Enclosures 6, 9, and 11 Contain Proprietary Information to be Withheld from Public Disclosure Pursuant to 10 CFR 2.390

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JUL 7 2017

10 CFR 50.90

LR-N17-0044
LAR H17-03

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Hope Creek Generating Station
Renewed Facility Operating License No. NPF-57
NRC Docket No. 50-354

Subject: **License Amendment Request for Measurement Uncertainty Recapture (MUR) Power Uprate**

In accordance with the provisions of 10 CFR 50.90, PSEG Nuclear LLC (PSEG) is submitting a request for an amendment to the Hope Creek Generating Station (Hope Creek) Renewed Facility Operating License (OL) NPF-57, and Technical Specifications (TS).

The proposed amendment will increase the rated thermal power (RTP) level from 3840 megawatts thermal (MWt) to 3902 MWt, and make TS changes as necessary to support operation at the uprated power level. The proposed change is an increase in RTP of approximately 1.6%, which does not exceed 120% of the Original Licensed Thermal Power (OLTP).

The proposed power uprate is characterized as a measurement uncertainty recapture (MUR) using the Cameron Leading Edge Flow Meter Check Plus (LEFM √+) ultrasonic flow measurement instrumentation. This reduces uncertainty in the feedwater flow and temperature measurement, which reduces the total power level measurement uncertainty.

PSEG developed this License Amendment Request using the guidelines in NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications." NRC requests for additional information (RAIs) associated with MUR applications for nuclear stations identified in Enclosure 1, Section 4.2, "Precedents," were reviewed for applicability. Information addressing the general topics of those requests is included within the body of this submittal.

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This submittal contains the following Enclosures:

Enclosure 1	Description and Evaluation of the Proposed Change.
Enclosure 2	Mark-up of Renewed Facility Operating License and Technical Specifications.
Enclosure 3	Mark-up of Technical Specification Bases "For Information Only."
Enclosure 4	Regulatory Issue Summary (RIS) 2002-03 Cross-Reference.
Enclosure 5	Summary of Regulatory Commitments.
Enclosure 6	GE-Hitachi Nuclear Energy (GEH) Document NEDC-33871P, "Safety Analysis Report for Hope Creek Generating Station Thermal Power Optimization," Revision 0, (Proprietary Version).
Enclosure 7	Affidavits from GEH and the Electric Power Research Institute (EPRI) Supporting the Withholding of Information in Enclosure 6 from Public Disclosure.
Enclosure 8	GEH Document NEDO-33871, "Safety Analysis Report for Hope Creek Generating Station Thermal Power Optimization," Revision 0, (Non-Proprietary Version).
Enclosure 9	Cameron Document ER-1123P, "Bounding Uncertainty Analysis for Thermal Power Determination at Hope Creek Unit 1 Nuclear Generating Station Using the LEFM √+ System," Revision 2 (Proprietary Version).
Enclosure 10	Cameron Document ER-1123NP, "Bounding Uncertainty Analysis for Thermal Power Determination at Hope Creek Unit 1 Nuclear Generating Station Using the LEFM √+ System," Revision 2 (Non-Proprietary Version).
Enclosure 11	Cameron Document ER-1132P, "Meter Factor Calculation and Accuracy Assessment for Hope Creek Nuclear Generating Station," Revision 2, (Proprietary Version).
Enclosure 12	Cameron Document ER-1132NP, "Meter Factor Calculation and Accuracy Assessment for Hope Creek Nuclear Generating Station," Revision 2, (Non-Proprietary Version).
Enclosure 13	Affidavits from Cameron International Corporation Supporting the Withholding of Information in Enclosures 9 and 11 from Public Disclosure.
Enclosure 14	Calculation SC-BB-0525, "Hope Creek Heat Balance Uncertainty Calculation," Revision 5.
Enclosure 15	LEFM Flow Meter Installation Location Drawings.

PSEG considers this LAR as linked to the previously submitted LARs for Power Range Neutron Monitor (PRNM LAR H15-01, LR-N15-0178, September 21, 2015), and Pressure-Temperature (P-T) Limits Curves (P-T Limits LAR H17-02, LR-N17-0032, March 27, 2017).

The PRNM LAR and this MUR LAR revise some of the same TS Reactor Trip Function and Control Rod Block function instrumentation setpoints. A new License Condition 2.C.(28) is proposed, as shown in Enclosure 2, to restrict Hope Creek operation at a thermal power level not to exceed 3840 MWt until the PRNM system license amendment request is approved by the NRC and implemented by PSEG.

The revisions to the P-T Limits curves affect the information required by the Enclosure 6 evaluations performed for this MUR LAR, as discussed in Section 2.0 of Enclosure 1. Therefore, a new License Condition 2.C.(29) is proposed, as shown in Enclosure 2, to restrict Hope Creek operation at a thermal power level not to exceed 3840 MWt until the P-T Limits curves license amendment request is approved by the NRC and implemented by PSEG.

Enclosure 6 contains proprietary information as defined by 10 CFR 2.390, which has been determined to be proprietary by GEH and the Electric Power Research Institute (EPRI). Affidavits supporting this request for withholding from public disclosure are provided in Enclosure 7. A non-proprietary version of Enclosure 6 is provided in Enclosure 8. GEH and EPRI, as the owners of the proprietary information, have executed the Enclosure 7 affidavits identifying that the proprietary information has been handled and classified as proprietary, is customarily held in confidence and withheld from public disclosure. GEH and EPRI request that the proprietary information in Enclosure 6 be withheld from public disclosure in accordance with the requirements of 10 CFR 2.390(a)(4).

Enclosures 9 and 11 contain proprietary information as defined by 10 CFR 2.390, which has been determined to be proprietary by Cameron International Corporation (Cameron). As the owner of the proprietary information, Cameron has executed the Enclosure 13 affidavits identifying that the proprietary information has been handled and classified as proprietary, is customarily held in confidence and withheld from public disclosure. Non-proprietary versions of Enclosures 9 and 11 are provided in Enclosures 10 and 12. Cameron requests that the proprietary information in Enclosures 9 and 11 be withheld from public disclosure in accordance with the requirements of 10 CFR 2.390(a)(4).

PSEG requests approval of this LAR by April 30, 2018, prior to completion of the 2018 refueling outage (H1R21). PSEG requests the license amendment be made effective upon NRC issuance, to be implemented within 120 days following completion of the H1R21 outage (breaker closure), during which time the LEFM system will be commissioned for operation.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated State of New Jersey Official.

This letter contains new regulatory commitments as identified in Enclosure 5.

The proposed changes have been reviewed by the Plant Operating Review Committee. If you have any questions or require additional information, please contact Mr. Brian Thomas at 856-339-2022.

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I declare under penalty of perjury that the foregoing is true and correct.

Executed on JUL 7 2017
(Date)

Respectfully,

Eric Carr
Site Vice President – Hope Creek Generating Station

cc: Mr. D. Dorman, Administrator, Region I, NRC
Ms. Carleen Parker, Project Manager, NRC
NRC Senior Resident Inspector, Hope Creek
Mr. P. Mulligan, Chief, NJBNE
PSEG Corporate Commitment Tracking Coordinator
Hope Creek Commitment Tracking Coordinator

Enclosure 1

Description and Evaluation of the Proposed Change

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1.0 SUMMARY DESCRIPTION

In accordance with the provisions of 10 CFR 50.90, “Application for Amendment of License, Construction Permit, or Early Site Permit”, and 10 CFR 50, Appendix K, “ECCS Evaluation Models,” PSEG Nuclear LLC (PSEG) requests an amendment to revise the Hope Creek Generating Station (Hope Creek) Renewed Facility Operating License (OL) No. NFP-57 and Technical Specifications (TS). Specifically, the proposed changes revise the OL and TS to implement an increase of approximately 1.6% in rated thermal power (RTP) from 3840 megawatts thermal (MWt) to 3902 MWt. The following sections are affected by these changes:

- Facility Operating License
- TS 1.0 Definitions
- TS 2.2 Limiting Safety System Settings
- TS 3/4.1.3.1 Control Rod Operability
- TS 3/4.1.4.1 Rod Worth Minimizer
- TS 3/4.3.6 Control Rod Block Instrumentation
- TS 3/4.4.1.1 Recirculation Loops
- TS 3/4.10.2 Rod Worth Minimizer

The proposed changes are based on reduced uncertainty in feedwater flow and feedwater temperature measurement that reduces the total power level measurement uncertainty. This is achieved by using the Cameron International (Cameron) Leading Edge Flow Meter Check Plus (LEFM √+) ultrasonic flow measurement instrumentation.

2.0 DETAILED DESCRIPTION

The proposed changes to the OL and TS are described in Section 2.1 below, with the associated marked-up pages included in Enclosure 2. PSEG considers this LAR as linked to the previously submitted LARs for Power Range Neutron Monitor (PRNM) LAR H15-01, and the Pressure-Temperature (P-T) Limits Curves LAR H17-02.

The PRNM LAR and this MUR LAR revise some of the same TS Reactor Trip Function and Control Rod Block function instrumentation setpoints, therefore NRC approval of the PRNM LAR is required to implement the MUR license amendment.

On October 31, 2016, PSEG reported to the NRC that P-T limits in the current Hope Creek TS were negatively impacted by the results of the evaluation of the 120° capsule which requires the P-T curves to be updated. In response to this issue, PSEG submitted a LAR on March 27, 2017, to revise the pressure-temperature limits curves. The assessment provided in Section 3.2.1 of Enclosure 6 was performed using the results of the 120° capsule at a power level of 3902 MWt.

New License Condition 2.C.(28) is proposed such that Hope Creek will operate at a thermal power level not to exceed 3840 MWt until the PRNM system license amendment request is approved by the NRC and implemented by PSEG.

Also, new License Condition 2.C.(29) is proposed such that Hope Creek will operate at a thermal power level not to exceed 3840 MWt until the P-T Limits curves license amendment request is approved by the NRC and implemented by PSEG.

Both of these new License Conditions are shown as markups to the Hope Creek Operating License in Enclosure 2.

The TS page markups in Enclosure 2 that are affected by the PRNM LAR have been revised to incorporate the TS changes proposed in the PRNM LAR.

There are no TS page markups required for the MUR LAR as a result of the P-T Limits curves LAR.

Proposed changes to the TS Bases are also described below, with marked-up pages included in Enclosure 3. The TS Bases changes are for information only and do not require NRC approval. Changes to the affected TS Bases pages will be incorporated in accordance with TS 6.15, "Technical Specification (TS) Bases Control Program."

2.1 OL and TS Changes

No.	Change	Justification
1	Page 3, Facility Operating License The value of RTP for Hope Creek Renewed Facility Operating License Number NFP-57, Section 2.C.(1), "Maximum Power Level," is revised from 3840 MWt to 3902 MWt.	The proposed RTP increase in the Hope Creek Operating License is acceptable based on the decreased uncertainty in the core thermal power calculation from using the LEFM feedwater flow measurement system, and the evaluations referenced in this License Amendment Request.
2	Page 5, Facility Operating License The value of rated thermal power feedwater temperature for Hope Creek Renewed Facility Operating License Number NFP-57, Section 2.C.(11) is revised from 329.6 °F to 331.5 °F.	Revised to maintain a differential temperature of 102 °F consistent with Enclosure 6, section 1.3.2.
3	Page 15, Facility Operating License New License Condition 2.C.(28) for Hope Creek Renewed Facility Operating License NFP-57 is proposed such that the facility will operate at a thermal power level not to exceed 3840 MWt until the Power Range Neutron Monitoring System license amendment request is approved by the NRC and implemented by PSEG.	NRC approval and implementation of the PRNM license amendment is necessary prior to operation above the 3840 MWt current licensed power level.

No.	Change	Justification
4	Page 15, Facility Operating License New License Condition 2.C.(29) for Hope Creek Renewed Facility Operating License NFP-57 is proposed such that the facility will operate at a thermal power level not to exceed 3840 MWt until the Pressure-Temperature Limits curves license amendment request is approved by the NRC and implemented by PSEG.	NRC approval and implementation of the P-T limits curves license amendment is necessary prior to operation above 3840 MWt current licensed power level.
5	Page 1-6, Definitions The definition of RTP in TS Section 1.35 is revised to increase the value of RTP from 3840 MWt to 3902 MWt.	The proposed RTP increase in the Hope TS definitions is acceptable based on the decreased uncertainty in the core thermal power calculation from using the LEFM feedwater flow measurement system, and the evaluations referenced in this License Amendment Request.
6	Page 2-4, TS 2.2 Limiting Safety System Settings Table 2.2.1-1, “Reactor Protection System Instrumentation Setpoints,” Function 2.b, “Simulated Thermal Power – Upscale 1) Flow Biased – Two Recirculation Loop Operation Trip Setpoint” is revised from the PRNM Value of ≤0.57w + 59% to ≤0.56w + 58%.	The proposed changes to the Nominal Trip Setpoints (NTSP) for the Simulated Thermal Power - Upscale functions are based on the approach described in Reference 6.1, Section F.4.2.1, "Flow Referenced APRM Trip and Alarm Setpoints." Absolute power is unchanged versus recirculation drive flow and decreased in proportion to the power uprate.
7	Page 2-4, TS 2.2 Limiting Safety System Settings Table 2.2.1-1, “Reactor Protection System Instrumentation Setpoints,” Function 2.b, “Simulated Thermal Power – Upscale 1) Flow Biased – Two Recirculation Loop Operation Allowable Value” is revised from the PRNM Value of ≤0.57w + 61% to ≤0.56w + 60%.	The proposed changes to the Allowable Values (AVs) for the Simulated Thermal Power - Upscale functions are based on the approach described in Reference 6.1, Section F.4.2.1, "Flow Referenced APRM Trip and Alarm Setpoints." Absolute power is unchanged versus recirculation drive flow and decreased in proportion to the power uprate.

No.	Change	Justification
8	Page 2-4, TS 2.2 Limiting Safety System Settings Table 2.2.1-1, “Reactor Protection System Instrumentation Setpoints,” Function 2.b, “Simulated Thermal Power – Upscale 2) Flow Biased – Single Recirculation Loop Operation Trip Setpoint” is revised from the PRNM Value of ≤0.57(w-10.6%) + 59% to ≤0.56(w-10.8%) + 58%.	The proposed changes to the NTSP for the Simulated Thermal Power - Upscale functions are based on the approach described in Reference 6.1, Section F.4.2.1, "Flow Referenced APRM Trip and Alarm Setpoints." Absolute power is unchanged versus recirculation drive flow and decreased in proportion to the power uprate.
9	Page 2-4, TS 2.2 Limiting Safety System Settings Table 2.2.1-1, “Reactor Protection System Instrumentation Setpoints,” Function 2.b, “Simulated Thermal Power – Upscale 2) Flow Biased – Single Recirculation Loop Operation Allowable Value” is revised from the PRNM Value of ≤0.57(w-9%) + 61% to ≤0.56(w-9%) + 60%.	The proposed changes to the AVs for the Simulated Thermal Power - Upscale functions are based on the approach described in Reference 6.1, Section F.4.2.1, "Flow Referenced APRM Trip and Alarm Setpoints." Absolute power is unchanged versus recirculation drive flow and decreased in proportion to the power uprate.
10	Page 3/4 1-4 LCO 3.1.3.1 Control Rod Operability Note ***** applicability is revised from 8.6% to 8.5% rated thermal power.	Revised to maintain the rated thermal power value in terms of absolute power, consistent with Enclosure 6, section 5.3.8.
11	Page 3/4 1-16 LCO 3.1.4.1 Rod Worth Minimizer LCO 3.1.4.1 Applicability is revised from 8.6% to 8.5% rated thermal power.	Revised to maintain the rated thermal power value in terms of absolute power, consistent with Enclosure 6, section 5.3.8.
12	Page 3/4 3-59, TS 3.3.6 Control Rod Block Instrumentation Table 3.3.6-2, Control Rod Block Instrumentation Setpoints,” Function 2.a “APRM Simulated Thermal Power – Upscale 1) Flow Biased – Two Recirculation Loop Operation Trip Setpoint” is revised from the PRNM Value of ≤0.57w + 54% to ≤0.56w + 53.1%.	The proposed changes to the NTSP for the Simulated Thermal Power - Upscale functions are based on the approach described in Reference 6.1, Section F.4.2.1, "Flow Referenced APRM Trip and Alarm Setpoints." Absolute power is unchanged versus recirculation drive flow and decreased in proportion to the power uprate.

No.	Change	Justification
13	<u>Page 3/4 3-59, TS 3.3.6 Control Rod Block Instrumentation</u> Table 3.3.6-2, "Control Rod Block Instrumentation Setpoints," Function 2.a "APRM Simulated Thermal Power – Upscale 1) Flow Biased – Two Recirculation Loop Operation Allowable Value" is revised from the PRNM Value of ≤0.57w + 56% to ≤0.56w + 55.1%.	The proposed changes to the AVs for the Simulated Thermal Power - Upscale functions are based on the approach described in Reference 6.1, Section F.4.2.1, "Flow Referenced APRM Trip and Alarm Setpoints." Absolute power is unchanged versus recirculation drive flow and decreased in proportion to the power uprate.
14	<u>Page 3/4 3-59, TS 3.3.6 Control Rod Block Instrumentation</u> Table 3.3.6-2, "Control Rod Block Instrumentation Setpoints," Function 2.a "APRM Simulated Thermal Power – Upscale 2) Flow Biased – Single Recirculation Loop Operation Trip Setpoint" is revised from the PRNM Value of ≤0.57(w-10.6%) + 54% to ≤0.56(w-10.8%) + 53.1%.	The proposed changes to the NTSP for the Simulated Thermal Power - Upscale functions are based on the approach described in Reference 6.1, Section F.4.2.1, "Flow Referenced APRM Trip and Alarm Setpoints." Absolute power is unchanged versus recirculation drive flow and decreased in proportion to the power uprate.
15	<u>Page 3/4 3-59, TS 3.3.6 Control Rod Block Instrumentation</u> Table 3.3.6-2, "Control Rod Block Instrumentation Setpoints," Function 2.a "APRM Simulated Thermal Power – Upscale 2) Flow Biased – Single Recirculation Loop Operation Allowable Value" is revised from the PRNM Value of ≤0.57(w-9%) + 56% to ≤0.56(w-9%) + 55.1%.	The proposed changes to the AVs for the Simulated Thermal Power - Upscale functions are based on the approach described in Reference 6.1, Section F.4.2.1, "Flow Referenced APRM Trip and Alarm Setpoints." Absolute power is unchanged versus recirculation drive flow and decreased in proportion to the power uprate.
16	<u>Page 3/4 4-1 LCO 3.4.1.1 Recirculation System</u> LCO 3.4.1.1 Action a.1.b is revised to change thermal power during single loop operation from 60.86% to 59.89%.	Thermal power rescaled to maintain the rated thermal power value in terms of absolute power, consistent with Reference 6.1, Section 5.2 and Enclosure 6 Section 1.2.1.

No.	Change	Justification
17	Page 3/4 4-2a SR 4.4.1.1.1 Recirculation System SR 4.4.1.1.1.a is revised to change thermal power during single loop operation from 60.86% to 59.89%.	Thermal power rescaled to maintain the rated thermal power value in terms of absolute power, consistent with Reference 6.1, Section 5.2 and Enclosure 6, Section 1.2.1.
18	Page 3/4 10-2, LCO 3.10.2 Rod Worth Minimizer LCO 3.10.2 Applicability is revised from 8.6% to 8.5% rated thermal power.	Revised to maintain the rated thermal power value in terms of absolute power, consistent with Enclosure 6, Section 5.3.8.

2.2 TS Bases Changes (Information Only)

No.	Change	Justification
1	Page B 3/4 1-2a, LCO 3/4 1.3 Control Rods Bases LCO 3/4.1.3 Bases are revised from 8.6% to 8.5% rated thermal power.	Revised to maintain the rated thermal power value in terms of absolute power, consistent with Enclosure 6, Section 5.3.8.
2	Page B 3/4 1-3, LCO 3/4 1.4 Control Rod Program Controls LCO 3/4.1.4 Bases are revised from 8.6% to 8.5% rated thermal power.	Revised to maintain the rated thermal power value in terms of absolute power, consistent with Enclosure 6, Section 5.3.8.
3.	Page B 3/4 4-1 (Insert 4), LCO 3/4.4.1 Recirculation System Insert 4 of LCO 3/4.4.1 (Added by the PRNM LAR Supplement, Reference 6.20) is revised to reflect the MUR changes to the recirculation system two loop operation and single loop operation setpoints.	Revised to account for power and flow offsets during single loop operation based on the thermal power optimization (TPO), consistent with Enclosure 6, Section 5.3.7 and Table 5-1.

2.3 Procedure Changes

As discussed in Section 3.2.4, “Response to Criteria 1” of this enclosure, a licensee commitment is established in Enclosure 5 which pertains to requirements, required actions, and associated allowed outage times when the LEFM is not fully functional. The plant procedures will be revised as appropriate to implement this licensee commitment. The specific procedural changes are not included in this LAR, but will be controlled through the 10 CFR 50.59 process.

3.0 TECHNICAL EVALUATION

3.1 Background and General Approach

10 CFR 50, Appendix K, "ECCS Evaluation Models", Paragraph I.A, "Sources of Heat During the LOCA," requires that emergency core cooling system (ECCS) evaluation models assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level to allow for instrumentation error.

Using the Cameron LEFM √+ System at Hope Creek reduces uncertainty in feedwater flow measurement, and subsequently reduces the total power level measurement uncertainty. As described in Section 3.2, "LEFM Feedwater Flow Measurement and Core Thermal Power Uncertainty" of this enclosure, the core thermal power measurement uncertainty is a maximum of 14.59 MWt (0.374% of the MUR uprate power level of 3902 MWt).

As summarized in Section 3.4.1, "Summary of Analyses" of this enclosure and Enclosure 6, the ECCS evaluation models and other plant safety analyses either assume an uncertainty of 2% of the CLTP (3840 MWt) or have been evaluated for operation at 3902 MWt. The LEFM system supports an increase in RTP to the requested 3902 MWt or approximately 1.6% of the CLTP. The sum of the requested RTP value (3902 MWt) and the maximum uncertainty value (14.59 MWt) is bounded by 102% of the CLTP value assumed in the plant safety analyses.

PSEG has evaluated the effects of an approximately 1.6% increase in RTP using an approach developed by GE-Hitachi Nuclear Energy (GEH) and approved by the NRC as documented in NEDC-32938P-A Revision 2, (Reference 6.1). These evaluations are summarized in Section 3.4.1 of this enclosure, and described in detail in GEH Document NEDC-33871P, "Safety Analysis Report for Hope Creek Generating Station Thermal Power Optimization," Revision 0, (Enclosure 6).

Enclosure 6 also includes Appendix A which lists the limitations from the Safety Evaluation for Licensing Topical Report (LTR) NEDC-33173P, "Applicability of GE Methods to Expanded Operating Domains" (Reference 6.17); and Appendix B which lists the limitations from the Safety Evaluation for LTR NEDC-33075P, Revision 8, "General Electric Boiling Water Reactor Detect and Suppress Solution – Confirmation Density" (Reference 6.18).

The scope and content of the evaluations performed and described in this request comply with the guidance contained in NRC Regulatory Issue Summary (RIS) 2002-03 (Reference 6.2). Enclosure 4 provides a cross-reference between the contents of this request and the guidance in RIS 2002-03.

3.2 LEFM Feedwater Flow Measurement and Core Thermal Power Uncertainty

3.2.1 LEFM Feedwater Flow and Temperature Measurement

Hope Creek will use the Cameron LEFM √+ ultrasonic multi-path, transit time flow meter. This LEFM system will replace the currently installed CE Nuclear Power Cross Flow Ultrasonic Flow Meter and resistance temperature detector (RTD) temperature indication, to provide feedwater flow input for the plant thermal heat balance calculation. The currently installed feedwater flow venturis will be used if the LEFM is not functional. The LEFM system uses ultrasonic transit time principles to determine fluid velocity and sound velocity. This flow measurement method is described in Caldon topical reports ER-80P, Revision 0 (Reference 6.3), ER-157P, Revision 8 and Revision 8 Errata (Reference 6.4). These topical reports were approved by the NRC in SERs dated March 8, 1999 (Reference 6.5) and August 16, 2010 (Reference 6.6).

PSEG has provided Hope Creek specific Cameron document ER-1123 (Enclosure 9), which is the analysis of the uncertainty contribution of the LEFM √+ System in its Normal mode of operation as well as when operating in its Maintenance mode to the overall thermal power uncertainty for Hope Creek. This report contains detailed calculations based on topical reports ER-80P and ER-157P, Revision 8 and Revision 8 Errata. Cameron document ER-972, Revision 2 (Reference 6.24) contains a detailed cross reference of the sections in the Cameron topical reports to the applicable sections in the plant-specific report ER-1123.

In approving topical reports ER-80P and ER-157P, the NRC established criteria that each licensee referencing these topical reports must address. PSEG's response to those criteria is provided in Section 3.2.4 of this enclosure.

The LEFM √+ System uncertainty analysis provided in Enclosure 9 is a bounding analysis for Hope Creek and was completed following calibration of the LEFM spool piece. Cameron document ER-1132 (Enclosure 11) provides the calibration and uncertainty analysis performed on the Hope Creek LEFM flow element. The commissioning tests for the Hope Creek LEFM √+ System will confirm that the time measurement uncertainties are within the bounding values used in the analysis.

The LEFM instrumentation is not safety-related. The LEFM system was designed and manufactured per Cameron's Quality Assurance Program.

The LEFM √+ System consists of a single measurement spool piece meter to be installed in the 30-inch common feedwater header, two transmitter signal processing units and two redundant central processing units (CPU). The measurement spool piece contains 16 ultrasonic, multi-path, transit time transducers grouped into the two planes of eight transducers each, two 4-wire RTDs, and two pressure transmitters.

The LEFM √+ System performs automatic continuous self-checking of the transducer signals and the calculation results. This testing provides verification that the digital circuits are operating correctly and the LEFM √+ System is within its specified accuracy envelope. These processes can identify failure conditions that will cause the LEFM to switch from the Normal mode to the Maintenance mode or to the Fail mode. Validated LEFM data including calculated results, status, and signal process information is sent to the plant computer at regular intervals.

The plant computer will provide an alarm upon a change in LEFM system status. An alarm is provided for a sustained loss of data between the LEFM and the plant computer. Core thermal power calculations automatically revert to the calibrated venturi output when the plant computer does not have a valid LEFM signal.

The LEFM √+ System has two operating modes (Normal and Maintenance) and a Fail mode.

- Normal : The LEFM √+ System measures the average flow of two independent LEFM √ subsystems, where each LEFM √ subsystem consists of four acoustic paths that are summed into the eight paths that comprise the LEFM √+ system. The LEFM √+ System Normal is displayed when the feedwater flow, temperature, and header pressure signals are normal and operating within design limits. Calculated power level uncertainty associated with the LEFM flow measuring system in this condition is 0.34%. The plant can operate at ≤ 3902 MWt as discussed in Section 3.2.3 of this enclosure.

- Maintenance: The Maintenance mode refers to the state when any LEFM √+ System has only one of the two LEFM√ subsystems fully operational, which results in flow computation based on the fully operational LEFM √ subsystem. A LEFM √+ System Alert alarm indicates a loss of system redundancy and the system shifts from the Normal mode to the Maintenance mode of operation. Typically, this occurs due to a malfunction of a single path or plane. The calculated power level uncertainty associated with the LEFM flow measuring system in this condition is 0.66%. The plant can operate indefinitely at ≤ 3889 MWt with only one LEFM √ subsystem operational as discussed in Section 3.2.3 of this enclosure. Power will be reduced to ≤ 3889 MWt (CLTP) within 72 hours if LEFM functionality cannot be restored to the Normal mode.

- Fail: A LEFM √+ System Fail alarm indicates a loss of function. Power will be reduced to ≤ 3840 MWt (CLTP) within 72 hours if LEFM functionality cannot be restored to either the Normal or Maintenance mode. If the plant experiences a power decrease below 3840 MWt (98.4% of RTP) with the LEFM in the Fail mode during the 72 hour allowed outage time, the maximum permitted power level will be 3840 MWt until the LEFM is restored to either Normal or Maintenance mode operation.

Justification for the proposed power level reductions is provided in Section 3.2.3 of this enclosure. Justification for the proposed 72-hour allowed outage time is provided in Section 3.2.4 of this enclosure.

3.2.2 Plant Implementation

The Hope Creek LEFM system is not currently installed. The installation is planned to be completed during the Spring 2018 refueling outage. The LEFM measurement spool piece will be installed in the 30 inch diameter common feedwater header, downstream of the 6th stage high pressure feedwater heaters. Drawings showing installation location are provided in Enclosure 15.

The LEFM system will be installed and commissioned per appropriate Cameron installation and test procedures. Final commissioning testing is described in Cameron's "Commissioning Procedure for LEFM® √+ C, M, 280Fi and 880 Series Systems" (Reference 6.7).

3.2.3 LEFM and Core Thermal Power Measurement Uncertainty and Methodology

Enclosure 9 provides an analysis of the LEFM √+ System uncertainty contributions, when operating in the Normal mode and Maintenance mode, to the overall calculated thermal power uncertainty. At Hope Creek with the system operating in the LEFM √+ mode, calculated core thermal power uncertainty due to the LEFM system is 0.34%. In the Maintenance mode, calculated core thermal power uncertainty due to the LEFM system is 0.66%. These uncertainties were calculated using the methodology described in Reference 6.4, which was approved by the NRC in Reference 6.6. These uncertainties, when combined with other uncertainties applicable to the heat balance calculation, yield a total thermal power uncertainty of 0.374% and 0.694% respectively, as demonstrated in the heat balance uncertainty calculation (Enclosure 14).

The MUR allows a licensed power level that maintains margin to 102% of CLTP. In Enclosure 14, 102% of 3840 MWt (3916.8 MWt) was used as a maximum value when determining the MUR power uprate value. This results in the following thermal power uncertainties and proposed power levels. The method used in performing the above calculation is based on NEDC-31336P-A, "General Electric Instrument Setpoint Methodology."

- With the LEFM system operating in the Normal mode, the heat balance calculation has an uncertainty of 14.59 MWt. This results in a power level of 3916.8.MWt – 14.59 MWt = 3902.21 MWt. The proposed power level in the Normal mode is rounded down to 3902 MWt. Therefore, the requested increase in power is approximately 1.6% above the CLTP of 3840 MWt.

- With the LEFM system operating in the Maintenance mode, the heat balance calculation has an uncertainty of 26.99 MWt. This results in a power level of 3916.8 MWt – 26.99 MWt = 3889.81 MWt. The proposed power level in the Maintenance mode is rounded down to 3889 MWt.

A revised heat balance calculation will be added to the plant computer to support feedwater input from the LEFM system or the existing venturi flow nozzles.

Caldon Topical Report ER-157P, Revision 8 (Reference 6.4) states that the redundancy inherent in the two measurement planes of an LEFM √+ System also makes this system more resistant to component failures when compared to the LEFM √ System. For any single component failure, continued operation at a power level greater than 3840 MWt can be justified with the LEFM √+ System since the system operating with the failure is no less accurate than the LEFM √ System operation. The NRC SER approving ER-157P, Revision 8 (Reference 6.6) required licensees referencing ER-157P, Revision 8 to ensure compliance with two limitations and conditions:

- 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.

- The only mechanical difference that potentially affects Topical Report ER-157P, Revision 8 statement above is that the LEFM √+ System has 16 transducer housing interfaces with the flowing water, whereas the LEFM √ System has 8. Consequently, a LEFM √+ System operating with a single failure that is assumed to disable one plane of transducers is not identical to an LEFM √ System. Although the effect on hydraulic behavior is expected to be negligible, this must be acceptably quantified if a license 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 the Alden Research Laboratory.

In the event the LEFM system is non-functional (Fail mode), the heat balance calculation will use the existing feedwater venturi flow nozzles and existing feedwater temperature instrumentation until the LEFM system is returned to a functional status (either Normal or Maintenance mode). To ensure that the venturi based heat balance calculation is consistent with the LEFM system based heat balance calculation, the venturi based flow rate and feedwater temperature RTDs will be normalized to the pre-failure LEFM system readings.

The loss of the data link between the LEFM system and the plant computer (beyond that associated with anticipated data flow interruptions) or a plant computer failure will require reducing core thermal power to ≤ 3840 MWt within 72 hours. It is conservative to limit the power within 72 hours to this level until the LEFM system is returned to functional status (either Normal or Maintenance mode).

Cameron reports ER-1123 (Enclosure 9) and ER-1132 (Enclosure 11) identify the uncertainties associated with LEFM operation in the Normal mode and Maintenance mode, including meter factor uncertainties specific to Hope Creek. These uncertainties were established by the calibration tests performed at Alden Research Laboratory. The impact of a failure disabling one plane of transducers on the LEFM system installed at Hope Creek has been quantified with an uncertainty of 0.694%. The associated increase in uncertainty from 0.374% to 0.694% results in a maximum allowable power level for this condition of 3889 MWt.

Hope Creek has satisfied the two limitations and conditions specified in the NRC SER for licensees referencing Caldon Topical Report ER-157P, Revision 8 as discussed above and in Section 3.2.4 under Criterion 1 and 7.

3.2.4 Disposition of NRC Criteria for Use of LEFM Topical Reports

In approving Topical Reports ER-80P and ER-157P, the NRC established criteria each licensee referencing these Topical Reports must address. The nine criteria are listed below along with a discussion of how Hope Creek is or will be satisfying them.

Criterion 1

Discuss maintenance and calibration procedures that will be implemented with the incorporation of the LEFM, including processes and contingencies for inoperable LEFM instrumentation and the effect on thermal power measurements and plant operation.

Response to Criterion 1

Maintenance and Calibration Procedures

> License amendment implementation will include developing the necessary procedures and documents required for maintenance and calibration at the uprated power level using the LEFM √+ System. The initial preventive maintenance scope and frequency will be based on vendor recommendations. This will ensure that the LEFM system is properly maintained and calibrated. Work on the LEFM will be performed by qualified site personnel.
>
> For instrumentation other than the LEFM system that contributes to the thermal power heat balance computation, maintenance and calibration is performed periodically using existing Hope Creek procedures. Instrument channel accuracy, drift, calibration error and instrument error were accounted for within the thermal power uncertainty calculation.
>
> The LEFM system software and the plant computer software configuration will be maintained using Hope Creek procedures, which include verification and validation of changes to software configuration. Hardware configuration associated with the LEFM system and the instrumentation that contributes to the heat balance calculation is maintained per Hope Creek configuration control procedures.

Hope Creek programs and procedures addressing corrective actions, reporting deficiencies, and receiving and evaluating manufacturer's deficiency reports are discussed in Section 3.2.5 "Deficiencies and Corrective Actions" of this enclosure.

LEFM Non-functionality and the Effect on Thermal Power Measurements and Plant Operations

The redundancy inherent in the two measurement planes of the LEFM system as described in Enclosure 9 makes the system tolerant to component failures. Continuously operating online self-diagnostic testing is provided to verify that the digital circuits are operating correctly and within the design basis uncertainty limits. LEFM data link and system malfunctions will result in control room alarms to alert the operators to changes in LEFM instrumentation status. In these cases, appropriate procedural actions will be applied.

Additionally, if the interface between the LEFM system and the plant computer has failed, the LEFM will be considered non-functional and the appropriate procedural actions will be applied. LEFM functionality requirements and the required actions and allowed outage times when the LEFM is not fully functional, will be added to plant procedures prior to raising power above the CLTP (refer to Enclosure 5, Item 1). The NRC has previously approved the use of the Maintenance mode at Shearon Harris (Reference 6.13) for operation at a power level greater than the CLTP, but less than MUR uprated power.

An allowed outage time of 72 hours is proposed for operation at any power level above the CLTP of 3840 MWt with the LEFM not fully functional. The basis for the proposed 72-hour allowed outage time follows:

1. If the LEFM system or a portion of the system becomes non-functional, operators will be promptly alerted by a control room alarm. With the LEFM non-functional, feedwater flow input to the core thermal power calculation would then be provided by the existing feedwater flow venturis and temperature input would be provided by the existing RTDs. The feedwater flow venturis and RTDs will be normalized to the last valid data from the LEFM system. With a portion of the LEFM non-functional (Maintenance mode), the LEFM will continue to provide the input into the core thermal power calculation.

2. The 72-hour allowed outage time (AOT) for the LEFM flow meter prior to reducing power is acceptable because:

    a. The existing feedwater flow nozzle-based signals will be calibrated to the last valid data from the LEFM system during this period. Any slight drift of the feedwater flow nozzle measurements due to fouling would result in a higher than actual indication of feedwater flow and an overestimation of the calculated calorimetric power level. This is conservative since the reactor will actually be operating below the calculated power level. A sudden de-fouling event during the 72-hour inoperability period is unlikely and any significant sudden de-fouling would be detected by other plant parameters. Calibration data for the venturi flow transmitters and plant historian data show that the venturis have remained stable since implementation of EPU in 2008. No significant fouling or de-fouling events have been observed.

b. The LEFM is operating in the Maintenance mode with a valid LEFM measured flow rate.

3. Industry experience for similar BWRs shows that the instrument drift associated with venturi feedwater flow measurements are insignificant over a 72-hour time period. In Reference 6.3, Table A-1 provides the systemic error associated with feedwater flow nozzle differential pressure as approximately 1.0% over an operating cycle. Thus, over a 72-hour period this would have an insignificant effect on the feedwater flow measurement.

The 72-hour allowed outage time begins when the alarm is received in the control room. A control room alarm response procedure will be developed providing guidance to the operators for initial alarm diagnosis. Methods to determine LEFM √+ System status and the cause of alarms are described in Cameron documentation. Cameron documentation will be used to develop the specific procedures for operators and maintenance response actions. Note that the NRC has previously approved power uprate applications with an allowed outage time up to 72 hours (References 6.8 through 6.10).

Enclosure 5, Item 1 establishes a regulatory commitment to provide procedural guidance to the operators regarding the required actions when the LEFM system is not in the Normal mode.

Criterion 2

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

Response to Criterion 2

Criterion 2 is not applicable to Hope Creek. The LEFM is not currently installed at Hope Creek.

Criterion 3

Confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation is based on the 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 ultrasonic flow measurement instrumentation installations for comparison.

Response to Criterion 3

The LEFM system uncertainty calculation is based on the American Society of Mechanical Engineers (ASME) PTC 19.1-2013, Part I Measurement Uncertainty, as described in Enclosure 9. This LEFM system uncertainty calculation methodology is based on the square root of the sum of the squares (SRSS) calculation, as described in Reference 6.4.
The Hope Creek heat balance uncertainty calculation (Enclosure 14) was completed per NEDC-31336, “General Electric Instrument Setpoint Methodology”.

Criterion 4

For plants where the ultrasonic meter (including LEFM) was not installed with flow elements calibrated to the site-specific piping configuration (i.e., flow profiles and meter factors not representative of the plant specific installation), additional justification should be provided 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, confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

Response to Criterion 4

The calibration factors for the Hope Creek ultrasonic LEFM flow meters were established by tests conducted at Alden Research Laboratory. These tests were performed on a full-scale model of the Hope Creek hydraulic geometry. The impact of the plant-specific installation factors of the feedwater flow measurement uncertainty is discussed in Cameron Report ER-1123, (Enclosure 9) and Cameron Report ER-1132, Rev.1 (Enclosure 11). The test configurations modeled the portion of piping upstream of the LEFM spool piece. The test configurations (ER-1132 Rev 1, Figure 2.1) can be compared to the plant drawings (Enclosure 15). There is no significant difference between the Hope Creek feedwater piping configuration and the model used at Alden Research Laboratory.

Criterion 5

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.

Response to Criterion 5

Plant-specific justification for continued operation at the pre-failure level for a pre-determined time, and the required actions if that time is exceeded (i.e., power reduction) is provided in the response to Criterion 1 above.

Criterion 6

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 a license wishes to operate using the degraded CheckPlus at an increased uncertainty.

Response to Criterion 6

The Alden Labs Test quantified the uncertainty of the LEFM √+ System operating with a single failure (Maintenance mode) using a full scale model of the Hope Creek piping geometry. The LEFM √+ System total uncertainty while operating in the Maintenance mode was evaluated with the results documented In Enclosure 9 and Section 3.2.3 listed above.

Criterion 7

An applicant with a comparable geometry can reference the Section 3.2.1 finding (of Reference 6.6) to support a conclusion that downstream geometry does not have a significant influence on CheckPlus calibration. However, CheckPlus test results do not apply to a Check and downstream effects with use of a CheckPlus with disabled components that make the CheckPlus comparable to a Check must be addressed. An acceptable method is to conduct applicable Alden Laboratory tests.

Response to Criterion 7

The NRC has determined in Reference 6.21 that for conditions in which the LEFM √+ System is operating with one or more transducers out of service, the effect of downstream piping should be addressed if the separation distance from the meter transducers to the downstream piping change is less than five pipe diameters. At Hope Creek, the LEFM flow meter is installed upstream of an elbow in the feedwater header, and the distance from meter transducers to the downstream change in piping, i.e., the piping elbow, is 11 feet 3 inches and is greater than five pipe diameters. Therefore, it is concluded that the downstream geometries for Hope Creek do not have a significant influence on Maintenance mode calibration.

Criterion 8

An applicant that requests a MUR with the upstream flow straightener configuration discussed in Section 3.2.2 (footnote 1) (of Reference 6.6) should provide justification for claimed CheckPlus uncertainty that extends the justification provided in Reference 17 (of footnote 1) (of Reference 6.6). Since the Reference 17 evaluation does not apply to the Check, a comparable evaluation must be accomplished if a Check is to be installed downstream of a tubular flow straightener.

Response to Criterion 8

Criterion 8 is not applicable to Hope Creek. Hope Creek does not have flow straighteners upstream of the LEFM spool piece installation.

Criterion 9

An applicant assuming large uncertainties in steam moisture content should have an engineering basis for the distribution of the uncertainties or, alternatively, should ensure that their calculations provide margin sufficient to cover the differences shown in Figure 1 of Reference 18 (of footnote 1) (of Reference 6.6)

Response to Criterion 9

Criterion 9 is not applicable to Hope Creek. Hope Creek conservatively assumes no moisture content in the heat balance uncertainty calculation (Enclosure 14). This approach is consistent with that described in Section 3.2.3 of Reference 6.6.

3.2.5 Deficiencies and Corrective Actions

Cameron has procedures to notify users of important LEFM deficiencies. Hope Creek has processes for addressing manufacturer's deficiency reports. Such deficiencies are documented and dispositioned in the Hope Creek corrective action program.

Problems with plant instrumentation identified by Hope Creek personnel are also documented and dispositioned in the Hope Creek corrective action program. Deficiencies associated with the vendor's processes or equipment will be reported to the vendor to support corrective actions.

3.2.6 Reactor Power Monitoring

Plant procedures provide requirements for monitoring and controlling reactor power in compliance with the TS.

3.3 Evaluation of OL and TS Changes

The proposed changes described in Section 2.1, "OL and TS Changes" of this enclosure are evaluated below.

<u>Changes to RTP</u>

The proposed RTP increase in the Hope Creek OL and TS definitions is acceptable based on the decreased uncertainty in the core thermal power calculation from using the LEFM feedwater flow measurement system, and the evaluations provided in this License Amendment Request.

<u>Changes to Limiting Safety System Settings and Control Rod Block Instrumentation</u>

The proposed changes to the Nominal Trip Setpoints (NTSP) and Allowable Values (AVs) for the Simulated Thermal Power - Upscale functions are based on the approach described in Reference 6.1, Section F.4.2.1, "Flow Referenced APRM Trip and Alarm Setpoints." The Simulated Thermal Power NTSPs and AVs, for both two-loop operation and single loop operation, are unchanged in units of absolute core thermal power versus recirculation drive flow. Because these values are expressed in percent of RTP, they decrease in proportion to the power uprate.

<u>Changes to Control Rod Operability and Rod Worth Minimizer Low Power Setpoint</u>

The proposed change to the Rod Worth Minimizer Low Power Setpoint is based on the approach described in Reference 6.1, Section F.4.2.9, "Rod Worth Minimizer Low Power Setpoint." The value of this setpoint is maintained in terms of absolute power, and its value relative to licensed power is revised accordingly.

<u>Change to Partial Feedwater Heating</u>

The proposed change to the value of feedwater temperature at rated thermal power is based on maintaining the current feedwater temperature differential reduction identified in Enclosure 6, Section 1.3.2.

Change to Recirculation Single Loop Operation Rated Thermal Power

The proposed change to the value of rated thermal power is based on rescaling to maintain the absolute thermal power value when operating the recirculation system with one loop in service, consistent with Reference 6.1., Section 5.2 and Enclosure 6, Section 1.2.1.

3.4 Additional Considerations

3.4.1 Summary of Analyses

TOPIC	CONCLUSION	ENCLOSURE 6 SECTION
Normal Plant Operating Conditions	MUR power uprate is accomplished by increasing core flow along previously established MELLLA rod line.	Section 1
Reactor Core and Fuel Performance	Reactor core and fuel design is adequate for operation at MUR uprated conditions.	Section 2
Reactor Coolant and Connected Systems	Overpressure protection, fracture toughness, structural, and piping evaluations are acceptable.	Section 3
Engineered Safety Features	Acceptable based on previous analyses at 102% of current licensed power.	Section 4
Instrumentation and Control	Current instrumentation is acceptable. Changes to some TS values are necessary.	Section 5 and Appendix B
Electrical Power and Auxiliary Systems	Minor increases in normal power system loads. Emergency power systems are unaffected. Auxiliary systems are acceptable.	Section 6
Power Conversion Systems	The high pressure (HP) turbine is being modified to provide flow margin. The #5 feedwater heaters are being re-rated.	Section 7
Radwaste and Radiation Sources	Small increase in normal operation radiation levels and effluents. Accident consequences are bounded by previous evaluations.	Section 8
Reactor Safety Performance Evaluations	Design basis accidents are bounded by previous evaluations. Special events meet acceptable criteria.	Section 9 and Appendix A
Other Evaluations	All evaluation results are acceptable.	Section 10

3.4.2 Adverse Flow Effects

Industry experience has revealed that power uprates can cause flow conditions that can lead to steam dryer and main steam line (MSL) valve degradation. This experience has been associated with extended power uprates (EPU) and not with smaller power uprates such as an MUR.

Hope Creek has performed steam dryer baseline examinations per Boiling Water Reactor Vessel Internals Project (BWRVIP)-139 (Reference 6.11). Re-examinations of the steam dryer have been conducted per BWRVIP-139-A (Reference 6.12). An independent steam dryer stress analysis (Reference 6.23) was performed at 3906 MWt. The analysis results indicate that steam dryer loads and stresses increase slightly due to the MUR uprate conditions. The available margin to minimizing the potential for fatigue failure is defined by the minimum alternating stress ratio (MASR). Although the MASR remains above 1.0 for all locations there are a relatively small number of locations below 2.0. PSEG is proposing to monitor the locations with a MASR below 2.0 as follows:

- Prior to reaching MUR conditions (baseline) and following the first scheduled refueling outage after reaching MUR conditions, a visual inspection shall be conducted of all accessible steam dryer locations with a MASR less than 2.0. One location with a MASR less than 2.0 will not be inspected due to accessibility and dose considerations. This location has an MASR of 1.74 that is considerably higher than the most limiting locations covered under the inspection plan. The inspections will be performed in accordance with BWRVIP-139-A guidelines.

- Moisture carryover shall be measured upon achieving 100% MUR rated power (baseline), and weekly for the first operating cycle after MUR implementation.

Two new Regulatory commitments are provided in Enclosure 5 (Items 7 and 8) for the above. Any adverse flow effects on steam dryer structural integrity would be identified by these inspections.

The generic evaluation for the main steam isolation valves (MSIVs) provided in Reference 6.1, Appendix J.2.3.7, “MSIVs and Main Steam Line Flow Restrictors,” is applicable to Hope Creek. The requirements for the MSIVs remain unchanged for MUR power uprate conditions. All safety and operational aspects of the MSIVs are within previous evaluations.

Based on the above, no adverse flow induced vibration effects are expected as a result of the MUR power uprate.

3.4.3 Plant Modifications

The evaluations performed to support the MUR power uprate identified the following additional modifications to plant systems to support operation at 3902 MWt:

- <u>HP Turbine Modification</u>

  The Hope Creek main turbine generator (T/G) is being modified to provide more flow margin. The HP first stage inlet nozzle and second stage through fourth stage diaphragms are to be modified. The modified configuration will provide excess capacity for TPO. The excess capacity ensures that the T/G can meet rated conditions for continuous operating capability with allowances for variations in flow coefficients from expected values, manufacturing tolerances, and other variables that may affect the flow-passing capability of the unit.

- <u>Replacement of Reactor Dome Pressure Transmitter</u>

  The Cameron analysis assumes a maximum of 15 psi total uncertainty on the reactor dome pressure input to the heat balance calculation. Hope Creek’s current reactor dome pressure loop exceeds this uncertainty. The existing Rosemount 1151 transmitter is being replaced with a Rosemount 3153N transmitter to reduce the loop uncertainty below 15 psi.

- Rerate of #5 Feedwater Heaters

  The #5 Feedwater Heaters are being re-rated for higher shell temperatures in accordance with applicable codes and standards. The shell design temperature will increase from 380° F to 400° F.

Software changes to the plant computer are required to support the interface with the LEFM system for operation above the CLTP limit of 3840 MWt. Setpoint or alarm point changes are also required.

These modifications will be made per the requirements of 10 CFR 50.59, "Changes, Tests, and Experiments," and will be implemented prior to, or concurrently with the proposed power uprate implementation (refer to Enclosure 5, Item 6).

3.4.4 Instrument Setpoint Methodology

The determination of required TS changes, as described in Section 2.0 of this enclosure, is based on the GEH setpoint methodology. Reference 6.1 used approved GEH setpoint methodology to determine these values. Each actual trip setting is established to preclude inadvertent initiation of the protective action, while assuring adequate allowances for instrument accuracy, calibration, and drift applicable under normal operating and design basis accident conditions.

Hope Creek addressed Technical Specification Task Force (TSTF) Traveler TSTF-493 (Reference 6.14) for the affected TS instrumentation in previously submitted PRNM LAR H15-01. New License Condition 2.C.(28) is provided in Enclosure 2 that the PRNM LAR must be approved by the NRC and implemented prior to operation above 3840 MWt.

3.4.5 Grid Stability Studies

Grid stability studies were performed for Hope Creek operation at a bounding electrical power output of 1320 MWe. These results bound operation at the proposed MUR power level of 3902 MWt.

The PJM studies were performed using generator operating curves defined in the Artificial Island Operating Guide (AIOG) A-5-500-EEE-1686 (Reference 6.22). These curves are not modified for operation at MUR power levels. Since Hope Creek will continue to operate within the existing generator curves, the existing PJM studies are bounding.

Grid stability is a function of the overall grid configuration with all the lines and equipment connected, and the balance of the generation compared to the grid loading. The Hope Creek contribution to grid stability is determined by the generator electrical output and the turbine, generator and main transformer characteristics which are all fixed by the equipment design.

Hope Creek is operated in close proximity with the PSEG Nuclear Salem Units 1 and 2 generating stations. Hope Creek has been analyzed for stability for the following transients, provided the station is operated per the AIOG:

- Loss of the Hope Creek Generator,
- Loss of the most critical generating unit on the grid,
- Loss of the most critical transmission line.

Electrical component ratings and design parameters are kept up to date in the AIOG to assure system stability. Sufficient margin exists for operation at 3902 MWt since all the equipment will remain within its nameplate rating. Hope Creek has determined that the MUR power uprate to 3902 MWt will have no significant effect on grid stability or reliability and no modifications to the transmission system are required.

3.4.6 Operator Training, Human Factors, and Procedures

Operator response to plant transients and accidents is unaffected by the proposed power uprate changes. There is no reduction in time for required operator actions. No new manual operator actions were created and no existing manual actions were automated. Necessary operating procedure revisions (including Emergency Operating Procedures and Abnormal Operating Procedures) will be completed prior to implementation of the proposed changes (Refer to Enclosure 5, Item 2). The plant simulator will be modified for the uprated conditions and the changes validated per the plant configuration control processes (refer to Enclosure 5, Item 3). Operator training will be completed prior to implementation of the proposed changes (Refer to Enclosure 5, Item 4).

3.4.7 Plant Testing

Plant testing for the proposed changes will be completed as described in Enclosure 6, Section 10.4, “Testing,” (Refer to Enclosure 5, Item 5).

4.0 REGULATORY ANALYSIS

4.1 Applicable Regulatory Requirements/Criteria

10 CFR 50, Appendix K, “ECCS Evaluation Models,” requires that emergency core cooling system evaluation models assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level to allow for instrumentation error. A change to this paragraph, which became effective on July 31, 2000, allows a lower assumed power level, provided the proposed value has been demonstrated to account for uncertainties due to power level instrumentation error.

Implementing the Cameron LEFM √+ System is an effective way to obtain additional plant power without significantly changing current reactor core operations. Feedwater flow measurement uncertainty is the most significant contributor to core power measurement uncertainty. The LEFM provides a more accurate measurement of feedwater flow and thus reduces the uncertainty in the feedwater flow measurement. This reduced uncertainty, in combination with other uncertainties, results in an overall power level measurement uncertainty of 0.374% at RTP. This supports an increase in RTP from the current 3840 MWt to the proposed 3902 MWt. 10 CFR 50, Appendix K does not permit licensees to utilize a lower uncertainty and increase thermal power without NRC approval. 10 CFR 50.90 requires that licensees desiring to amend an operating license file an amendment with the NRC.

NRC RIS 2002-03, “Guidelines on the Content of Measurement Uncertainty Recapture Power Uprate Applications,” provides criteria for the content of license amendment requests involving power uprates based on measurement uncertainty recapture. This application is consistent with the requirements and criteria described in 10 CFR 50, Appendix K, 10 CFR 50.90, and the guidelines of NRC RIS 2002-03 (Enclosure 4).

4.2 Precedents

The following facilities have recently received NRC approval for power uprates based on using the LEFM √+ system.

Facility	Amendment No.	Approval Date	Accession No.
Limerick, Units 1 and 2	201/163	April 8, 2011	ML110691095
Shearon Harris	139	May 30, 2012	ML11356A096
Fermi 2 Correction	196	February 10, 2014 March 14, 2014	ML13364A131 ML14066A410
Catawba 1	277	April 29, 2016	ML16081A333

Unlike this Hope Creek submittal, the precedent submittals of Limerick and Fermi also included a request that included TSTF-493, “Clarify Application of Setpoint Methodology for LSSS Functions,” Revision 4. Hope Creek has addressed TSTF-493 as discussed in Section 3.4.4, “Instrument Setpoint Methodology,” of this enclosure.

Similar to the approved Shearon Harris submittal (Reference 6.13), Hope Creek is proposing allowing the use of Maintenance mode for operation at a power level greater than the CLTP, but less than the MUR uprated power as discussed in Section 3.2.1, “LEFM Feedwater Flow and Temperature Measurement,” of this enclosure.

4.3 No Significant Hazards Consideration

PSEG has evaluated this License Amendment Request (LAR) against the 10 CFR 50.92 criteria to determine if any significant hazards consideration is involved, and concluded that this proposed LAR does not involve a significant hazards consideration. The following is a discussion of how each of the 10 CFR 50.92 “Issuance of amendment,” criteria is satisfied.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change will increase the Hope Creek Generating Station rated thermal power (RTP) from 3840 megawatts thermal (MWt) to 3902 MWt. The reviews and evaluations performed to support the proposed uprated power conditions included all structures, systems, and components that would be affected by the proposed changes. The reviews and evaluations determined that these structures, systems, and components are capable of performing their design function at the proposed uprated RTP of 3902 MWt. Accident mitigation systems will function as designed. The performance requirements for these systems have been evaluated and found acceptable. Thus, the proposed changes do not create any new accident initiators or increase the probability of an accident previously evaluated.

The primary loop components (e.g., reactor vessel, reactor internals, control rod drive housings, piping and supports, and recirculation pumps) remain within their applicable structural limits and will continue to perform their intended design function at the uprated power level. Thus, there is no increase in the probability of a structural failure from these components. The safety relief valves and containment isolation valves meet design sizing requirements at the uprated power level. Because the plant integrity will not be affected by operation at the uprated condition, PSEG Nuclear LLC (PSEG) has concluded that all structures, systems, and components required to mitigate a transient remain capable of fulfilling their intended functions.

The current safety analyses were evaluated for operation at 3902 MWt. The results demonstrate that acceptance criteria for applicable analyses continue to be met at the uprated conditions. As such, applicable accident analyses continue to comply with the relevant event acceptance criteria. The analyses performed to assess the effects of mass and energy releases remain valid. Source terms used to assess radiological consequences have been determined to bound operation at the uprated power level.

Power level is an input assumption to equipment design and accident analyses, but is not a transient or accident initiator. Accident initiators are not affected by the power uprate, and plant safety barrier challenges are not created by the proposed change.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No

   No new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed change. Structures, systems, and components previously required for transient mitigation remain capable of fulfilling their intended design functions. The proposed change has no adverse effect on any safety-related structures, systems, or components and does not challenge the performance or integrity of any safety-related system.

   The proposed change does not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than previously evaluated. Plant operation at 3902 MWt does not create any new accident initiators or precursors. Credible malfunctions are bounded by the current accident analyses of record or recent evaluations demonstrating that applicable criteria are still met with the proposed change.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

   Response: No

   The margins of safety associated with the power uprate are those pertaining to core thermal power. Analyses of the primary fission product barriers have concluded that relevant design criteria remain satisfied, both from the standpoint of primary fission product barrier integrity and compliance with the required acceptance criteria. As appropriate, evaluations have been performed using methods that have either been reviewed and approved by the Nuclear Regulatory Commission, or are in compliance with regulatory review guidance and standards.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

4.4 Conclusions

Based upon the above, PSEG concludes that the proposed license amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of “no significant hazards consideration” is justified. Further, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission’s regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

10 CFR 51.22(c) provides criteria for, and identification of, licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed facility operating license amendment requires no environmental assessment in accordance with 10 CFR 51.22(c)(9) if facility operation per the proposed amendment would not:

- (i) involve a significant hazards consideration,
- (ii) result in a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or
- (iii) result in a significant increase in individual or cumulative occupational radiation exposure.

The Final Environmental Assessment (EA) and Finding of No Significant Impact (Reference 6.19) that was previously performed to support Hope Creek extended power uprate conditions assessed the environmental impacts up to a maximum thermal power level of 3952 MWt. The EA concluded that there would be no significant radiological environmental impacts associated with the proposed change.

There is no significant change in the types or significant increase in the amounts of any effluents. The effects of the proposed change on effluent sources were evaluated and concluded that the increase in effluents will be small and within the current EA, applicable permits, and regulations.

There is no significant increase in individual or cumulative occupational radiation exposure. Evaluations of projected radiation exposure concluded that normal occupational exposure is controlled by the plant radiation protection program and is maintained well within the current EA and the values required by regulations.

Accordingly, the proposed amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment is required in connection with the proposed amendment.

6.0 REFERENCES

6.1 GE-Hitachi Nuclear Energy (GEH) Report NEDC-32938P-A, "Licensing Topical Report: Generic Guidelines and Evaluations for General Electric Boiling Water Reactor Thermal Power Optimization," Revision 2, dated May 2003.

6.2 NRC Regulatory Issue Summary 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002 (ML013530183).

6.3 Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM System," Revision 0, dated March 1997.

6.4 Caldon Topical Report ER-157P, "Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or an LEFM CheckPlus System," Revision 8, dated June 2008.

6.5 Letter from John N. Hannon (USNRC) to C. Lance Terry (TU Electric), "Comanche Peak Steam Electric Station, Units 1 and 2 – Review of Caldon Engineering Topical Report ER-80P, 'Improving Thermal Power Accuracy and Plant Safety While Increasing Power Level Using the LEFM System,' (TACS Nos. MA2298 and MA2299)," dated March 8, 1999 (9903190065).

6.6 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-157P, Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM Check or CheckPlus System,' (TAC No. ME1321)," dated August 16, 2010 (ML102160663).

6.7 Cameron Procedure EFP68 "Commissioning Procedure for LEFM √+C, M, 280Fi and 880 Series Systems," Revision 4, dated 2/11/2016.

6.8 Letter from Carl F. Lyon (USNRC) to Stewart B. Minahan (Nebraska Public Power District), "Cooper Nuclear Station – Issuance of Amendment Re: Measurement Uncertainty Recapture Power Uprate (TAC No. MD7385)," dated June 30, 2008 (ML081540280).

6.9 Letter from Cristopher Gratton (USNRC) to Michael J. Pacilio (Exelon Nuclear), "LaSalle County Station, Units 1 and 2 – Issuance of Amendments Re: Measurement Uncertainty Recapture Power Uprate (TAC Nos. ME3288 and ME3289)," dated September 16, 2010 (ML101830361).

6.10 Letter from Peter Bamford (USNRC) to Michael J. Pacilio (Exelon Nuclear), "Limerick Generating Station, Units 1 and 2 – Issuance of Amendments Re: Measurement Uncertainty Recapture Power Uprate and Standby Liquid Control System Changes (TAC Nos. ME3589, ME3590, ME3591, and ME3592)," dated April 8, 2011 (ML110691095).

6.11 BWRVIP-139, "BWR Vessel and Internals Project Steam Dryer Inspection and Flaw Evaluation Guidelines, dated April 2005.

6.12 BWRVIP-139-A, "BWR Vessel and Internals Project Steam Dryer Inspection and Flaw Evaluation Guidelines, dated July 2009.

6.13 Letter from Araceli T. Billoch Colon (USNRC) to Chris Burton (Progress Energy Carolinas, Inc.), "Shearon Harris Nuclear Power Plant, Unit 1 – Issuance of Amendment Re: Measurement Uncertainty Recapture Power Uprate (TAC No. ME169)," dated May 30, 2012 (ML11356A096).

6.14 Technical Specification Task Force (TSTF) Traveler TSTF-493, "Clarify Application of Setpoint Methodology for LSSS Functions," Revision 4, dated July 2009.

6.15 Letter from Paul Davison (PSEG) to USNRC, "Hope Creek License Amendment Request – Digital Power Range Neutron Monitoring (PRNM) System Upgrade", (PRNM LAR H15-01, LR-N15-0178), dated September 21, 2015.

6.16 Letter from Eric Carr (PSEG) to USNRC, "Hope Creek License Amendment Request – Pressure - Temperature Limits Curves Revision (P-T Limits LAR H17-02, LR-N17-0032), dated March 27 2017.

6.17 GE Hitachi Nuclear Energy, "Applicability of GE Methods to Expanded Operating Domains," NEDC-33173P-A, Revision 4, November 2012.

6.18 GE Hitachi Nuclear Energy, "GE Hitachi Boiling Water Reactor, Detect and Suppress Solution – Confirmation Density," NEDC-33075P-A, Revision 8, November 2013.

6.19 73 FR 13032, "PSEG Nuclear, LLC; Hope Creek Generating Station Final Environmental Assessment and Finding of No Significant Impact; Related to the Proposed License Amendment to Increase the Maximum Reactor Power Level", dated March 11, 2008.

6.20 Letter from Paul Davison (PSEG) to USNRC, "Supplemental Information - License Amendment Request – Digital Power Range Neutron Monitoring (PRNM) System Upgrade", (PRNM LAR H15-01, LR-N16-0092), dated June 17, 2016.

6.21 Letter from John P. Boska, NRC to Steven D, Capps, "McGuire Nuclear Stations Units 1 and 2, Issuance of Amendments Regarding Measurement Uncertainty Power Uprate (TAC NOS. ME8213 AND ME8214), dated May 16, 2013 (ADAMS Accession No. ML13073A041).

6.22 Artificial Island Operating Guide (AIOG), A-5-500-EEE-1686, Rev.11, dated 4/30/12

6.23 CDI Technical Note (TN) 16-23P, Steam Dryer Analysis

6.24 Cameron Document ER-972, “Traceability Between Topical Report (ER-157P-A Rev. 8 and Rev. 8 Errata) and the System Uncertainty Report,” Revision 2,

Enclosure 2

Mark-up of Renewed Facility Operating License and Technical Specifications

The following Technical Specifications pages for Renewed Facility Operating License NPF-57 are affected by this change request:

Technical Specification	**Page**
Operating License	3, 5, and 15
Definitions	1-6
2.2, “Limiting Safety System Settings”	2-4
3/4.1.3.1, “Control Rod Operability”	3/4 1-4
3/4.1.4.1, "Rod Worth Minimizer"	3/4 1-16
3/4.3.3.6, “Control Rod Block Instrumentation”	3/4 3-59
3/4.4.1, “Recirculation System”	3/4 4-1 and 4-2a
3/4.10.2, “Rod Worth Minimizer”	3/4 10-2

reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

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

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

(6) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility. Mechanical disassembly of the GE14i isotope test assemblies containing Cobalt-60 is not considered separation.

(7) PSEG Nuclear LLC, pursuant to the Act and 10 CFR Part 30, to intentionally produce, possess, receive, transfer, and use Cobalt-60.

C. This renewed 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) <u>Maximum Power Level</u>

PSEG Nuclear LLC is authorized to operate the facility at reactor core power levels not in excess of ~~3840~~ 3902 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

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

The Technical Specifications contained in Appendix A, as revised through Amendment No. ~~200~~, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the renewed license. PSEG Nuclear LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(7) Fire Protection (Section 9.5.1.8, SSER No. 5; Section 9.5.1, SSER No. 6)

PSEG Nuclear LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility through Amendment No. 15 and as described in its submittal dated May 13, 1986, and as approved in the SER dated October 1984 (and Supplements 1 through 6) subject to the following provision:

PSEG Nuclear LLC may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

(8) Solid Waste Process Control Program (Section 11.4.2, SER; Section 11.4, SSER No. 4)

DELETED

(9) Emergency Planning (Section 13.3, SSER No. 5)

DELETED

(10) Initial Startup Test Program (Section 14, SSER No. 5)

DELETED

(11) Partial Feedwater Heating (Section 15.1, SER; Section 15.1, SSER No. 5; Section 15.1, SSER No. 6)

The facility shall not be operated with a rated thermal power feedwater temperature less than ~~329.6°F~~ for the purpose of extending the normal fuel cycle.

331.5 °F

(12) Detailed Control Room Design Review (Section 18.1, SSER No. 5)

a. Submit a report to the NRC staff in accordance with 10 CFR 50.4 describing the final drain line configuration and summarizing the testing results that demonstrate drainage has been established for all four quadrants.

b. Monitor penetration sleeve J13 daily for water leakage when the reactor cavity is flooded up. In addition, perform a walkdown of the torus room to detect any leakage from other drywell penetrations. These actions shall continue until corrective actions are taken to prevent leakage through J13 or through the four air gap drains.

c. Perform UT measurements of the drywell shell betwe n elevation 86'-11" (floor of the drywell concrete) and elevation 93'-0" (bottom of penetration J13) below penetration J13 area during the next three refueling outages. In addition, UT measurements shall be performed around the full 360 degree circumference of the drywell between elevations 86'-11" and 88'-0" (underside of the torus down comer vent piping penetrations). The r sults of the UT measurements will be used to identify drywell surfaces requiring augmented inspections in accordance with IWE requirements for the period of extended operation, establish a corrosion rate, and demonstrate that the effects of aging will be adequately managed such that the drywell can perform its intended function until April 11, 2046. Within 90 days of completion of each refueling outage, submit a report to the NRC staff in accordance with 10 CFR 50.4 summarizing the results from the UT measurements and if appropriate, corrective action.

2.C.(28) PSEG will operate the facility at a thermal power level not to exceed 3,840 MWt until the Power Range Neutron Monitoring System license amendment request is approved by the NRC and implemented by PSEG.

2.C.(29) PSEG will operate the facility at a thermal power level not to exceed 3,840 MWt until the Pressure - Temperature (PT) Limits license amendment request is approved by the NRC and implemented by PSEG.

D. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70. An exemption from the criticality alarm requirements of 10 CFR 70.24 was granted in Special Nuclear Material License No. 1953, dated August 21, 1985. This exemption is described in Section 9.1 of Supplement No. 5 to the SER. This previously granted exemption is continued in this renewed operating license. An exemption from certain requirements of Appendix A to 10 CFR Part 50, is described in Supplement No. 5 to the SER. This exemption is a schedular exemption to the requirements of General Design Criterion 64, permitting delaying functionality of the Turbine Building Circulating Water System-Radiation Monitoring System until 5 percent power for local indication, and until 120 days after fuel load for control room indication (Appendix R of SSER 5). Exemptions from certain requirements of Appendix J to 10 CFR Part 50, are described in Supplement No. 5 to the SER. These include an exemption from the requirement of Appendix J, exempting main steam isolation valve leak-rate testing at 1.10 Pa (Section 6.2.6 of SSER 5); an exemption from Appendix J, exempting Type C testing on traversing incore probe system shear valves (Section 6.2.6 of SSER 5); an exemption from Appendix J,

PROCESS CONTROL PROGRAM

1.33 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packing 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, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

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

RATED THERMAL POWER

1.35 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~3840~~ 3902 MWt.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.36 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

REPORTABLE EVENT

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

ROD DENSITY

1.38 ROD DENSITY shall be the number of control rod notches inserted as a fraction of the total number of control rod notches. All rods fully inserted is equivalent to 100% ROD DENSITY.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, (Setdown)	≤ 17% of RATED THERMAL POWER	≤ 19% of RATED THERMAL POWER
b. Simulated Thermal Power – Upscale**		
1) Flow Biased-Two Recirculation Loop Operation	≤0.56w+ 58% ~~≤0.57w + 59%~~**(a) with a maximum of ≤113.5% of RATED THERMAL POWER	≤0.56w + 60 % ~~≤0.57w + 61%~~** with a maximum of ≤115.5% of RATED THERMAL POWER
2) Flow Biased – Single Recirculation Loop Operation	≤0.56(w-10.8%) + 58% ~~≤0.57(w-10.6%)+59%~~**(a) with a maximum of ≤ 113.5% of RATED THERMAL POWER	≤0.56(w-9%) + 60% ~~≤0.57(w-9%)+61%~~** with a maximum of ≤115.5% of RATED THERMAL POWER
c. Neutron Flux-Upscale	≤ 116.3% of RATED THERMAL POWER	≤ 118.3% of RATED THERMAL POWER
d. Inoperative	NA	NA
e. 2-Out-Of-4 Voter	NA	NA
f. OPRM Upscale	See CORE OPERATING LIMITS REPORT	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1037 psig	≤ 1057 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11.0 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed

*See Bases Figure B 3/4 3-1.

**The Average Power Range Monitor Scram function varies as a function of recirculation loop drive flow (w).

(a) When the Automated BSP Scram Regions Setpoints are implemented in accordance with Action 10 of Table 3.3.1-1, the Simulated Thermal Power-Upscale Flow Biased Setpoint will be adjusted per the CORE OPERATING LIMITS REPORT

REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

d. One or more BPWS groups with four or more inoperable control rods*****, within 4 hours, restore control rod(s) to OPERABLE status.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

e. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

f. With one or more scram discharge volume (SDV) vent or drain lines*** with one valve inoperable, isolate the associated line within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.****

g. With one or more SDV vent or drain lines*** with both valves inoperable, isolate the associated line within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.****

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by:

a. Verifying each valve to be open,* and

b. Cycling each valve through at least one complete cycle of full travel.

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* These valves may be closed intermittently for testing under administrative controls.

** May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

*** Separate Action entry is allowed for each SDV vent and drain line.

**** An isolated line may be unisolated under administrative control to allow draining and venting of the SDV.

***** Not applicable when THERMAL POWER is greater than ~~8.6%~~ RATED THERMAL POWER.

8.5%

REACTIVITY CONTROL SYSTEMS

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

3.1.4.1 The Rod worth minimizer (RWM) shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*#, when THERMAL POWER is less than or equal to ~~8.6%~~ 8.5% of RATED THERMAL POWER, minimum allowable low power setpoint.

ACTION:

a. With the RWM inoperable after the first 12 control rods are fully withdrawn, operation may continue provided that control rod movement and compliance with the prescribed control rod pattern is verified by a second licensed operator or other technically qualified member of the unit technical staff who is present at the reactor control console.

b. With the RWM inoperable before the first twelve (12) control rods are fully withdrawn, one startup per calendar year may be performed provided that the control rod movement and compliance with the prescribed control rod pattern are verified by a second licensed operator or other technically qualified member of the unit technical staff who is present at the reactor control console.

c. Otherwise, control rod movement may be only by actuating the manual scram or placing the reactor mode switch in the Shutdown position.

SURVEILLANCE REQUIREMENTS

4.1.4.1 The RWM shall be demonstrated OPERABLE:

a. In OPERATIONAL CONDITION 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical, and in OPERATIONAL CONDITION 1 within 8 hours prior to RWM automatic initiation when reducing THERMAL POWER, by verifying proper indication of the selection error of at least one out-of-sequence control rod.

* Entry into OPERATIONAL CONDITION 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

See Special Test Exception 3.10.2.

TABLE 3.3.6-2
CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale[(a)]		
i. Low Trip Setpoint (LTSP)[(b)]	**	**
ii. Intermediate Trip Setpoint (ITSP)[(c)]	**	**
iii. High Trip Setpoint (HTSP)[(d)]	**	**
b. Inoperative	NA	NA
c. Downscale	**	**
2. APRM		
a. Simulated Thermal Power – Upscale		
1) Flow Biased – Two Recirculation Loop Operation	≤0.56w + 53.1% ~~≤0.57w + 54%~~* with a maximum of ≤ 108% of RATED THERMAL POWER	≤0.56w + 55.1% ~~≤0.57w + 56%~~* with a maximum of ≤ 111% of RATED THERMAL POWER
2) Flow Biased – Single Recirculation Loop Operation	≤0.56(w-10.8%) + 53.1% ~~≤0.57(w-10.6%) + 54%~~* with a maximum of ≤ 108% of RATED THERMAL POWER	~~≤0.57(w-9%) + 56%~~* with a maximum of ≤ 111% of RATED THERMAL POWER ≤0.56(w-9%) + 55.1%
b. Inoperative	NA	NA
c. Downscale	≥ 4% of RA ED THERMAL POWER	≥ 2% of RATED THERMAL POWER
d. Simulated Thermal Power – Upscale (Setdown)	≤ 11% of RATED THERMAL POWER	≤ 13% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 1.0 x 10^5 cps	≤ 1.6 x 10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps	≥ 1.8 cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 108/125 divisions of full scale	≤ 110/125 divisions of full scale
c. Inoperative	NA	NA
d. Downscale	≥ 5/125 divisions of full scale	≥ 3/125 divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High (Float Switch)	109'1" (North Volume) 108'11.5" (South Volume)	109'3" (North Volume) 109'1.5" (South Volume)
6. Deleted		
7. REACTOR MODE SWITCH SHUTDOWN POSITION	NA	NA

* The rod block function is varied as a function of recirculation loop flow (w).

** Refer to the CORE OPERATING LIMITS REPORT for these values.

a. Each upscale trip level is applicable over its specified rated power range. All RBM trips are automatically bypassed below the low power setpoint (LPSP). The upscale LTSP is applied between the LPSP and the intermediate power setpoint (IPSP). The upscale ITSP is applied between the IPSP and the high power setpoint (HPSP). The HTSP is applied above the HPSP.

b. APRM Simulated Thermal Power is ≥ 28% and < 63% RTP

c. APRM Simulated Thermal Power is ≥ 63% and < 83%

d. APRM Simulated Thermal Power is ≥ 83%

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

1. Within 4 hours:

a) Place the recirculation flow control system in the Local Manual mode, and
b) Reduce THERMAL POWER to ≤ ~~60.86%~~ 59.89% of RATED THERMAL POWER, and
c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit per Specification 2.1.2, and
d) Reduce the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) limit to a value specified in the CORE OPERATING LIMITS REPORT for single loop operation, and
e) Reduce the LINEAR HEAT GENERATION RATE (LHGR) limit to a value specified in the CORE OPERATING LIMITS REPORT for single loop operation, and
f) Limit the speed of the operating recirculation pump to less than or equal to 90% of rated pump speed, and
g) Perform surveillance requirement 4.4.1.1.2 if THERMAL POWER is ≤ 38% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is ≤ 50% of rated loop flow.

2. Within 4 hours, reduce the Average Power Range Monitor (APRM) Scram Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specification 2.2.1; otherwise, with the Trip Setpoints and Allowable Values associated with one trip system not reduced to those applicable for single recirculation loop operation, place the affected trip system in the tripped condition and within the following 6 hours, reduce the Trip Setpoints and Allowable Values of the affected channels to those applicable for single recirculation loop operation per Specification 2.2.1.

3. Within 4 hours, reduce the APRM Control Rod Block Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation per Specification 3.3.6; otherwise, with the Trip Setpoint and Allowable Values associated with one trip function not

*See Special Test Exception 3.10.4.

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 With one reactor coolant system recirculation loop not in operation in accordance with the Surveillance Frequency Control Program verify that:

a. Reactor THERMAL POWER is ≤ ~~60.86%~~ 59.89% of RATED THERMAL POWER, and

b. The recirculation flow control system is in the Local Manual mode, and

c. The speed of the operating recirculation pump is less than or equal to 90% of rated pump speed.

4.4.1.1.2 With one reactor coolant system recirculation loop not in operation, within no more than 15 minutes prior to either THERMAL POWER increase or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is ≤ 38% of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is ≤ 50% of rated loop flow:

a. ≤ 145°F between reactor vessel steam space coolant and bottom head drain line coolant, and

b. ≤ 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c. ≤ 50°F between the reactor coolant within the loop not in operation and the operating loop.

The differential temperature requirements or Specifications 4.4.1.1.2b and 4.4.1.1.2c do not apply when the loop not in operation is isolated from the reactor pressure vessel.

4.4.1.1.3 DELETED.

SPECIAL TEST EXCEPTIONS

3/4.10.2 ROD WORTH MINIMIZER

LIMITING CONDITION FOR OPERATION

3.10.2 The sequence constraints imposed on control rod groups by the rod worth minimizer (RWM) per Specification 3.1.4.1 may be suspended for the following tests provided that control rod movement prescribed for this testing is verified by a second licensed operator or other technically qualified member of the unit technical staff present at the reactor console:

a. Shutdown margin demonstrations, Specification 4.1.1.

b. Control rod scram, Specification 4.1.3.2.

c. Control rod friction measurements.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2 when THERMAL POWER is less than or equal to ~~8.6%~~ 8.5% of RATED THERMAL POWER.

ACTION:

With the requirements of the above specification not satisfied, verify that the RWM is OPERABLE per Specifications 3.1.4.1.

SURVEILLANCE REQUIREMENTS

4.10.2 When the sequence constraints imposed by the RWM are bypassed, verify:

a. That movement of the control rods from 75% ROD DENSITY to the RWM low power setpoint is limited to the approved control rod withdrawal sequence during scram and friction tests.

b. That movement of control rods during shutdown margin demonstrations is limited to the prescribed sequence per Specification 3.10.3.

c. Conformance with this specification and test procedures by a second licensed operator or other technically qualified member of the unit technical staff.

Enclosure 3

Mark-up of Technical Specification Bases "For Information Only"

The following Technical Specification Bases pages for Renewed Facility Operating License NPF-57 are affected by this change request:

Technical Specification Bases	Page
3/4.1.3, "Control Rods"	B 3/4 1-2a
3/4.1.4, "Control Rod Program Controls"	B 3/4 1-3
3/4.4.1, "Recirculation System"	B 3/4 4-1 (Insert 4)

REACTIVITY CONTROL SYSTEMS

BASES

CONTROL RODS (Continued)

Out of sequence control rods may increase the potential reactivity worth of a dropped control rod during a CRDA. At ≤ ~~8.6%~~ 8.5% RTP, the generic banked position withdrawal sequence (BPWS) analysis requires inserted control rods not in compliance with BPWS to be separated by at least two OPERABLE control rods in all directions, including the diagonal. Therefore, if two or more inoperable control rods are not in compliance with BPWS and not separated by at least two OPERABLE control rods, action must be taken to restore compliance with BPWS or restore the control rods to OPERABLE status. LCO 3.1.3.1.c is modified by a Note indicating that the Condition is not applicable when > ~~8.6%~~ 8.5% RTP, since the BPWS is not required to be followed under these conditions, as described in the Bases for LCO 3.1.4. The allowed Completion Time of 4 hours is acceptable, considering the low probability of a CRDA occurring. In lieu of restoring compliance with BPWS or restoring the control rods to OPERABLE status, an evaluation of the postulated CRDA may be performed to verify that the maximum incremental rod worth of an assumed dropped control rod would not result in exceeding the CRDA design limit of 280 cal/gm fuel enthalpy and would not result in unacceptable dose consequences due to the number of fuel rods exceeding 170 cal/gm fuel enthalpy as described in the UFSAR. The allowed Completion Time of 8 hours is acceptable, considering the low probability of a CRDA occurring.

In addition to the separation requirements for inoperable control rods, an assumption in the CRDA analysis is that no more than three inoperable control rods are allowed in any one BPWS group. Therefore, with one or more BPWS groups having four or more inoperable control rods, the control rods must be restored to OPERABLE status. LCO 3.1.3.1.d is modified by a Note indicating that the Condition is not applicable when THERMAL POWER is > ~~8.6%~~ 8.5% RTP since the BPWS is not required to be followed under these conditions, as described in the Bases for LCO 3.1.4. The allowed Completion Time of 4 hours is acceptable, considering the low probability of a CRDA occurring.

Control rod insertion capability is demonstrated by surveillance 4.1.3.1.2 inserting each partially or fully withdrawn control rod at least one notch and observing that the control rod moves. The control rod may then be returned to its original position. This ensures the control rod is not stuck and is free to insert on a scram signal. At any time, a control rod is immovable for reasons not associated with the control rod drive mechanism, a determination of that control rod's trippability (Operability) must be made and appropriate actions taken. As an example, if the control rod can be scrammed, but can not be moved due to a RMCS failure, the rod(s) may continue to be considered OPERABLE provided all other related surveillances are current.

Damage within the control rod drive mechanism could be a generic problem, therefore with a withdrawn control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than ~~8.6%~~ 8.5% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RWM to be OPERABLE when THERMAL POWER is less than or equal to ~~8.6%~~ 8.5% of RATED THERMAL POWER provides adequate control.

The RWM provides automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section 15.4.9 of the FSAR and the techniques of the analysis are presented in Reference 1.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system back up the written sequence used by the operator for withdrawal of control rods.

This page is a markup of the TS Bases Insert 4 included in Reference 6.20 (LR-N16-0092, "Supplemental Information - License Amendment Request - Digit l Power Range Neutron Monitoring (PRNM) System Upgrade (CAC No. MF6768)), dated June 17, 2016

INSERT 4

The Average Power Range Monitor Scram and rod block functions vary as a function of recirculation loop driv flow (w). The effective drive flow correction term (Δw) is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop operation (TLO) and single loop operation (SLO) at the same core flow. Δw is based on a physical phenomenon and represents the amount of drive flow from the active loop that flows backwards through the inactive loop's jet pumps during SLO. The flow input to the APRM STP Scram function Allowable Value (AV) and Nominal Trip Set Point (NTSP) is adjusted by Δw during SLO to account for this phenomenon.

The form of the function equation is: Slope x (Flow [w] – Flow Offset [Δw]) + Power Offset.

GEH's setpoint methodology is described in NEDC-33864P Appendix P, P1 and P2 (VTD 432598). The methodology also accounts for increased uncertainty in the idle recirculation loop flow signal, which requires the NTSP to be further from the AV under SLO than it is under TLO. This is accomplished by reducing the power offset term for the APRM STP-Upscale RPS Trip (Table 2.2.1-1 Function 2.b):

TLO AV: ~~0.57w + 61%~~ 0.56w + 60%
TLO NTSP: ~~0.57w + 59%~~ 0.56w + 58%

SLO AV: ~~0.57(w – Δw) + 61%~~ 0.56(w-Δw) + 60%
SLO NTSP: ~~0.57(w – Δw) + 58.1%~~ 0.56(w-Δw) + 57%

When the SLO mode is manually enabled the NUMAC APRM instrument applies an offset term to the flow signal. To avoid an additional action to manually adjust the power offset (from ~~59%~~ 58% to ~~58.1%~~ 57%), the SLO NTSP equation is solved for the same power offset as the TLO NTSP equation. Using Δw = 9% yields a flow offset of ~~10.6%~~ 10.8% to maintain the power offset at ~~59%~~ 58%.

~~0.57(w – 9%) + 58.1% = 0.57(w – 10.6%) + 59%~~ 0.56(w-9%) + 57% ≈ .56(w-10.8%) + 58%

This ~~10.6%~~ 10.8% flow offset term is defined as the "SLO Setting Adjustment" (the actual value is ~~10.58~~ 10.79% but it is rounded up to one decimal place for conservatism since the SLO Setting Adjustment is programmed to one decimal place in the NUMAC quipment). This term is applied to the NTSP during SLO by the NUMAC APRM to both account for the 9% Δw flow offset and the increased margin required to the AV. The Δw and SLO Setting Adjustment values have been inserted into the APRM STP-Upscale equations in Table 2.2.1-1.

This same methodology is also applied to the APRM STP-Upscale Rod Block Trip (Table 3.3.6-2 Function 2.a).

Use of the SLO Setting Adjustment simplifies the process for adjusting APRM scram and control rod block setpoints for SLO, as required by TS 3/4.4.1. Expressing the SLO Trip Setpoint in terms of SLO Setting Adjustment reflects how the NUMAC PRNM system is setup and operated.

Enclosure 4

Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		HOPE CREEK RESPONSE		
NRC RIS 2002-03		Hope Creek MUR LAR		
Section	Description	Document	Section	Title / Description

I. Feedwater Flow Measurement Technique and Power Measurement Uncertainty

I.1	Detailed description of plant-specific implementation of feedwater flow measurement technique and power increase gained as a result of implementing technique	Enclosure 1	3.1 3.2	Background and General Approach LEFM Feedwater Flow Measurement and Core Thermal Power Uncertainty
I.1.A	NRC approval of topical report on flow measurement technique	Enclosure 1	3.2.1	LEFM Feedwater Flow and Temperature Measurement
I.1.B	Reference to NRC's approval of proposed measurement technique	Enclosure 1	3.2.1	LEFM Feedwater Flow and Temperature Measurement
I.1.C	Plant Implementation	Enclosure 1	3.2.2	Plant Implementation
I.1.D	Disposition of NRC criteria	Enclosure 1	3.2.4	Disposition of NRC Criteria for Use of LEFM Topical Reports
I.1.E	Total power measurement uncertainty calculation for the plant	Enclosure 1 Enclosure 14	3.2.3	LEFM and Core Thermal Power Measurement Uncertainty and Methodology Heat Balance Uncertainty Calculation
I.1.F	Calibration and maintenance procedures	Enclosure 1	3.2.4 3.2.5	Disposition of NRC Criteria for Use of LEFM Topical Reports Deficiencies and Corrective Actions
I.1.G	Proposed allowed outage time for LEFM, and basis for selected time	Enclosure 1	3.2.4	Disposition of NRC Criteria for Use of LEFM Topical Reports
I.1.H	Proposed actions if outage time is exceeded, and basis for actions	Enclosure 1	3.2.1	LEFM Feedwater Flow and Temperature Measurement

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		HOPE CREEK RESPONSE		
NRC RIS 2002-03		Hope Creek MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**

II. Accidents and Transients For Which the Existing Analyses of Record Bound Plant Operation at the Proposed Uprated Power Level

II.1	Matrix for bounded accidents and transients	Enclosure 6	9.0	Reactor Safety Performance Evaluations

III. Accidents and Transients for Which the Existing Analyses of Record Do Not Bound Plant Operation at the Proposed Uprated Power level

III.1	Matrix for unbounded accidents and transients	Enclosure 6	9.0	Reactor Safety Performance Evaluations
III.2	Matrix for unbounded accidents and transients	Enclosure 6	9.0	Reactor Safety Performance Evaluations
III.3	Matrix for unbounded accidents and transients	Enclosure 6	9.0	Reactor Safety Performance Evaluations

IV. Mechanical / Structural / Material Component Integrity and Design

IV.1.A.i	Reactor vessel, nozzles, and supports	Enclosure 6	3.2 3.2.1 3.2.2	Reactor Vessel Fracture Toughness Reactor Vessel Structural Evaluation
IV.1.A.i	Reactor core support structures and vessel internals	Enclosure 1 Enclosure 6	3.4.2 3.3 3.3.1 3.3.2 3.3.3 3.4	Adverse Flow Effects Reactor Internals Reactor Internal Pressure Difference Reactor Internals Structural Evaluation Steam Separator and Dryer Performance Flow-Induced Vibration
IV.1.A.iii	Control rod drive mechanisms	Enclosure 6	2.5	Reactivity Control

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		HOPE CREEK RESPONSE		
NRC RIS 2002-03		Hope Creek MUR LAR		
Section	Description	Document	Section	Title / Description
IV.1.A.iv	Nuclear Steam Supply System (NSSS) piping, pipe supports, branch nozzles	Enclosure 6	3.4 3.5 3.5.1 3.6 3.7 3.8 3.9 3.10 3.11	Flow-Induced Vibration Piping Evaluation Reactor Coolant Pressure Boundary Piping Reactor Recirculation System Main Steam Line Flow Restrictors Main Steam Isolation Valves Reactor Core Isolation Cooling Residual Heat Removal System Reactor Water Cleanup System
IV.1.A.v	Balance of plant (BOP) piping (NSSS interface systems, safety-related cooling water systems, and containment systems)	Enclosure 6	3.5 3.5.2 6.4.1 4.1 4.7	Piping Evaluation Balance-of-Plant Piping Evaluation Cooling Water Systems Containment System Performance Post-LOCA Containment Atmosphere Control System
IV.1.A.vi	Steam generator tubes, secondary side internal support structures, shell and nozzles	N/A	N/A	N/A
IV.1.A.vii	Reactor coolant pumps	N/A	N/A	N/A
IV.1.A.viii	Pressurizer shell, nozzles, and surge line	N/A	N/A	N/A

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		HOPE CREEK RESPONSE		
NRC RIS 2002-03		Hope Creek MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**
IV.1.A.ix	Safety-related valves	Enclosure 6	3.1	Nuclear System Pressure Relief / Overpressure Protection
			3.8	Main Steam Isolation Valves
			4.1	Containment System Performance
			4.1.1	Generic Letter 89-10 Program
			4.1.2	Generic Letter 96-05
			4.1.3	Generic Letter 95-07 Program
			6.5	Standby Liquid Control System
IV.1.B.i	Stresses	Enclosure 6	3.2	Reactor Vessel
			3.2.2	Reactor Vessel Structural Evaluation
			3.4	Flow-Induced Vibration
			3.5	Piping Evaluation
			3.5.1	Reactor Coolant Pressure Boundary Piping
			3.5.2	Balance-of-Plant Piping Evaluation
IV.1.B.ii	Cumulative usage factors	Enclosure 6	3.2.2	Reactor Vessel Structural Evaluation
IV.1.B.iii	Flow induced vibration	Enclosure 6	3.4	Flow-Induced Vibration
		Enclosure 1	3.4.2	Adverse Flow Effects

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		HOPE CREEK RESPONSE		
NRC RIS 2002-03		Hope Creek MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**
IV.1.B.iv	Changes in temperature (pre-and post-uprate)	Enclosure 6	1.3 1.3.1 1.3.2 Table 1-2	TPO Plant Operating Conditions Reactor Heat Balance Reactor Performance Improvement Features Thermal-Hydraulic Parameters at TPO Uprate Conditions
IV.1.B.v	Changes in pressure (pre-and post-uprate)	Enclosure 6	1.3 1.3.1 1.3.2 Table 1-2	TPO Plant Operating Conditions Reactor Heat Balance Reactor Performance Improvement Features Thermal-Hydraulic Parameters at TPO Uprate Conditions
IV.1.B.vi	Changes in flow rates (pre-and post-uprate)	Enclosure 6	1.3 1.3.1 1.3.2 Table 1-2	TPO Plant Operating Conditions Reactor Heat Balance Reactor Performance Improvement Features Thermal-Hydraulic Parameters at TPO Uprate Conditions
IV.1.B.vii	High-energy line break locations	Enclosure 6	10.1 10.1.1 10.1.2	High Energy Line Break Steam Line Breaks Liquid Line Breaks

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		HOPE CREEK RESPONSE		
NRC RIS 2002-03		Hope Creek MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**
IV.1.B.viii	Jet impingement and thrust forces	Enclosure 6	10.1 10.1.1 10.1.2 10.1.2.7	High Energy Line Break Steam Line Breaks Liquid Line Breaks Pipe Whip and Jet Impingement
IV.1.C.i	Reactor vessel pressurized thermal shock calculations	Enclosure 6	3.1	Nuclear System Pressure Relief / Overpressure Protection
IV.1.C.ii	Reactor vessel fluence evaluation	Enclosure 6	3.2 3.2.1	Reactor Vessel Fracture Toughness
IV.1.C.iii	Reactor vessel heatup and cooldown pressure-temperature limit curves	Enclosure 6	3.2.1	Fracture Toughness
IV.1.C.iv	Reactor vessel low-temperature overpressure protection	Enclosure 6	3.2 3.2.1	Reactor Vessel Fracture Toughness
IV.1.C.v	Reactor vessel upper shelf energy	Enclosure 6	3.2 3.2.1	Reactor Vessel Fracture Toughness
IV.1.C.vi	Reactor vessel surveillance capsule withdrawal schedule	Enclosure 6	3.2 3.2.1	Reactor Vessel Fracture Toughness
IV.1.D	Code of record and any changes to the code of record	Enclosure 6	3.2 3.2.2 3.5 3.5.1	Reactor Vessel Reactor Vessel Structural Evaluation Piping Evaluation Reactor Coolant Pressure Boundary Piping

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		HOPE CREEK RESPONSE		
NRC RIS 2002-03		Hope Creek MUR LAR		
Section	Description	Document	Section	Title / Description
IV.1.E	Any changes to component inspection and testing programs and erosion / corrosion programs	Enclosure 6	3.5 3.5.1 3.5.2 10.6	Piping Evaluation Reactor Coolant Pressure Boundary Piping Balance-of-Plant Piping Evaluation Plant Life
IV.1.F	NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes"	N/A	N/A	N/A

V. Electrical Equipment Design

V.1.A	Emergency diesel generators	Enclosure 6	6.1 6.1.2	AC Power On-Site Power
V.1.B	Station blackout equipment	Enclosure 6	9.3.2	Station Blackout
V.1.C	Environmental qualification of electrical equipment	Enclosure 6	10.3	Environmental Qualification
V.1.D	Grid stability	Enclosure 1 Enclosure 6	3.4.5 6.1 6.1.1	Grid Stability Studies AC Power Off-Site Power

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		HOPE CREEK RESPONSE		
NRC RIS 2002-03		Hope Creek MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**

VI. System Design

Section	Description	Document	Section	Title / Description
VI.1.A	NSSS Interface Systems for BWRs (e.g., suppression pool cooling)	Enclosure 6	3.4	Flow-Induced Vibration
			3.5	Piping Evaluation
			3.5.1	Reactor Coolant Pressure Boundary Piping
			3.5.2	Balance-of-Plant Piping Evaluation
			3.6	Reactor Recirculation System
			3.7	Main Steam Line Flow Restrictors
			3.8	Main Steam Isolation Valves
			3.9	Reactor Core Isolation Cooling
			3.10	Residual Heat Removal System
			3.11	Reactor Water Cleanup System
VI.1.B	Containment systems	Enclosure 6	4.1	Containment System Performance
			4.7	Post-LOCA Containment Atmosphere Control System
VI.1.C	Safety-related cooling water systems	Enclosure 6	6.4	Water Systems
			6.4.1	Cooling Water Systems
			6.4.2	Main Condenser/Circulating Water/Normal Heat Sink Performance
			6.4.3	Ultimate Heat Sink

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		HOPE CREEK RESPONSE		
NRC RIS 2002-03		Hope Creek MUR LAR		
Section	**Description**	**Document**	**Section**	**Title / Description**
VI.1.D	Spent fuel pool storage and cooling systems	Enclosure 6	6.3 6.3.1 6.3.2 6.3.3 6.3.4	Fuel Pool Fuel Pool Cooling Crud Activity and Corrosion Products Radiation Levels Fuel Racks
VI.1.E	Radioactive waste systems	Enclosure 6	4.5 8.1 8.2 8.3 8.4 8.4.1 8.4.2 8.4.3 8.5 8.6	Standby Gas Treatment System Liquid and Solid Waste Management Gaseous Waste Management Radiation Sources in the Reactor Core Radiation Sources in Reactor Coolant Coolant Activation Products Activated Corrosion Products Fission Products Radiation Levels Normal Operation Off-Site Doses
VI.1.F	Engineered safety features (ESFs) heating, ventilation, and air conditioning systems	Enclosure 6	4.4 6.6	Main Control Room Atmosphere Control System Power Dependent Heating, Ventilation, and Air Conditioning

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		HOPE CREEK RESPONSE		
NRC RIS 2002-03		Hope Creek MUR LAR		
Section	Description	Document	Section	Title / Description

VII. Other

VII.1	Operator actions sensitive to the power uprate and effects on time available for operator actions	Enclosure 1	3.4.6	Operator Training, Human Factors, and Procedures
		Enclosure 6	4.1	Containment System Performance
			6.7	Fire Protection
			9.3	Special Events
			10.5	Operator Training and Human Factors
VII.2.A	Emergency and abnormal operating procedures	Enclosure 6	10.9	Emergency Operating Procedures
VII.2.B	Control room controls, displays (including the safety parameter display system) and alarms	Enclosure 1	3.2.4	Disposition of NRC Criteria for Use of LEFM Topical Reports
			3.4.3	Plant Modifications
		Enclosure 6	10.5	Operator Training and Human Factors
VII.2.C	Control room reference simulator	Enclosure 6	10.5	Operator Training and Human Factors
VII.2.D	Operator training program	Enclosure 6	10.5	Operator Training and Human Factors
VII.3	Modification completion	Enclosure 1	3.4.3	Plant Modifications
VII.4	Procedure Revisions - License Power Level	Enclosure 1	3.2.6	Reactor Power Monitoring
VII.5.A	10 CFR 51.22, Exclusion of Environmental Review, including discussion of effect of the power uprate on types and amounts of effluents released offsite, and whether bounded by final environmental statement and previous Environmental Assessments for the plant	Enclosure 1	5.0	Environmental Consideration
		Enclosure 6	6.4.2.1	Discharge Limits
			8.6	Normal Operation Off-Site Doses
VII.5.B	10 CFR 51.22, Exclusion of Environmental Review, including discussion of effect of the power uprate on individual and cumulative occupational radiation exposure	Enclosure 1	5.0	Environmental Consideration
		Enclosure 6	8.5	Radiation Levels

NRC Regulatory Issue Summary (RIS) 2002-03 Cross-Reference

NRC REQUIREMENT		HOPE CREEK RESPONSE		
NRC RIS 2002-03		Hope Creek MUR LAR		
Section	Description	Document	Section	Title / Description

VIII. Changes to Technical Specifications, Protection System Settings, and Emergency System Settings

VIII.1	A detailed discussion of each change to the plant's technical specifications, protection system settings, and/or emergency system settings needed to support the power uprate	Enclosure 1 Enclosure 2	1.0 2.0	Description Detailed Discussion Markup of Proposed Operating License and Technical Specification Pages
VIII.1.A	Description of the change	Enclosure 1 Enclosure 2	1.0 2.0	Description Detailed Discussion Markup of Proposed Operating License and Technical Specification Pages
VIII.1.B	Identification of analyses affected by and/or supporting the change	Enclosure 1 Enclosure 6	3.3	Evaluation of Operating License and Technical Specifications Changes GEH Safety Analysis Report NEDC-33871P
VIII.1.C	Justification for the change, including the type of information discussed in Section III, above, for any analyses that support and/or are affected by change	Enclosure 1 Enclosure 6	3.3	Evaluation of Operating License and Technical Specifications Changes GEH Safety Analysis Report NEDC-33871P

Enclosure 5

Summary of Regulatory Commitments

	COMMITMENT	COMMITTED DATE OR OUTAGE	ONE-TIME ACTION (YES/NO)	ON-GOING COMMITMENT (YES/NO)
1	LEFM functionality requirements and required actions and allowed outage times when the LEFM is not fully functional, will be added to appropriate plant procedures	Prior to operation above 3840 MWt	NO	YES
2	Necessary operating procedure revisions (including Emergency Operating Procedures and Abnormal Operating Procedures) will be completed prior to implementation of the proposed power uprate	Prior to operation above 3840 MWt	YES	NO
3	The plant simulator will be modified for the uprated conditions and the changes will be validated in accordance with plant configuration control processes	Prior to operation above 3840 MWt	YES	NO
4	Operator training will be completed prior to implementation of the proposed power uprate	Prior to operation above 3840 MWt	YES	NO
5	Plant testing for the proposed changes will be completed as described in Enclosure 6, Section 10.4, "Testing"	Upon reaching 100% MUR rated power	YES	NO
6	The plant process computer will have an alarm to alert the operators to LEFM status changes	Prior to operation above 3840 MWt	YES	NO
7	Prior to reaching MUR conditions (baseline) and following the first scheduled refueling outage after reaching MUR conditions, a visual inspection shall be conducted of all accessible steam dryer locations with a MASR less than 2.0. One location with a MASR less than 2.0 will not be inspected due to accessibility and dose considerations. This location has an MASR of 1.74 that is considerably higher than the most limiting locations covered under the inspection plan. The inspections will be performed in accordance with BWRVIP-139-A guidelines	RF21 and RF22	YES	YES
8	Moisture carryover shall be measured upon achieving 100% MUR rated power, and weekly for the first operating cycle after MUR implementation.	Upon reaching 100% MUR rated power	YES	YES