculations and determined that both the FW and condensate systems are capable of providing sufficient flow to the SGs under the MUR power uprate conditions.

The condensate system contains two 50 percent capacity condensate pumps discharging to two 50 percent variable speed condensate booster pumps. The condensate booster pumps discharge through low pressure FW heaters to the suction of the main FW pumps. The two

50 percent heater drain pumps take suction from four (A and B) FW heaters and discharge to the main FW pump suction. The MUR power uprate results in increased condensate flow of approximately 1.9 percent. Piping P-T are bounded by their design limiting values. Some condensate system piping flow velocities exceed the recommended limits at power uprate conditions. The lines that have operating temperatures less than 200 °F do not significantly impact the associated piping and are out of scope for the FAC Program. The remaining lines are included in the HNP FAC Program and will be monitored to ensure minimum wall thickness is maintained. Therefore, the condensate system will perform its design basis function adequately and is capable of supporting the MUR power uprate conditions.

The licensee evaluated the effect of the MUR on flow velocities within the MFW and condensate systems. The MUR power uprate results in increased condensate flow of approximately 1.9 percent. Piping pressures and temperatures are bounded by their design limiting values. Some condensate system, extraction steam, and feedwater heating piping component flow velocities exceed the recommended limits at power uprate conditions. The lines that have operating temperatures less than 200 °F do not significantly impact the associated piping and are not included in the FAC Program. The remaining lines are included in the HNP FAC Program and will be monitored to ensure minimum wall thickness is maintained. Therefore, the condensate and MFW systems will perform its design basis function adequately and is capable of supporting the MUR power uprate conditions.

Auxiliary Feedwater System

The AFW system provides FW to the SGs when the FW or condensate systems are unavailable The AFW analysis is based on 102 percent of 2900 MWt or 2958 MWt. The licensee stated that the analyzed core power level remains conservative and bounds the power uprate. AFW system maximum operating temperature and pressure remain essentially unchanged. There are no changes in AFW system minimum flow requirements, and no proposed changes to AFW pump design or operation. Therefore, the AFW system is capable of supporting the MUR power uprate.

Conclusion

The NRC staff reviewed the licensee's evaluations and found the results acceptable. The licensee determined that there is no adverse impact on the NSSS interface systems from the MUR power uprate because there is sufficient operating margin to produce an additional 1.66 percent power. The NRC staff concludes that an MUR power uprate will not challenge the NSSS interface systems. Therefore, the NRC staff finds that the NSSS systems are acceptable for the MUR power uprate.

3.13.2 Containment Systems

The safety-related containment cooling systems are the containment building spray system and the containment air cooling system. The spray system removes fission products from the post-accident containment atmosphere and assists in post-accident temperature and pressure control. The containment cooling system provides general area cooling and direct cooling to critical components. The containment air cooling system is designed to limit containment temperature to a maximum of 120 °F under normal operating conditions, and, in conjunction with the containment spray system, remove heat produced during a LOCA or MSLB. As discussed in Section 3.6 and 3.9, Reactor Systems and Electrical Systems, of this SE, respectively, the containment response analyses to both LOCA and MSLB were evaluated using M&E release based on 102 percent of current RTP. These analyses are bounding for the MUR power uprate. Therefore, the NRC staff finds the containment systems acceptable for the MUR power uprate.

3.13.3 Safety-Related Cooling Water Systems

The safety-related cooling water systems include the CCW system, the SW system, and the ultimate heat sink (UHS).

The CCW system provides the cooling requirements for all phases of plant operation including: startup, power operation, shutdown, refueling, and DBA cooldown. The licensee evaluated the CCW system to confirm that the heat removal capabilities are sufficient to satisfy the power uprate heat removal requirements during normal plant operations, refueling, shutdown and accident cooldown conditions. The power uprate will not increase the normal system operating heat loads and will not significantly increase heat loads for refueling, shutdown and accident cooldown cases. The revised CCW heat exchanger tube plugging requirements reduce the CCW heat exchanger effectiveness, causing a small increase in analyzed CCW operating temperatures for all cases. The increased RHR and letdown heat loads further affect the temperatures for refueling, shutdown, and accident cooldown cases.

The bounding case was more rigorously analyzed to ensure that the maximum system conditions do not exceed the currently analyzed maximum conditions. The cooldown times for the normal and single-train cases increase with the revised CCW tube plugging requirements and increased RHR and letdown heat exchanger loads, but remain reasonable. All component outlet temperatures are below the CCW system design temperature of 200 °F, with the exception of the gross failed fuel detector heat exchanger outlet temperature during plant shutdown at 350 °F with minimum CCW flow. HNP will increase CCW flowrate to the gross failed fuel detector heat exchanger for the plant shutdown at 350 °F, to ensure that the CCW outlet temperature from this component remains below the system design temperature of 200 °F for all plant operating modes. There are no required CCW system modifications as a result of the power uprate. The licensee's evaluation of CCW system heat removal capabilities confirmed that at uprated conditions, cooling of the affected NSSS components during normal and post-accident operation continues to meet the applicable system functional requirements and performance criteria, therefore, is capable of supporting the MUR power uprate.

The SW system is made up of two subsystems: the normal SW system and the emergency SW system. The normal SW system removes heat from plant auxiliary components during normal plant operation, including startup and shutdown, transfers the heat into the cooling tower and provides all cooling water requirements to the emergency SW loads during normal operation, but is not safety-related. Following an accident, the emergency SW pumps take suction from the UHS, circulate the water through the plant components required for reactor safe shutdown, and return it to the UHS. The licensee evaluated the various systems and components cooled by the SW system were evaluated to confirm that the SW system remains capable of removing power uprate heat loads during normal, shutdown, and accident conditions. The MUR power uprate will not significantly impact the heat loads and temperatures during normal and emergency operations. Therefore, the NRC staff finds the SW systems acceptable for the MUR power uprate.

The UHS uses two alternate sources of cooling water: the auxiliary reservoir and the main reservoir. The auxiliary reservoir is the preferred source of cooling water for emergency conditions, with the main reservoir providing a backup supply. The licensee evaluated the UHS auxiliary heat loads in the UHS analysis, and the loads are not changing. There are no changes to the CCW and SW flow rates. The most limiting CCW temperature for the power uprate is bound by the current analyzed conditions. The licensee stated that no system modifications are required to support the power uprate. Therefore, NRC staff finds that the UHS analysis is bounding and the UHS remains acceptable for operation at MUR power uprate conditions.

The NRC staff reviewed the licensee's evaluation of safety-related cooling water systems. Based upon the licensee's determination that the existing analyses for these systems were evaluated for 102 percent RTP, the NRC staff finds there is reasonable assurance that the systems are acceptable for the MUR power uprate.

3.13.4 Spent Fuel Pool (SFP) Storage and Cooling Systems

The principal function of the SFP storage and cooling system is to provide storage and cooling of the spent fuel. The primary impact of a power uprate would be to the decay heat of the fuel recently discharged from the core. SFP cooling heat exchangers are cooled by CCW. The licensee stated that the SFP bounding heat loads and system performance parameters are not affected by the power uprate and that the existing makeup capabilities can maintain the SFP at its required water level. Section 9.1.3 "Fuel Pool Cooling and Cleanup System" of the HNP FSAR states that calculations of the maximum amount of thermal energy to be removed by the spent fuel cooling system are made using the ORIGEN2 computer code, and the ORIGEN2 calculation will include a reactor power uncertainty value of 2 percent. The NRC staff does not expect that the MUR power uprate will result in a significant change to the operation of the SFP storage and cooling system. Therefore, the NRC staff concurs with the licensee's conclusion and finds that the SFP storage and cooling system will not be impacted by the power uprate.

3.13.5 Radioactive Waste Systems

The waste processing systems provide the means to sample, collect, process, store/hold, re-use, and/or release gaseous and liquid low-level effluents. The gaseous waste processing system functions and the volume of waste gas processed are unaffected by the MUR power uprate. The licensee stated that the existing system and equipment design margins are maintained, because system flow rates, gaseous inventories, and process conditions remain within the original system design parameters. The NRC staff reviewed the licensee's assessment and does not expect a 1.66 percent increase in power to result in a significant change to the operation of the radioactive waste systems.

The liquid radwaste system collects, monitors, processes, stores, and returns processed radwaste to the plant for reuse, discharge, or shipment. The licensee stated that the concentration of radionuclides in the liquid is expected to increase by a small amount, which does not significantly impact the system operation. The licensee concluded that the existing system and equipment design margins are maintained, because system flow rates, liquid inventories, and process conditions remain within the original system design parameters. The NRC staff concludes that the liquid waste processing system is bounded by the existing system design parameters and is acceptable at MUR power uprate conditions.

The licensee stated that the solid waste processing system is not currently in-service and most of the original processing equipment has been abandoned. The solid waste volumes are not significantly affected by the power uprate. The solid radwaste processing system is capable of processing the expected levels of uprated solid waste because the quantities of solid waste and the processing conditions are not significantly affected and remain within system operating margins. The solid waste processing system and solid radwaste processing system remain adequate and are acceptable at MUR power uprate conditions. Therefore, based on the licensee's assessment, the NRC staff finds that the radioactive waste systems will function adequately for the MUR power uprate.

3.13.6 Turbine Technical Specifications Terminology Change

The proposed change replaces the TS terminology "Turbine Impulse Pressure" or "Turbine Impulse Chamber Pressure" to "Turbine Inlet Pressure."

The applicable GDC for HNP is found in the UFSAR Section 3.1.9, Criterion 13, "Instrumentation and Control." UFSAR GDC 13, states that "[i]nstrumentation shall be provided to monitor variables and systems over their anticipated range for normal operation, for anticipated operational occurrences, and for accident conditions, as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges." HNP UFSAR GDC 13 is analogous to 10 CFR Part 50, Appendix A, GDC 13, "Instrumentation and control."

The February 9, 2012, supplemental letter indicates that the revision in the terminology of the function supporting interlock P-7 and P-13 from "Turbine Impulse Pressure" or "Turbine Impulse Chamber Pressure" to "Turbine Inlet Pressure" is due to the replacement of the turbine during the spring 2012 outage to support the MUR. The proposed change is in the following TSs Tables:

- TS Table 2.2-1: "Reactor Trip System Instrumentation Trip Setpoints," Functional Unit 19 Reactor Trip System Interlocks, b. Low Power Reactor Trips Block, P-7, 2) P-13 input Trip Setpoint, and Allowable Value.

- TS Table 2.2-1: "Reactor Trip System Instrumentation Trip Setpoints," Functional Unit 19. Reactor Trip System Interlocks, e. Functional Unit description, Trip Setpoint, and Allowable Value.

- TS Table 3.3-1: "Reactor Trip System Instrumentation," Functional Unit 19, Reactor Trip System Interlocks, e. Functional Unit description.

- TS Table 4.3-1: "Reactor Trip System Instrumentation Surveillance Requirements," Functional Unit 19, Reactor Trip System Interlocks, e. Functional Unit description.

In the letter, dated February 9, 2012, the licensee described the reason for this change as stated below:

> There is no technical impact in the terminology change from [Turbine Impulse Pressure or] Turbine Impulse Chamber Pressure to Turbine Inlet Pressure. The existing terminology is a specific sensing location for the function of Turbine Inlet Pressure. The turbine to be installed does not have an Impulse Chamber and the monitoring location will be changed to a functionally equivalent sensing location. Both the existing Turbine Impulse Chamber Pressure and the new Turbine Inlet Pressure measure the inlet pressure to the first full-arc stage which is used as an indirect reactor power equivalent. The proposed changes to replace the phrases "Turbine Impulse Pressure" and "Turbine Impulse Chamber Pressure" with "Turbine Inlet Pressure" does not involve any physical or design change to the P-13 function.

The requirement for the P-13 interlock within the RPS design is that the P-13 signal be representative of overall turbine power. This is accomplished by measuring the turbine first stage pressure, since turbine first stage pressure exhibits a consistent and accurate relationship with overall turbine power. The term "impulse" refers to a particular type of turbine blade design. The licensee is planning to replace its existing turbine during the spring 2012 outage as a result of the HNP modifications supporting the MUR. Hence, the licensee proposed TS changes to replace the words "Turbine Impulse Pressure" or "Turbine Impulse Chamber Pressure" to "Turbine Inlet Pressure," which result in text that states the basic P-13 requirement.

The NRC staff has determined that the proposed change in the description of HNP turbine is editorial in nature, does not involve any physical or design change for the P-13 function, and will have no effect on the operation of the RPS. The NRC staff further concludes that the licensee's proposed revision is acceptable since the change retains the required P-13 safety function, and will, thus, maintain conformance with GDC 13.

3.13.7 Plant Systems Conclusion

The licensee reviewed the design and operation of the plant systems and determined that the proposed MUR power uprate does not adversely impact any of the systems. The NRC staff also reviewed the proposed terminology changes for the turbine in the TSs dealing with the P-13 interlock. Therefore, the NRC staff concludes that the plant systems and the proposed TSs changes will be acceptable for the MUR power uprate.

4.0 REGULATORY COMMITMENTS

The list below identifies regulatory commitments made by the licensee in its LAR dated April 28, 2011 with a completion prior to operating above 2900 MWt (approximately 98.4 percent RTP), except for the commitment No. 4 for the plant electrical output:

1. Relocated Technical Specifications and Design Basis Requirements procedure

(PLP-114) will be revised to include LEFM controls (Enclosure 1 Section 2.5, Enclosure 2 Section I. 1.H).

2. Procedures and documents for the new LEFM will be established or revised (Enclosure 2 Sections I.1I.D.i.a, I.1.H, VII.2.A).

3. Appropriate personnel will receive training on the LEFM and affected procedures (Enclosure 2 Sections I. 1.D.i.a, VII.2.A, and VII.2.D).

4. Plant electrical output will be limited to the capability of the existing main transformers prior to installation of replacement main transformers. (Enclosure 2 Section V.1.F.iii).

5. Simulator changes and validation will be completed (Enclosure 2 Section VII.2.C).

6. Existing plant operating procedures related to temporary operation above full steady-state licensed power levels will be revised, as necessary (Enclosure 2 Section VII.4).

7. Required plant modifications (Enclosure 2 Section VII.3).

5.0 STATE CONSULTATION

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

6.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (September 13, 2011; 76 FR 56486). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 CONCLUSION

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

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Date: May 30, 2012

LIST OF ACRONYMS

A	amperes
AC or ac	alternating current
ADAMS	Agencywide Documents Access and Management System
ALT	as-left tolerance
AFT	as-found tolerance
AFW	auxiliary feedwater
AOP	abnormal operating procedure
AOR	analysis/analyses of record
AOT	allowed outage time
AOV	air-operated valve
ARL	Alden Research Laboratory
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
AST	alternative source term
ASTM	American Standard for Testing Materials
AV	allowable value
B&PV	Boiler and Pressure Vessel
BOP	balance of plant
C	Centigrade
CASS	cast austenitic stainless steel
CCW	component cooling water
CFR	Code of Federal Regulations
CLTP	current licensed thermal power
CPU	central processing unit
CP&L	Carolina Power & Light Company
CREA	control rod eject accident
CRDM	control rod drive mechanisms
CSA	channel statistical allowance
CUF	cumulative usage factors
CVCS	chemical and volume control system
DBA	design-basis accident
DBLOCA	design-basis loss-of-coolant accident
DC or dc	direct current
DEHL	double ended hot leg
DEPS	double ended pump suction
DNBR	departure from nucleate boiling ratio
DM	dissimilar metal
EAB	exclusion area boundary
ECCS	emergency core cooling system
EDG	emergency diesel generator
EFPY	effective full power years
EOL	end of life
EOLE	end of life extended
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
EQ	equipment qualification

ERG	emergency response guide
ESF	engineered safety features
F	Farenheit
FAC	flow-accelerated corrosion
FHA	fuel handling accident
FHB	fuel-handling building
FIV	flow-induced vibration
FOL	Facility Operating License
FR	*Federal Register*
FSAR	Final Safety Analysis Report
FW	feedwater
GDC	General Design Criteria/Criterion
GL	Generic Letter
gpm	gallons per minute
HELB	high energy line break
HFP	hot full power
HNP	Shearon Harris Nuclear Power Plant Unit 1
HVAC	heating ventilation and air conditioning
hp	horsepower
HZP	hot zero power
IASCC	irradiation assisted stress corrosion cracking
I&C	instrumentation and controls
IEEE	Institute of Electrical and Electronics Engineers
ISA	Instrument Society of America
IST	inservice test
IPB	isolated phase bus
kV	KiloVolt
LAR	license amendment request
LBB	leak-before-break
LBLOCA	large break loss-of-coolant accident
LEFM	leading edge flow meter
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPZ	low population zone
LR	locked rotor
LSSS	limiting safety system setting
LTOP	low-temperature overpressure protection
M&E	mass and energy
MeV	megaelectron Volt
MFW	main feedwater
MFWLB	main feedwater line break
Mlbm/hr	million pounds per hour
MOV	motor operated valve
MRP	materials reliability project
MS	main steam
MSIV	main steam isolation valves
MSIP	mechanical stress improvement process
MSLB	main steam line break

MSR	moisture separator reheater
MSSV	main steam safety valves
MUR	measurement uncertainty recapture
MWe	megawatt electric
MWt	megawatt thermal
MVA	MegaVolt Ampere
MVAR	MegaVolt Amperes Reactive
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
NSSS	nuclear steam supply system
NTSP	nominal trip setpoint
ODSCC	outside diameter stress corrosion cracking
OM Code	Code for Operation and Maintenance of Nuclear Power Plants
PEC	Progress Energy Carolinas Inc.
PF	power factor
pH	potential of hydrogen
PLTB	pressure locking/thermal binding
PORV	pressure operated relief valve
psi	pounds per square inch
psia	pounds per square inch atmosphere
psig	pounds per square inch gauge
P-T	pressure and temperature
PTS	pressurized thermal shock
PWR	pressurized-water reactor
PWSCC	primary water stress corrosion cracking
RAI	Request for Additional Information
RCA	rack calibration accuracy
RCCA	rod cluster control assembly
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RD	rack drift
RFO	refueling outage
RG	Regulatory Guide
rem	roentgen equivalent man
RHR	residual heat removal
RIS	Regulatory Issue Summary
RMTE	rack measurement and test equipment error
RPS	reactor protection system
RPV	reactor pressure vessel
RTD	resistance temperature detector
RTP	rated thermal power
RT_{PTS}	reference temperature for pressurized thermal shock
RV	reactor vessel
RVI	reactor vessels internals
SAL	safety analysis limit
SBLOCA	small break loss-of-coolant accident

SBO	station blackout
SCC	stress-corrosion cracking
SE	Safety Evaluation
SER	Safety Evaluation Report
SFA	steam feedwater application
SFP	spent fuel pool
SG	steam generator
SGBS	steam generator blowdown system
SGTR	steam generator tube rupture
SRP	Standard Review Plan
SRSS	square root of the sum of squares
SSC	structures, systems and components
SUT	startup-transformer
SW	service water
T	thickness
TA	total allowance
TEDE	total effective dose equivalent
TS	Technical Specification
TSTF	Technical Specification Task Force
UAT	unit auxiliary transformers
UFM	ultrasonic flow meters
UFSAR	Updated Final Safety Analysis Report
UHS	ultimate heat sink
USE	upper shelf energy
V	Volt
V&V	verification and validation
VCT	volume control tank
WCAP	Westinghouse Commercial Atomic Power (report)

May 30, 2012

Mr. Chris Burton, Vice President
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SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 – ISSUANCE OF AMENDMENT RE: MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NO. ME6169)

Dear Mr. Burton:

The Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 139 to Renewed Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant Unit 1 (HNP). The amendment is in response to an application submitted by Carolina Power & Light, dated April 28, 2011, as supplemented by letters dated June 23, August 3, August 15, August 25, August 30, August 31, September 6, September 7, October 20, October 21, October 28, November 28, December 20, 2011, February 9, and March 26, 2012.

The amendment revises the HNP renewed facility operating license and certain technical specifications to implement an increase of approximately 1.66 percent in rated thermal power from the current licensed thermal power of 2900 megawatts thermal (MWt) to 2948 MWt. The changes are based on increased feedwater flow measurement accuracy, which will be achieved by utilizing Cameron International Corporation (formerly Caldon) Cameron Leading Edge Flow Meter CheckPlus system to improve the HNP calorimetric heat balance measurement accuracy.

A copy of the related safety evaluation is enclosed. A notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,
/RA/
Araceli T. Billoch Colón, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Amendment No. 139 to NPF-63
2. Safety Evaluation

cc w/enclosures: Distribution via ListServ

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