May 22, 2006

Mr. Gene F. St. Pierre, Site Vice President
c/o James M. Peschel
Seabrook Station
PO Box 300
Seabrook, NH 03874

SUBJECT: SEABROOK STATION, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NO. MC8434)

Dear Mr. St. Pierre:

The Commission has issued the enclosed Amendment No. 110 to Facility Operating License No. NPF-86 for the Seabrook Station, Unit No 1, in response to your application dated September 22, 2005, as supplemented by letters dated March 24, 2006, and April 28, 2006, whereby FPL Energy Seabrook, LLC submitted an application requesting to increase the licensed thermal power level for Seabrook Station, Unit No. 1 (Seabrook).

The amendment increases the licensed core power level for Seabrook by 1.7% to 3648 megawatts thermal. This increase will be achieved by the installation of the Caldon LEFM [leading edge flow measurement] CheckPlus™ ultrasonic flow measurement system, which allows for more accurate measurement of feedwater flow.

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

Sincerely,

/RA/

G. Edward Miller, Project Manager
Plant Licensing Branch I-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-443

Enclosures:
1. Amendment No. 110 to NPF-86
2. Safety Evaluation

cc w/encls: See next page

May 22, 2006

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NAME	GEMiller	CRaynor	MMitchell	EMurphy for AHeiser	GWilson	AHowe	JNakoski
DATE	5/22/06	5/19/06	12/15/2005	4/21/06	5/8/06	5/12/06	5/12/06

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NAME	MKotzalas	DSolorio	KManoly	NO'Keefe	JMoore	DRoberts	CHaney
DATE	1/23/06	4/7/06	3/6/06	4/19/06	05/18/06	5/22/06	5/22/06

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FPL ENERGY SEABROOK, LLC, ET AL.*

DOCKET NO. 50-443

SEABROOK STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 110
License No. NPF-86

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

   A. The application for amendment filed by FPL Energy Seabrook, LLC, et al. (the licensee), dated September 22, 2005, as supplemented by letters dated March 24, 2006, and April 28, 2006, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

*FPL Energy Seabrook, LLC (FPLE Seabrook) is authorized to act as agent for the: Hudson Light & Power Department, Massachusetts Municipal Wholesale Electric Company, and Taunton Municipal Light Plant and has exclusive responsibility and control over the physical construction, operation and maintenance of the facility.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-86 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No.110, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

   In addition, the license is amended to revise paragraph 2.C.(1) to reflect the increase in the reactor core power level. Paragraph 2.C.(1) is hereby amended to read as follows:

   FPL Energy, Seabrook, LLC, is authorized to operate the facility at reactor core power levels not in excess of 3648 megawatts thermal (100% of rated power).

3. This license amendment is effective as of its date of issuance and shall be implemented within 12 months of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Catherine Haney, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: May 22, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 110

FACILITY OPERATING LICENSE NO. NPF-86

DOCKET NO. 50-443

Replace the following page of Facility Operating License No. NPF-86 with the attached revised page. The revised page is identified by amendment number and contains marginal lines indicating the areas of change.

Remove	Insert
3	3

Replace the following page of the Appendix A, Technical Specifications, with the attached revised page as indicated. The revised page is identified by amendment number and contains marginal lines indicating the area of change.

Remove	Insert
1-5	1-5

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 110 TO FACILITY OPERATING LICENSE NO. NPF-86

FPL ENERGY SEABROOK, LLC

SEABROOK STATION, UNIT NO. 1

DOCKET NO. 50-443

1.0 INTRODUCTION

By letter dated September 22, 2005, as supplemented by letters dated March 24, 2006, and April 28, 2006, FPL Energy Seabrook, LLC (FPLE or the licensee) submitted License Amendment Request (LAR) No. 05-04, requesting to increase the licensed thermal power level for Seabrook Station, Unit No. 1 (Seabrook).

The amendment would increase the licensed core power level for Seabrook by 1.7% to 3648 megawatts thermal (MWt). This increase will be achieved by the installation of the Caldon LEFM [leading edge flow measurement] CheckPlus™ ultrasonic flow measurement (UFM) system, which allows for more accurate measurement of feedwater (FW) flow. The supplements dated March 24, 2006, and April 28, 2006, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC or the Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 8, 2005 (70 FR 67748).

2.0 BACKGROUND

Nuclear power plants are licensed to operate at a specified core thermal power. Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix K, requires licensees to assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level when performing loss-of-coolant (LOCA) and emergency core cooling system (ECCS) analyses. This requirement is included to ensure that instrumentation uncertainties are adequately accounted for in the analyses. Appendix K to 10 CFR Part 50 allows licensees to assume a power level less than 1.02 times the licensed power level (but not less than the licensed power level), provided the licensee has demonstrated that the proposed value adequately accounts for instrumentation uncertainties. The licensee has proposed to use a power measurement uncertainty of 0.3%. To achieve this level of accuracy, the licensee will install a Caldon LEFM CheckPlus™ UFM system for measuring the main FW flow at Seabrook. The Caldon system provides a more accurate measurement of FW flow than the FW flow measurement accuracy assumed during the development of the original 10 CFR, Part 50, Appendix K requirements and that of the current method of FW flow measurement used to calculate reactor thermal output. The Caldon system will measure FW mass flow to within plus or minus (±) 0.28% for Seabrook. This bounding FW mass flow uncertainty would be used to calculate a total power measurement uncertainty of 0.3%. On the basis of this, FPLE proposed to reduce the power measurement uncertainty required by 10 CFR, Part 50, Appendix K to 0.3%. The improved power measurement uncertainty would obviate the need for the 2% power margin originally required by 10 CFR, Part 50, Appendix K, thereby allowing an increase in the reactor power available for electrical generation by 1.7%.

This accuracy is supported by Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check™ System," which, by safety evaluation report (SER) dated March 8, 1999 (Agencywide Documents and Management System (ADAMS) Accession Number 9903190065 (legacy library)), was approved by the NRC staff for use in justification of measurement uncertainty recapture (MUR) power uprates up to 1%. Subsequently, by Safety Evaluation (SE) dated December 20, 2001 (ADAMS Accession Number ML013540256), the NRC staff approved Caldon Topical Report ER-157P, "Basis for a Power Uprate With the LEFM Check™ or LEFM CheckPlus™ System," for use in justifying MUR power uprates up to 1.7%.

3.0 EVALUATION

3.1 Instrumentation and Controls (I&C)

3.1.1 Background

The NRC staff review in the area of I&C covers the proposed plant-specific implementation of the FW flow measurement technique and the power increase gained as a result of implementing this technique in accordance with the guidelines (A through H) provided in Section I of Attachment 1 to Regulatory Information Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications." The NRC staff review was conducted to confirm that the licensee's implementation of the proposed FW flow measurement device is consistent with the staff-approved Caldon Topical Reports ER-80P and ER-157P and adequately addresses the four additional criterion listed in the NRC staff SER of the Caldon Topical Reports ER-80P and ER-157P. The NRC staff also reviewed the power uncertainty calculations to ensure that the proposed uncertainty value of 0.3% correctly accounted for all uncertainties due to power level instrumentation errors, and that the calculations met the relevant requirements of Appendix K to 10 CFR Part 50 as described in Section 2.0 of this SE.

The neutron flux instrumentation is calibrated to the core thermal power, which is determined by an automatic or manual calculation of the energy balance around the plant's nuclear steam supply system (NSSS). This calculation is called a "secondary calorimetric" for a pressurized-water reactor (PWR). The accuracy of this calculation depends primarily upon the accuracy of FW flow and FW enthalpy measurements. FW flow uncertainty is the most significant contributor to the overall core thermal power uncertainty. An accurate measurement of this parameter will result in an accurate determination of core thermal power.

Currently, the instrumentation used for measuring FW flow rate at Seabrook is a venturi. This device generates a differential pressure proportional to the FW velocity in the pipe. Due to the high cost of calibration of the venturi and the need to improve flow instrumentation

measurement uncertainty, the industry assessed other flow measurement techniques and found the LEFM Check^{TM} and the LEFM CheckPlus^{TM} UFM system UFMs to be viable alternatives. Both of these systems use the transit time methodology to measure fluid velocity.
The basis of the transit time methodology is that ultrasonic pulses transmitted into a fluid stream travel faster in the direction of the fluid flow than opposite the flow. The difference in the upstream and downstream traversing times of the ultrasonic pulses is proportional to the fluid velocity in the pipe. Temperature is determined using a pre-established correlation between the mean propagation velocity of the ultrasound pulses in the fluid and the fluid pressure. The mean fluid density may be obtained using the measured pressure and the derived mean fluid temperature as an input to a table of thermodynamic properties of water.

Both UFMs use multiple diagonal acoustic paths, instead of a single diagonal path, so that velocities measured along each path can be numerically integrated over the pipe cross section to determine the average fluid velocity in the pipe. This fluid velocity is multiplied by a velocity profile correction factor, the pipe cross section area, and the fluid density to determine the FW mass flow rate in the piping. The velocity profile correction factor is derived from calibration testing of the LEFM in a plant-specific piping model at a calibration laboratory.

Caldon first developed the LEFM Check^{TM} system as described in Topical Report ER-80P and subsequently issued its supplement Topical Report ER-157P on an improved design, the LEFM CheckPlus^{TM} UFM system. Topical Report ER-80P, which describes the LEFM technology, includes calculations of power measurement uncertainty using an LEFM Check^{TM} UFM system in a typical two-loop PWR or two-feedwater-line boilling-water reactor (BWR), and provides guidelines and equations for determining the plant-specific power calorimetric uncertainties. The supplement Topical Report ER-157P describes the LEFM CheckPlus^{TM} UFM system and lists non-proprietary results of a typical PWR or BWR thermal power measurement uncertainty calculation using a single meter LEFM Check^{TM} or LEFM CheckPlus^{TM} UFM system. These two reports collectively provide a generic basis and guidelines for power uprate using a Caldon LEFM Check^{TM} or LEFM CheckPlus^{TM} UFM system for FW flow and temperature measurements. The LEFM Check^{TM} UFM system uses eight transducers, two on each of four acoustic paths in a single plane of the spool piece, where the velocity measured by any one of the four acoustic paths is the vector sum of the axial and the transverse components of fluid velocity as projected onto the path. The LEFM CheckPlus^{TM} UFM system uses sixteen transducers, eight each in two orthogonal planes of the spool piece. As such, the LEFM CheckPlus^{TM} UFM system is a combination of two LEFM Check^{TM} UFM systems. In the LEFM CheckPlus^{TM} UFM system, when the fluid velocity measured by an acoustic path in one plane is averaged with the fluid velocity measured by its companion path in the second plane, the transverse components of the two velocities are canceled and the result reflects only the axial velocity of the fluid. This makes the numerical integration of four pairs of averaged axial velocities and computation of volumetric flow inherently more accurate than can be obtained using four acoustic paths in a single plane. Also, since there are twice as many acoustic paths and there are two independent clocks to measure the transit time, errors due to uncertainties in path length and transit time measurements are reduced.

FPLE has proposed to install the Caldon LEFM CheckPlus^{TM} UFM system at Seabrook to measure FW flow and temperature, and referenced proprietary Topical Reports ER-80P and ER-157P in its submittal for the proposed MUR power uprate. The LEFM CheckPlus^{TM} UFM system at Seabrook will consist of an electronic cabinet and a permanently-installed measurement section (spool piece). The spool piece is installed in a common portion of the

FW flow loops upstream of the FW header and the electronic unit is installed in the turbine building.

3.1.2 Regulatory Evaluation

As discussed above, the NRC staff review in the area of I&C covers compliance with Guidelines A through H provided in Section I of Attachment 1 to RIS 2002-03. Additionally, this review confirms that the implementation of the proposed FW flow measurement device will be consistent with the NRC staff-approved Caldon Topical Reports ER-80P and ER-157P, including the four additional criterion listed in the NRC staff SER of these reports.

3.1.3 Technical Evaluation

Currently, Seabrook has a Caldon 2-path chordal spool piece UFM to measure individual FW flow to each of the four steam generators (SGs). The data from these UFMs are used to periodically normalize the steam mass flow to the secondary power calorimetric. The existing UFMs will remain in place for data collection only and will not interact (hydraulically or electrically) with the newly installed LEFM CheckPlus™ UFM system. These UFMs will not be relied upon or used to support any function associated with the power calorimetric.

The LEFM CheckPlus™ UFM system will be used for continuous calorimetric power determination by linking with the main plant computer via an Ethernet digital interface using fiber optic cables and data converters. The system will incorporate self-verification features to ensure that the hydraulic profile and signal processing requirements are within its design basis uncertainty analysis. Hard-wired alarms will provide additional assurance of operator notification of a system failure.

As stated in LAR 05-04, the Seabrook LEFM CheckPlus™ UFM system will be calibrated prior to installation by a site-specific model test at Alden Research Laboratories and this calibration will be confirmed during the in-situ site acceptance testing. FPLE confirmed that the LEFM CheckPlus™ UFM system calibration will be bounded by the uncertainty established in the licensee's request for MUR power uprate. Installation and commissioning of Seabrook's LEFM CheckPlus™ UFM system will be in accordance with the FPLE and Caldon installation and test procedures.

Based on the NRC staff's review of FPLE's submittals as reflected in the above discussion, the NRC staff finds that the licensee has sufficiently addressed the plant-specific implementation of the LEFM CheckPlus™ UFM system topical report guidelines, and that the licensee's description of the FW flow measurement technique and the MUR power uprate due to implementing this technique adequately addresses the guidance in Items A through C of Section I of Attachment 1 to RIS 2002-03.

The NRC staff's SER on Caldon Topical Report ER-80P included four additional criteria to be addressed by a licensee referencing this topical report to support a MUR power uprate. In its LAR 05-04, including supplements, FPLE addressed each of the four criteria as follows:

(1) The licensee should discuss the maintenance and calibration procedures that will be implemented with the incorporation of the LEFM. These procedures should include processes and contingencies for an

inoperable LEFM and the effect on thermal power measurement and plant operation.

In response, FPLE stated that implementation of the MUR power uprate license amendment will include developing the necessary procedures and documents required for operation, maintenance, calibration, testing, and training with the new Caldon LEFM CheckPlus™ UFM system. These procedures will incorporate Caldon's maintenance and calibration requirements for the LEFM CheckPlus™ UFM system. The Caldon LEFM CheckPlus™ UFM system is designed and manufactured in accordance with Caldon's 10 CFR, Part 50, Appendix B, Quality Assurance Program and its Verification and Validation Program. Caldon's Verification and Validation Program fulfills the requirements of American National Standards Institute (ANSI)/Institute of Electrical and Electronics Engineers (IEEE)-American Nuclear Society (ANS) Standard 7-4.3.2 and American Society of Mechnical Engineers Boiler and Pressure Vessel Code (ASME Code)-NQA-2a. In addition, the program is consistent with guidance for software verification and validation in Electric Power Research Institue (EPRI) TR-103291S. Specific examples of quality measures undertaken in the design, manufacture, and testing of the Caldon LEFM CheckPlus™ UFM system are provided in Caldon Topical Report ER-80P, Section 6.4 and Table 6-1.

Selected I&C personnel will be trained and qualified per FPLE's Institute for Nuclear Power Operations-accredited training program before maintenance or calibration is performed and prior to increasing power above 3587 MWt. This training will include lessons learned from industry experience. Initially, formal training by Caldon will be provided to Seabrook personnel. Corrective action involving maintenance will be performed by personnel qualified in accordance with FPLE's Instrumentation and Calibration Training Program and formally trained on the LEFM CheckPlus™ UFM system. The Seabrook LEFM CheckPlus™ UFM system will be included in Caldon's Verification and Validation Program, and procedures will be maintained for user notification of important deficiencies in accordance with 10 CFR Part 21 reporting requirements.

The LEFM CheckPlus™ UFM system is assumed to be inoperable if one or more paths is lost. The proposed allowed outage time (AOT) for operation at any power level in excess of the current licensed core power level (3587 MWt) with the LEFM CheckPlus™ UFM system out of service is 48 hours provided steady-state conditions persist (i.e., no power changes in excess of 10%) throughout the 48-hour period.

For the LEFM CheckPlus™ UFM system out-of-service condition, the 48-hour AOT will start at the time of the failure and this failure will be annunciated in the control room. FPLE stated that the plant operating procedures will be revised to state that if the inoperable LEFM CheckPlus™ UFM system is not restored to an operable status or the plant experiences a power change of greater than 10% during the 48-hour period, then the permitted maximum power level will be reduced to the current licensed core thermal power level of 3587 MWt.

Additionally, FPLE stated that there are alternate plant instruments (FW venturies, main steam flow, and Caldon 2-path chordal devices) available to be used if the LEFM CheckPlus™ UFM system is out of service. The alternate instrumentation and the LEFM CheckPlus™ UFM system calorimetric are completely separate, and the calculations of core thermal power are performed independently by the main plant computer. The preferred alternate method is the main steam flow instruments normalized to the LEFM CheckPlus™ UFM system flow. The steam flow normalization is performed by taking the ratio of total steam flow measured by the alternate instrumentation to the FW flow measured by the LEFM CheckPlus™ UFM system. The flow input can be provided by either the main steam flow normalized to the FW venturies, or directly from the FW venturies, which are continuously calibrated to the last good value provided by the LEFM CheckPlus™ UFM system. All three methods are bounded by the 2% uncertainty for a core power level of 3587 MWt. The accuracy of the FW venturies and the main steam flow instrumentation will gradually degrade over time as a result of nozzle fouling and transmitter drift. The values of this drift, however, are typically in the range of tenths of a percent of the calibrated span over 18 to 24 months or more. This typical drift value will not result in any significant drift for the instrumentation associated with the calorimetric measurements over a 48-hour period.

A main plant computer system failure will be treated as a loss of both the Caldon LEFM CheckPlus™ UFM system and the ability to obtain a corrected calorimetric power using alternate plant instrumentation. Thus, operation at the MUR core power level of 3648 MWt may continue until the next required nuclear instrumentation heat balance adjustment which could be up to 24 hours. The main plant computer system failure will then result in reducing core thermal power to the current licensed core power level of 3587 MWt, as needed, to support the manual calorimetric measurement. The 48-hour time period will not apply in this specific case, as a manual calorimetric will be required.

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

In response, FPLE stated that Seabrook currently has flow measurement venturies on the FW system, and differential pressure instrumentation on the main steam system. The FW system flow venturies and the main steam differential pressure instrumentation will serve as backup inputs to the calorimetric to be used when the LEFM CheckPlus™ UFM system is not available. The new LEFM CheckPlus™ UFM system will be independent of the FW system venturies, the main steam system flow instrumentation, and the Caldon 2-path chordal devices. Thus, operational and maintenance history associated with the Caldon 2-path chordal devices is not applicable to the new LEFM CheckPlus™ UFM system.

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

In response, FPLE stated that the total power calorimetric accuracy using the LEFM CheckPlus™ UFM system is determined by evaluating the reactor thermal power sensitivity to deviations in the process parameters used to calculate reactor thermal power. Uncertainties for parameters that are not statistically independent are arithmetically summed to produce groups that are independent of each other, which can be statistically combined. Then all independent parameters/groups that contribute to the power measurement uncertainty are combined using a statistical summation to determine the total power measurement uncertainty.

(4) Licensees for plant installations where the ultrasonic meter (including the LEFM) was not installed with flow elements calibrated to a site-specific piping configuration (flow profiles and meter factors not representative of the plant-specific installation), should provide additional justification for use. This justification should show either that the meter installation is 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 the plant configuration for the specific installation, including the propagation of flow profile effects at higher Reynolds numbers. Additionally, for previously installed calibrated elements, the licensee should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

FPLE stated that Criterion 4 does not apply to Seabrook. The calibration factor for the Seabrook spool piece will be established by tests of this spool piece at Alden Research Laboratory to standards traceable to National Institute of Standards and Technology standards. These tests will include a full-scale model of Seabrook's hydraulic geometry and tests in a straight pipe. An Alden Research Laboratory data report for these tests and a Caldon engineering report evaluating the test data will be provided to Seabrook. The calibration factor used for the LEFM CheckPlus™ UFM system at Seabrook will be based on these reports. The uncertainty in the calibration factor for the flow meter spool piece will be based on the Caldon engineering report. The site-specific uncertainty analysis will document these analyses. This document will be maintained on file, as part of the technical basis for the Seabrook MUR.

Final acceptance of the site-specific uncertainty analyses will occur after the completion of the commissioning process. The commissioning process will verify bounding calibration test data and provide final positive confirmation that actual performance in the field will meet the uncertainty bounds established for the instrumentation. Final commissioning is expected to be completed during the fall 2006 refueling outage (RFO).

Given the above listed responses provided by FPLE to the four questions, the NRC staff finds that FPLE has fully addressed the four criteria specified in the staff's SER of topical reports ER-80P and ER-157P and, therefore, has adequately addressed the guidance in Items D, G, and H of Section I of Attachment 1 to RIS 2002-03.

In addition to the above, Items E and F in Section I of Attachment 1 to RIS 2002-03, respectively, request licensees to submit a plant-specific total power measurement uncertainty calculation, explicitly identify all parameters and their individual contribution to the power uncertainty, and provide information to address the specified aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric.

To address Item E of RIS 2002-03, FPLE provided a summary of the Seabrook core thermal power measurement uncertainty in a table format listing uncertainty values from the Caldon Engineering Report ER-482P which provides a detailed calculation of the uncertainties. FPLE stated that the values in the uncertainty column of the table and the total power uncertainty determination are bounding values. The staff audit of ER-482P found that the calculations determined individual measurement uncertainties of all parameters contributing to the core thermal power measurement uncertainty and those uncertainties were then combined using square root of sum of squares methodology, as described in Regulatory Guide (RG) 1.105 and Instrument Society of America S67.04.

Upon review of the submitted information, the NRC staff finds that the licensee has provided calculations of the total power measurement uncertainty at the plant, explicitly identifying all parameters and their individual contribution to the power uncertainty and, therefore, has adequately addressed the guidance in Item E of Section I of Attachment 1 to RIS 2002-03.

To address the five aspects contained in Item F of RIS 2002-03 as applicable to the LEFM CheckPlus™ UFM system, FPLE provided detailed information in their response to Criterion 1 of the NRC staff SER on ER-80P. To address these five aspects applicable to all other instruments that affect the power calorimetric and the main plant computer, FPLE listed all those process inputs and stated that the process inputs are obtained from analog instrumentation channels that are maintained and calibrated in accordance with required periodic calibration procedures. Additionally, FPLE stated that the configuration of the hardware associated with these process inputs is maintained in accordance with the Seabrook change control process. FPLE further stated that the maintenance and calibration of the main plant computer inputs is performed in accordance with the Seabrook periodic maintenance program, and the software and hardware configuration is maintained in accordance with the Seabrook change control process, which includes verification and validation of changes to software and hardware configuration.

Based on the information provided by FPLE, the NRC staff finds that FPLE has addressed the calibration and maintenance aspects of the LEFM CheckPlus™ UFM system and all other instruments affecting power calorimetric and, thus, complied with the guidance in item F of Section I of Attachment 1 to RIS 2002-03.

3.1.4 Summary

The NRC staff reviewed of the licensee's proposed plant-specific implementation of the FW flow measurement device and the power uncertainty calculations and determined that the licensee's proposed use of Topical Report ER-80P, and its supplement ER-157P, is consistent with the staff's approval of the topical reports. The NRC staff has also determined that the licensee adequately accounted for instrumentation uncertainties in the reactor thermal power measurement uncertainty calculations and demonstrated that the calculations meet the relevant requirements of 10 CFR Part 50, Appendix K as described in Section 2 of this SE. Therefore, the NRC staff finds the I&C aspect of the proposed MUR power uprate acceptable.

The NRC staff finds that there is reasonable assurance that when the licensee implements FW flow measurement with the LEFM CheckPlus^{TM} UFM system ultrasonic flow meter, Seabrook thermal to determine plant thermal power, power measurement uncertainty will be limited to ±0.3% of the reactor thermal power and, therefore, is supportive of the proposed 1.7% thermal power uprate proposed for Seabrook.

3.2 Reactor Systems

3.2.1 Regulatory Evaluation

The NRC staff's review in the area of reactor systems covers the impact of the proposed MUR power uprate on (1) fuel design, (2) nuclear design, (3) thermal-hydraulic design, (4) performance of control and safety systems connected to the reactor coolant system (RCS), and (5) LOCA and non-LOCA transients (RIS 2002-03, Attachment 1, Sections II, III, and VI). The review is conducted to verify that FPLE's analyses bound plant operation at the proposed power level and that the results of FPLE's analyses related to the areas under review continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate.

3.2.2 Technical Evaluation

The NSSS design parameters provide the RCS and secondary system conditions (pressures, temperatures, and flow) that are used as the basis for the design transients and for systems, components, accidents and transient analyses and evaluations. The parameters are established using conservative assumptions to provide bounding conditions to be used in the Design Basis Accident (DBA) analyses. Table 8.1-1 of Seabrook LAR 05-04 lists the existing and MUR design operating parameters in detail for various cases analyzed by the licensee. The major input parameters and assumptions used in the analyses are as follows:

1. Analyzed core power level of 3659 MWt (3678 MWt NSSS power level)

2. Thermal design flow of 93,600 gpm/loop

3. SG tube plugging values of 0% and 10%

4. Design core bypass flow of 8.3%

5. Full power, normal operating Tavg from 571.0 °F to 589.1 °F

6. FW temperature from 390 °F to 452.4 °F

7. 17x17 robust fuel assemblies with intermediate flow mixers.

The licensee also considered 2% power uncertainties in its safety analyses. These design operating parameters were used in the licensee's safety analyses to support the 5.2% stretch power uprate (SPU) from 3411 MWt to 3587 MWt.

The licensee re-analyzed the Updated Final Safety Evaluation Report (UFSAR) Chapter 15 LOCA and non-LOCA transients and accidents in support of the Seabrook 5.2% SPU. The licensee used NRC-approved computer codes and methodologies for each accident and transient analysis. These analyses were performed at a rated core power of 3587 MWt using plant parameter values for those operating conditions plus a 2% initial conditions uncertainty. Thus, the analyzed core power level of 3659 MWt is 2% greater than the current licensed core power level of 3587 MWt and 0.3% greater than the proposed MUR core power level of 3648 MWt. The staff reviewed and approved the licensee's transient and accident analyses at 3659 MWt conditions assumed by the SPU, confirming that the acceptance criteria were still met under these conditions. The results of this review are summarized in Table 3.2 below.

Table 3.2
Pressurized Water Reactor Systems - Summary of Staff Review

Topic	LAR 05-04 Section	UFSAR Section(s)	Bounding Analysis (Including Reference)	NRC Approved
Large-Break LOCA	Table 3.1-1, Row 3.1	15.6.5	Seabrook LAR 04-03, Attachment 1, Section 6.1.1	Yes
Small-Break LOCA	Table 3.1-1, Row 3.2	15.6.5	Seabrook LAR 04-03, Attachment 1, Section 6.1.2	Yes
Post-LOCA Long-Term Cooling	Table 3.1-1, Row 3.3	15.6.5	Seabrook LAR 04-03, Attachment 1, Section 6.1.3	Yes
Excessive Heat Removal Due to FW System Malfunctions	Table 3.1-1, Row 3.9	15.1.1, 15.1.2	Seabrook LAR 04-03, Attachment 1, Section 6.3.2.1	Yes
Excessive Increase in Steam Flow	Table 3.1-1, Row 3.10	15.1.3	Seabrook LAR 04-03, Attachment 1, Section 6.3.2.2	Yes
Inadvertent Opening of a Steam Generator Dump, Relief, or Safety Valve	Table 3.1-1, Row 3.11	15.1.4	Seabrook LAR 04-03, Attachment 1, Section 6.3.2.3	Yes
Steam System Piping Failure	Table 3.1-1, Row 3.12	15.1.5	Seabrook LAR 04-03, Attachment 1, Section 6.3.2.4	Yes

Loss of External Load / Turbine Trip	Table 3.1-1, Row 3.13	15.2.2, 15.2.3	Seabrook LAR 04-03, Attachment 1, Section 6.3.3.1	Yes
Loss of Normal FW Flow	Table 3.1-1, Row 3.14	15.2.7	Seabrook LAR 04-03, Attachment 1, Section 6.3.3.2	Yes
Loss of Offsite Power (LOOP)	Table 3.1-1, Row 3.15	15.2.6	Seabrook LAR 04-03, Attachment 1, Section 6.3.3.3	Yes
FW System Pipe Breaks	Table 3.1-1, Row 3.16	15.2.8	Seabrook LAR 04-03, Attachment 1, Section 6.3.3.4	Yes
Total Loss of Forced Reactor Coolant Flow	Table 3.1-1, Row 3.18	15.3.2	Seabrook LAR 04-03, Attachment 1, Section 6.3.4.1.2	Yes
Single Reactor Coolant Pump Locked Rotor / Shaft Break	Table 3.1-1, Row 3.19	15.3.3, 15.3.4, 15.3.5	Seabrook LAR 04-03, Attachment 1, Section 6.3.4.2	Yes
Uncontrolled Rod Cluster Control Assembly (RCCA) Withdrawal from Subcritical	Table 3.1-1, Row 3.0	15.4.1	Seabrook LAR 04-03, Attachment 1, Section 6.3.5.1	Yes
Uncontrolled RCCA Withdrawal at Power	Table 3.1-1, Row 3.21	15.4.2	Seabrook LAR 04-03, Attachment 1, Section 6.3.5.1	Yes
RCCA Misoperation	Table 3.1-1, Row 3.22	15.4.3	Seabrook LAR 04-03, Attachment 1, Section 6.3.5.3	Yes
Startup of an Inactive Reactor Coolant Pump	Table 3.1-1, Row 3.23	15.4.4	Three-loop operation is not allowed per Seabrook Technical Specifications	N/A
Inadvertent Boron Dilution	Table 3.1-1, Row 3.24	15.4.6	Seabrook LAR 04-03, Attachment 1, Section 6.3.5.5	Yes

Inadvertent Loading and Operation of a Fuel Assembly	Table 3.1-1, Row 3.25	15.4.7	Seabrook LAR 04-03, Attachment 1, Section 6.3.5.6	Yes
RCCA Ejection	Table 3.1-1, Row 3.26	15.4.8	Seabrook LAR 04-03, Attachment 1, Section 6.3.5.7	Yes
Inadvertent Actuation of ECCS	Table 3.1-1, Row 3.27	15.5.1	Submitted by letter dated November 7, 2005	No, see discussion
Chemical and Volume Control System Malfunction that Increases RCS Inventory	Table 3.1-1, Row 3.28	15.5.2	Seabrook LAR 04-03, Attachment 1, Section 6.3.6.2	Yes
Inadvertent Opening of a Pressurizer Safety or Relief Valve	Table 3.1-1, Row 3.29	15.6.1	Seabrook LAR 04-03, Attachment 1, Section 6.3.7.1	Yes
Anticipated Transients Without Scram	Table 3.1-1, Row 3.30	15.8	Seabrook LAR 04-03, Attachment 1, Section 6.3.8	Yes
Station Blackout (SBO)	Table 3.1-1, Row 3.31	8.4	Seabrook LAR 04-03, Attachment 1, Section 6.3.9	Yes

It should be noted that during its SPU evaluation, the staff expressed concern regarding the possibility of the pressurizer becoming water-solid in response to the inadvertent actuation of the ECCS event, potentially leading to a more serious ANS Condition III event that would be a violation of one of the Condition II acceptance criteria. To resolve this concern, License Condition 2.K was included in Amendment Number 101 to Facility Operating License NPF-86, issued February 28, 2005, which required, prior to startup from RFO 11, that FPLE qualify the power operated relief valves for the ability to close after passing water. Alternately, the license condition allows the re-analysis of the inadvertent actuation of the ECCS event, using NRC-approved methods, demonstrating that operators will be able to secure injection prior to filling the pressurizer. Currently, FPLE has submitted a re-analysis to the NRC, performed at an initial core power of 3659 MWt, which is bounding of the proposed MUR uprate conditions. This re-analysis is currently under review and will be resolved independently of this review, however, prior to implementation of the MUR power uprate. Therefore, this does not impact the NRC staff's approval of the MUR power uprate.

3.2.3 Summary

During its review and approval of the 5.2% SPU, the NRC staff focused on the licensee's assessment of the impact of the SPU on fuel design, nuclear design, thermal-hydraulic design, performance of control and safety systems connected to the RCS, and LOCA and non-LOCA transient analyses. Based on the above, the NRC staff determined that the results of the licensee's analyses related to these areas continue to meet the applicable acceptance criteria following implementation of the MUR power uprate. The current analysis of record is based on 3659 MWt that includes 2% measurement uncertainty. The proposed amendment is based on the use of the Caldon LEFM CheckPlus™ UFM system that would decrease the uncertainty in the FW flow, thereby decreasing the power level measurement uncertainty from 2% to 0.3%. The proposed MUR rated thermal power 3648 MWt is bounded by the current analyses of record.

The NRC staff also finds that the Caldon LEFM CheckPlus™ UFM system hydraulic characteristics as described in the licensee's references are accurately portrayed. Also, the hydraulic aspects of the Caldon LEFM CheckPlus™ UFM system and the claimed associated uncertainties are acceptable for the determination of FW flow rate at Seabrook in support of the requested MUR power uprate. It should be noted that Caldon is generically evaluating the effect of transducer replacement on the Caldon LEFM CheckPlus™ UFM system uncertainties. Though the expected impact of transducer replacement is minimal, should it be determined to be detrimental to the system uncertainties, 10 CFR, Part 21, would control the resolution of these impacts. Therefore, the NRC staff does not consider this activity inimical to approval of the Seabrook MUR power uprate.

The NRC staff finds that the hydraulic aspects of the Caldon LEFM CheckPlus™ UFM system have been accurately described in applicable Caldon documentation, that there is a firm theoretical and operational understanding of behavior, and, with one exception, there is no further need to re-examine the hydraulic bases for use of the CheckPlus™ system in nuclear power plant FW applications. The exception, which was described in the previous paragraph, is to confirm the effect of transducer replacement on the Caldon LEFM CheckPlus™ UFM system uncertainties.

Therefore, the NRC staff has concluded, based on the considerations discussed above, that the proposed changes are acceptable with respect to reactor systems.

3.3 Electrical Systems

3.3.1 Regulatory Evaluation

The NRC staff applied the following regulatory requirements in its review of the impact of the proposed change on Seabrook's electrical systems.

General Design Criterion (GDC) 17, "Electric Power Systems," of 10 CFR Part 50, Appendix A requires that an onsite power system and an offsite electrical power system be provided with sufficient capacity and capability to permit functioning of structures, systems, and components (SSCs) important to safety.

Section 50.63 of 10 CFR requires that all nuclear plants have the capability to withstand a loss of all alternating current (ac) power (SBO) for an established period of time, and to recover therefrom.

Section 50.49 of 10 CFR, "Environmental Qualification of Electric Equipment important to Safety for Nuclear Power Plants," requires licensees to establish programs to qualify electric equipment important to safety.

3.3.2 Technical Evaluation

The NRC staff reviewed the impact of the MUR power uprate on the following electrical systems/components:

- AC distribution system
- Power block equipment (generator, exciter, transformers, isolated-phase bus duct, generator circuit breaker)
- Direct current (dc) system
- Emergency diesel generators (EDGs)
- Switchyard
- Grid stability
- SBO
- Equipment qualification program

3.3.2.1 AC Distribution System

The ac distribution system is the source of power to non-safety-related buses, and to safety-related emergency buses supplying the redundant engineered safety features (ESFs) loads. It consists of the 13,800 V system, the 4160 V system (not including the EDGs), and the 480 V system.

Seabrook LAR 05-04, Attachment 1, Table 6.1-1, "Electrical Equipment Review Summary" states that the ac distribution system is bounded by the analyses provided Seabrook in LAR 04-03, Attachment 1, Section 8.4.16.1.

The NRC staff reviewed the ac distribution system analyses previously provided for LAR 04-03, and the supplemental information provided by the licensee in its letter dated March 24, 2006. This review concluded that there are no significant changes in ac distribution system loads. Therefore, the NRC staff agrees that the analyses for the ac distribution system provided in LAR 04-03 reasonably bounds the MUR power uprate conditions.

3.3.2. Power Block Equipment (Generator, Exciter, Transformers, Iso-phase Bus Duct,Generator Circuit Breaker)

Seabrook LAR 05-04, Attachment 1, Table 6.1-1, identified that the power block equipment (except main generator and exciter) - the generator step-up transformers, unit auxiliary transformers, reserve auxiliary transformers, iso-phase bus duct, and generator circuit breaker, are bounded by the analyses provided in the Seabrook LAR 04-03, Attachment 1, Section 8.4.16.2.

The NRC staff reviewed the impact of the MUR power uprate on the main generator and exciter, including the following proposed modifications:

- Rewinding of the main generator stator
- Replacement of the Alterrex exciter with a solid-state static exciter

These modifications would increase the generator load capability to 1318 megawatts electric (MWe) with 375 MVA lagging (present values without modification are 1295 MWe and 367 MVA). The operation at these values corresponds to 0.96 lagging power factor at MUR power uprate conditions and 75 psig generator hydrogen pressure. The 1318 MWe real output of the main generator bounds the expected generator electrical output corresponding to the MUR core power level of 3648 MWt. The Alterrex excitation is being replaced with a solid-state excitation system. The new excitation system will have a high initial response with a field-forcing voltage of 200% which supports the MUR power uprate and Seabrook's commitments to Independent System Operator-New England (ISO-NE). Additionally, the plant auxiliary loads will be slightly increased from 48 megawatts (MW) and 28 MVA to 52.6 MW and 34 MVAr, mainly due to static excitation system load.

The staff reviewed the generator step-up transformers, unit auxiliary transformers, reserve auxiliary transformers, iso-phase bus duct, and generator circuit breaker analyses previously provided in Seabrook LAR 04-03, Attachment 1, Section 8.4.16.2, and the supplemental information provided by the licensee in its letter dated March 24, 2005.

The small increase in generator output (23 MWe) does not cause overloading of the generator circuit breaker or the iso-phase bus duct or the generator step-up transformer. There are no significant changes in ac distribution system loads. Therefore, the ratings of unit auxiliary transformers and reserve auxiliary transformers will not be impacted by expected MUR power uprate conditions. The impact of revised ratings of the generator, and excitation system on the grid stability is further discussed in Section 3.3.2.6.

3.3.2.3 <u>DC System</u>

FPLE stated in Seabrook LAR 05-04, Attachment 1, Table 6.1-1, that the dc system is bounded by the analyses provided in Seabrook LAR 04-03, Attachment 1, Section 8.4.16.3.

Seabrook's 125 volts direct current (Vdc) system is comprised of batteries, battery chargers and distribution equipment that supply 125 Vdc power to station loads. The nuclear safety-related (Class 1E) portion of the dc system consists of four 125 Vdc batteries, battery chargers and dc buses. It provides the source of power for direct current load groups, vital control and instrumentation, power and control of Class 1E and selected non-Class 1E electrical equipment.

The NRC staff reviewed the dc system analyses previously provided in the Seabrook LAR 04-03, Attachment 1, Section 8.4.16.3. The NRC staff identified no significant changes in dc system loads. Therefore, the NRC staff agrees that the analyses for dc system for provided in Seabrook LAR 04-03 reasonably bounds the MUR power uprate conditions.

3.3.2.4 EDGs

FPLE stated in Seabrook LAR 05-04, Attachment 1, Table 6.1-1, that the EDG system is bounded by the analyses provided in LAR 04-03, Attachment 1, Section 8.4.16.4.

The EDG system provides a safety-related source of ac power to sequentially energize and restart loads necessary to shutdown the reactor safely, and to maintain the reactor in a safe shutdown condition. The system is capable of performing this function during a loss of offsite power, with or without a coincident LOCA. There are two EDG sets of identical design, each dedicated to one of the safety-related, redundant ESF electrical trains.

The NRC staff reviewed the EDG system analyses previously provided in Seabrook LAR 04-03, Attachment 1, Sections 8.4.16.4 and 8.4.16.5. The NRC staff did not identify any significant changes in EDG system loads. Therefore, the NRC staff agrees that the analyses for the EDG system in Seabrook LAR 04-03 reasonably bounds the MUR power uprate conditions.

3.3.2.5 Switchyard

FPLE stated in Seabrook LAR 05-04, Attachment 1, Table 6.1-1, that the switchyard system is bounded by the analyses provided in the Seabrook LAR 04-03, Attachment 1, Subsection 8.4.16.6.

The switchyard equipment and associated components are classified as non-safety-related. The primary function of the 345 KV switchyard and distribution system is to connect Seabrook's electrical system to the New England transmission grid. The interconnection allows for:

- The normal flow of power out of the station to the grid when the main generator is operating, and;

- The flow of power from the grid to the station auxiliaries when the main generator is shut down.

The NRC staff reviewed the switchyard system analyses previously provided in Seabrook LAR 04-03, Attachment 1, Section 8.4.16.6. The small increase in plant output does not significantly impact the switchyard equipment. Therefore, the staff agrees that the analyses for the switchyard system for SPU reasonably bounds the MUR power uprate conditions.

3.3.2.6 Grid Stability

Grid stability is discussed in Seabrook LAR 05-04, Attachment 1, Section 6.1.3. FPLE included a copy of the "Seabrook Uprate System Impact Study" which was performed to evaluate the system impacts in accordance with "New England Power Pool (NEPOOL) Reliability Standards" and "NEPOOL Minimum Interconnection Standards." The report was prepared by General Electric Energy's Energy Consulting group based on work sponsored by ISO-NE.

The approach used in the study was to utilize NEPOOL study models, updated for the year 2007. It compares performance of the system before and after implementation of the MUR power uprate to demonstrate the impact under a prescribed set of initial conditions and contingencies established in cooperation with the NEPOOL transmission owners and ISO-NE.

The evaluation considered a calculated electrical output of 1318 MWe. This is 23 MWe above the present generator capability of 1295 Mwe.

Section 9.3 of the ISO-NE report concluded that:

> The Seabrook Phase 2 uprate [MUR power uprate] meets all system reliability criteria and requires no mitigating measures. However, as was the case for the Phase 1 uprate [SPU], since the output of Seabrook after the uprate may be greater than the 1200 MW loss of source limit for design contingencies, the following condition must be applied:
>
>> "The Seabrook unit, with implementation of its proposed 1,318 gross MW uprate or any lesser uprate, will be required to limit its gross output level in real-time operation such that the net loss of source that results from a contingent Seabrook generator trip is at or below the real-time-based maximum allowable net source loss for the NEPOOL control area. Any reductions to the gross output of Seabrook to meet this requirement will be required within 30 minutes of being directed to do so by ISO-NE."

The NRC staff reviewed the grid stability study conducted by ISO-NE and concurs that implementation of the MUR power uprate will continue to meet all system reliability criteria. Additionally, the NRC staff does not consider the condition discussed in Section 9.3 of the ISO-NE report to be inimical to this conclusion.

3.3.2.7 <u>Station Black Out</u>

FPLE stated in Seabrook LAR 05-04, Attachment 1, Table 3.1-1 (Accident/Transient Analyses Review Summary), that the SBO analysis is bounded by the analyses provided in Seabrook LAR 04-03, Attachment 1, Section 6.3.9. Section 50.63 of 10 CFR requires that each light-water cooled nuclear power plant be able to withstand and recover from a loss of all ac power, a condition referred to as SBO.

Seabrook's SBO coping duration is four hours. This is based on an evaluation of the offsite power design characteristics, emergency ac power system configuration and EDG reliability in accordance with the evaluation procedure outlined in NUMARC 87-00 and RG 1.155. The offsite power design characteristics include the expected frequency of grid-related LOOP, the estimated frequency of LOOP from severe and extremely severe weather, and the independence of offsite power.

In its supplement dated March 24, 2006, FPLE stated that the SBO analysis was performed at an analyzed core power level of 3659 MWt, which bounds the MUR operating conditions. The MUR power uprate does not impact the offsite power design characteristics, modify the emergency ac power system configuration or affect the EDG reliability. Considering this, the NRC staff agrees that the MUR power uprate will have no impact on Seabrook's SBO coping duration. Therefore, the NRC staff finds that Seabrook will continue to meet the requirements of 10 CFR 50.63.

3.3.2.8 <u>Equipment Qualification Program</u>

FPLE stated in Seabrook LAR 05-04, Attachment 1, Table 5.1-1 (Component and Program Review Summary) that the Equipment Qualification Program analysis is bounded by the analyses provided in Seabrook LAR 04-03, Attachment 1, Section 9.2.

In its letter dated March 24, 2006, the licensee stated that the environmental qualification of electrical equipment was performed at a core power level of 3659 MWt, which bounds the MUR operating conditions. Considering this, the NRC staff agrees that the MUR power uprate will have no adverse impact on Seabrook's Equipment Qualification Program.

3.3.3 Summary

Based on technical evaluation provided in Sections 3.1 through 3.8, the NRC staff agrees that implementation of the MUR power uprate, including completion of the main generator rewind and exciter replacement for increased generator output and reliability, will continue to meet the requirements of applicable sections of 10 CFR, Part 50, with respect to electrical systems.

3.4 Engineering Mechanics

3.4.1 Regulatory Evaluation

The NRC staff's review in the area of mechanical engineering covers the structural and pressure boundary integrity of NSSS and balance-of-plant (BOP) systems and components. This review focuses on the impact of the proposed MUR power uprate on (1) NSSS piping, components, and supports; (2) BOP piping, components, and supports; (3) the reactor vessel (RV) and its supports; (4) control rod drive mechanisms; (5) SGs and supports; (6) reactor coolant pumps and supports; (7) the pressurizer and its supports; (8) reactor internals and core supports; and (9) safety related valves. Technical areas covered by this review include stresses, cumulative usage factors (CUFs), flow-induced vibration, high-energy line break locations, jet impingement and thrust forces, and safety-related valve programs.

3.4.2 Technical Evaluation

The NRC staff has reviewed FPLE's application as related to the mechanical engineering areas discussed above, and determined that the existing analyses of record would bound plant operation at the proposed uprated power level. The results of the NRC staff's review of the mechanical engineering aspects of the proposed MUR power uprate are summarized in Table 3.4.

Table 3.4 Mechanical Engineering - Summary of Staff Review				
Topic	LAR 05-04 Section	UFSAR Section(s)	Bounding Analysis (Including Reference)	NRC Approved
Primary Coolant System				
Reactor Internals	Table 5.1-1, Row 5.2	4.5.2	Seabrook LAR 04-03, Attachment 1, Section 5.2	Yes
Reactor Vessel, Nozzles, and Supports	Table 5.1-1, Row 5.1	3.8.3 5.3	Seabrook LAR 04-03, Attachment 1, Section 5.1	Yes
Fuel	Table 5.1-1, Row 5.3	4.2.1	Seabrook LAR 04-03, Attachment 1, Sections 5.3 and 7.0	Yes
Control Rod Drive Mechanisms	Table 5.1-1, Row 5.4	3.8.3 4.5.1	Seabrook LAR 04-03, Attachment 1, Section 5.4	Yes
Reactor Coolant Loops	Table 5.1-1, Row 5.5	5.4.3	Seabrook LAR 04-03, Attachment 1, Section 5.5	Yes
Reactor Coolant Pumps and Motors	Table 5.1-1, Row 5.9	3.8.3 5.4.1	Seabrook LAR 04-03, Attachment 1, Section 5.8	Yes
SGs	Table 5.1-1, Row 5.8	3.8.3 5.4.2	Seabrook LAR 04-03, Attachment 1, Section 5.7	Yes
NSSS Piping and Supports	Table 5.1-1, Row 5.6	Chapter 9 10.3.1	Seabrook LAR 04-03, Attachment 1, Section 8.5.2	Yes
Pressurizer	Table 5.1-1, Rows 5.10 and 5.11	3.8.3	Seabrook LAR 04-03, Attachment 1, Sections 4.3.3.1 and 5.6	Yes
Balance-of-Plant				
BOP Piping and Supports	Table 5.1-1, Row 5.7	Chapter 9 Chapter 10	Seabrook LAR 04-03, Attachment 1, Section 8.5.1	Yes

Programs				
High Energy Line Break / Jet Impingement	Table 3.1-1, Row 3.45	Chapter 3, Appendix A	Seabrook LAR 04-03, Attachment 1, Section 10.4	Yes
Motor-Operated, Air-Operated, and Solenoid Valve Programs	Table 5.1-1, Row 5.18	3.9(B).6	Seabrook LAR 04-03, Attachment 1, Section 9.1.2	Yes

3.4.3 Summary

The NRC staff has reviewed FPLE's assessment of the impact of the proposed MUR power uprate on NSSS and BOP systems and components with regard to stresses, CUFs, and safety-related valve programs and has determined that the current analyses of record consider conditions that bound those which would follow implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the area of engineering mechanics.

3.5 Dose Consequences Analysis

3.5.1 Regulatory Evaluation

The NRC staff evaluates the potential impact of the MUR power uprate on the results of the Seabrook dose consequence analyses, guided by Sections II and III of Attachment 1 to RIS 2002-03. The review is conducted to verify that the results of the Seabrook dose consequence analyses continue to meet the acceptance criteria in 10 CFR 50.67 and GDC 19 following implementation of the MUR power uprate.

The review of the dose consequence analyses is divided into two parts, those DBAs which bound plant operation at the proposed uprated power level and those DBAs for which the existing analyses of record do not bound the proposed uprated power level. The licensee should provide a detailed discussion of the reanalysis.

3.5.2 Technical Evaluation

Sections 3 and 4 of Attachment 1 to LAR 05-04 discussed FPLE's review of the impacts of the proposed MUR power uprate on DBAs of record. As discussed previously, FPLE applied for a SPU of 5.2% for Seabrook on March 17, 2004, which was subsequently approved on February 28, 2005, as Amendment No. 101 to Facility Operating License No. NPF-86. The amendment increased the licensed core power level from 3411 MWt to 3587 MWt. To support this increase in core power level, FPLE provided a reanalysis of all DBAs based on an analyzed core power level of 3659 MWt, which bounded the requested core power level of 3587 MWt. The assumed core power level for these analyses will continue to remain bounding for the proposed MUR core power level of 3648 MWt and as such, the dose consequence analyses of the DBAs would remain valid following implementation of the MUR power uprate and assuming a 0.3% uncertainty in the core power calorimetric.

3.5.3 Summary

Based on the discussion above, the NRC staff finds that the existing dose consequence analyses will remain bounding for the proposed 1.7% MUR power uprate, considering the higher accuracy of the FW flow measurement instrumentation. These analyses will continue to show that the radiological consequences of postulated DBAs meet the dose limits given in 10 CFR 50.67 and GDC 19. Therefore, the MUR power uprate is acceptable with respect to dose consequence analysis.

3.6 Materials and Chemical Engineering

3.6.1 Regulatory Evaluation

The NRC staff's review in the area of materials and chemical engineering covers the effects of the proposed MUR power uprate on licensee programs for addressing SG tube degradation mechanisms, erosion/corrosion, and other NSSS systems.

The NRC staff also reviews the integrity of the RV and internals and pressurizer. This review focuses on the impact of the proposed MUR power uprate on pressurized thermal shock (PTS) calculations, fluence evaluations, heatup and cooldown pressure-temperature (PT) limit curves, low-temperature overpressure protection, uppershelf energy, surveillance capsule withdrawal schedules, the pressurizer shell and RV internals. This review is conducted to verify that the results of the licensee's analyses related to these areas continue to meet the requirements of 10 CFR 50.60, 10 CFR 50.61, 10 CFR 50.55a, and 10 CFR Part 50, Appendices G and H, following implementation of the proposed MUR. Additional guidance for the NRC staff's review of the topics within the vessel and internals integrity area include the guidance contained in RIS 2002-03, and RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes."

The RV material surveillance program provides a means for determining and monitoring the fracture toughness of the RV beltline materials to support analyses for ensuring the structural integrity of the feritic components of the RV. Appendix H of 10 CFR Part 50 provides the NRC staff's requirements for the design and implementation of the RV material surveillance program. The NRC staff's review primarily focused on the effects of the proposed MUR on FPLE's RV surveillance capsule withdrawal schedule.

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

The RV internals and core supports include SSCs that perform safety functions whose failure could affect safety functions by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the reactor coolant pressure boundary). The NRC staff's review covers the materials' specifications and mechanical properties, weld controls, nondestructive examination procedures, corrosion resistance, and susceptibility to degradation. The NRC's acceptance criteria for reactor internal and core support materials are based on GDC-1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of reactor internals and core supports. Matrix 1 of NRC RS-001, Revision 0, provides references to the NRC's approval of the recommended guidelines for RV internals in Topical Reports WCAP-14577, Revision 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals" (March 2001), and BAW-2248A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" (March 2000).

3.6.2 Technical Evaluation

The NRC staff reviewed FPLE's application as related to the SG tube integrity in the areas discussed above and determined that the existing analyses of record would bound plant operation at the proposed uprated power level. The results of the NRC staff's review of the SG tube integrity aspects of the proposed MUR power uprate are summarized in Table 3.6 below.

Table 3.6
Materials and Chemical Engineering - Summary of Staff Review

Topic	LAR 05-04 Section	UFSAR Section	Bounding Analysis (Including Reference)	NRC Approved
Primary Coolant System				
SG Structural Integrity Evaluation	Table 5.1-1, Row 5.8	5.4.2	Seabrook LAR 04-03, Attachment 1, Section 5.7.2	Yes
SG Tube Vibration and Wear and Other Modes of Tube Degradation	Table 5.1-1, Row 5.8	5.4.2	Seabrook LAR 04-03, Attachment 1, Section 5.7.5	Yes
RG 1.121 Analysis	Table 5.1-1, Row 5.8	1.8	Seabrook LAR 04-03, Attachment 1, Section 5.7.6	Yes

Regarding the Seabrook RV surveillance program and capsule withdrawal schedule, FPLE stated in Section 5.1.3.5 of LAR 04-03:

> The calculation determined that the maximum end-of-license [EOL] transition temperature shift using the Stretch Power Uprate (SPU) fluences for Seabrook

> Station at the end of license is less than 100 EF. Per reference 5.1-8[[1]], these end of license transition temperature shift values would require three capsules to be withdrawn from Seabrook Station, while the original withdrawal schedule in Reference 5.1-9[[2]] called for four capsules. Therefore, the current surveillance capsule withdrawal schedule remains acceptable for the SPU.

FPLE's calculation confirmed that the maximum EOL transition temperature shift using SPU fluence will remain less than 100 EF. Per the ASTM Standard Practice E185-82, these EOL transition temperature shift values would require three capsules to be withdrawn from Seabrook, while the original withdrawal schedule called for four capsules. Since the transition temperature shift using SPU fluence is less than 100 EF, the third capsule needs to be withdrawn at not less than once or greater than twice the peak EOL fluence. The licensee has already withdrawn two capsules (U and Y). Capsule V is planned to be removed when the capsule fluence reaches 2.30×10^{19} n/cm^2 (E>1.0 MeV), which occurs at 11.1 effective full power years. The peak vessel EOL fluence using SPU is 2.2×10^{19} n/cm^2 (E>1.0 MeV). Hence, FPLE's plan for the withdrawal of Capsule V is within the acceptable limit of not less than once or greater than twice the peak EOL fluence. Therefore, there is no impact of capsule withdrawal schedules because of the SPU. It follows that, because the SPU fluence bounds the MUR power uprate fluence, there is no impact on withdrawal schedules due to implementation of the MUR power uprate.

Regarding the Seabrook PTS analyses for the Seabrook RV, FPLE provided RT_{PTS} values for the beltline materials of the Seabrook vessel in LAR 04-03 and a supplement to the LAR dated October 12, 2004, concluding:

> The pressurized thermal shock calculations were performed for the Seabrook Station beltline materials using the latest procedures required by the NRC in 10 CFR 50.61. To evaluate the effects of the SPU, the pressurized thermal shock values for the beltline region materials from Seabrook Station were re-evaluated using the SPU fluences. Based on this evaluation, the reference temperature - pressurized thermal shock values will remain below the Nuclear Regulatory Commission screening criteria values using the projected SPU fluence values through end of license for 40 Effective Full Power Years for Seabrook Station and thus meet the requirements of 10 CFR 50.61.

The NRC staff evaluated the information provided by FPLE as well as the information contained in the NRC staff's Reactor Vessel Integrity Database. Using this data, and based on the fact that the SPU fluence bounds the MUR fluence, the NRC staff independently confirmed that the Seabrook RPV materials would continue to meet the PTS screening criteria requirements of 10 CFR 50.61 following implementation of the MUR power uprate.

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[1] American Society for Testing and Materials (ASTM) E185-82, Annual Book of ASTM Standards, Section 12, Volume 12.02, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels."

[2] Singer, L.R., "Public Service Company of New Hampshire Seabrook Station Unit No. 1 Reactor Vessel Radiation Surveillance Program," WCAP-10110, March 1983.

The NRC staff reviewed FPLE's evaluation of the limiting design locations of the pressurizer components. The licensee determined that changes to the hot leg and cold leg temperatures would be minimal due to the NRC-approved SPU, which bounds the MUR. FPLE determined that the parameters used in the existing design report bound the SPU conditions, therefore, the current design analysis remains bounding. FPLE also stated that the design report shows a CUF close to the allowable limit of 1.0 for the surge nozzle, and that the original design did not include an evaluation of the effects of thermal stratification on the surge line. In LAR 04-03, FPLE stated that an additional evaluation of thermal stratification was performed for Millstone Power Station, Unit No. 3, (Millstone 3) which utilizes the same pressurizer model (including critical dimensions, materials, and ASME Code of record) as Seabrook. That detailed evaluation removed excessive conservatism from the original design basis and demonstrated a significantly lower CUF (approximately 0.3) for the surge nozzle. In its submittal, FPLE concluded that, based on the comparative analysis, the CUF would remain below 1.0 even if thermal stratification were considered. As a result, the licensee concluded that the existing pressurizer components will remain adequate for plant operation at the proposed power levels. The NRC staff concurs that the data from the Millstone 3 pressurizer is applicable for comparative analysis in this case and, therefore, the Seabrook CUF is significantly lower than 1.0. Thus, the NRC staff agrees with FPLE's conclusion that operation at the proposed power levels will be within the design of the Seabrook pressurizer.

FPLE discussed the impact of the SPU (which is bounding of the MUR conditions) on the structural integrity of the Seabrook RV internal components in Section 5.2 of LAR 04-03. FPLE concluded that the SPU would not impact the safety margins associated with the structural integrity of the Seabrook RV internal components because the SPU does not significantly increase the operating temperature for the reactor coolant (based on hot leg temperature) and the SPU actually results in a decrease in neutron exposure.

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

- Cracking induced by thermal cycling (fatigue induced cracking), stress corrosion cracking, or irradiation assisted stress corrosion cracking (IASCC).

- Loss of fracture toughness properties induced by irradiation exposure for all stainless steel grades, or the synergistic effects of irradiation exposure and thermal aging for cast austenitic stainless steel (CASS) grades.

- Stress by relaxation in bolted, fastened, keyed, or pinned RV internal component induced by irradiation exposure and/or exposure to elevated temperatures.

- Void swelling (induced by irradiation exposure).

As discussed above, Matrix 1 of NRC RS-001, Revision 0, provides the NRC staff's basis for evaluating the potential for extended power uprates to induce these aging effects. Although Seabrook is not applying for an extended power uprate, the NRC staff finds that the guidance remains applicable. In Matrix 1, the NRC staff states that guidance on the neutron irradiation-related threshold levels inducing IASCC in the RV internal components are given in WCAP-14577, Revision 1-A, which established a threshold of 1×10^{21} n/cm^2 (E$ 0.1 MeV) for the initiation if IASCC, loss of fracture toughness, and/or void swelling in PWR RV internal components made from stainless steel (including CASS) or Alloy 600/82/182 materials. During review of the Seabrook SPU, the NRC staff issued a request for additional information (RAI) informing FPLE that, consistent with Matrix 1, either an inspection plan would need to be established to manage the age-related degradation in the Seabrook RV internals, or the licensee should commit to participate in the industry initiatives on age-related degradation of PWR RV internal components. In response, FPLE committed to evaluate the results of the following EPRI programs and to factor them into the RV internals inspections as appropriate:

- Material testing of baffle/former bolts removed from the Point Beach, Farley, and Ginna nuclear power plants and determination of bolt operating parameters.
- Evaluation of the effects of irradiation, which include IASCC, swelling, and stress relaxation in PWRs.
- Evaluation of irradiated material properties.
- Void swelling assessment including available data and effects on RV internals.
- Development of a long-term RV internals aging management strategy.

FPLE's commitment to participate in the industry initiatives to research the degradation of PWR RV internal components and to develop an inspection program for the RV internals that is based on the recommendations of the initiatives is consistent with Matrix 1 of NRC RS-001, Revision 0, is acceptable. Based on this assessment, and given that the SPU is bounding on the requested MUR power uprate, the NRC staff finds that FPLE has established an acceptable course of action for managing age-related degradation of the Seabrook RV internals under the proposed MUR power uprate conditions.

3.6.3 Summary

The NRC staff has reviewed FPLE's assessment of the impact of the proposed MUR power uprate on SG tube integrity, erosion/corrosion programs, RV and pressurizer integrity. Based on the above, the NRC staff concludes that FPLE has adequately addressed these impacts and has demonstrated that the plant will continue to meet the applicable requirements following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate to be acceptable with respect to the materials and chemical engineering issues discussed above.

3.7 Human Factors

3.7.1 Regulatory Evaluation

The area of human factors deals with programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC staff's human factors evaluation is conducted to confirm that operator performance will not be adversely affected as a result of system changes required for the proposed MUR power uprate. The NRC staff's review covers FPLE's plans for addressing changes to operator actions, human-system interfaces, and procedures and training required for implementation of the proposed MUR power uprate. The NRC's acceptance criteria for human factors are based on

10 CFR 50.54 Sections (i) and (m), 10 CFR 50.120, 10 CFR Part 55, GDC 19, and Generic Letter 82-33. The NRC staff's review in the area of human factors is guided by Standard Review Plan (SRP) Sections 13.2.1, 13.2.2, 13.4.2.1, and 18.0.

3.7.2 Technical Evaluation

The NRC staff has developed a standard set of questions for the review of the human factors area (NRC RIS 2002-03, Attachment 1, Section VII, Items 1 through 4). The following evaluates FPLE's addressal of these questions.

3.7.2.1 Operator Actions

FPLE identified that the impact of the MUR power uprate on operator actions has been identified and evaluated and that only minor procedure changes would be required. Included in this evaluation was an analysis of the time required to perform these actions. FPLE stated that no changes to these actions were necessary as a result of the MUR power uprate. The NRC staff finds that this satisfies Section VII.1 of Attachment 1 to RIS 2002-03.

3.7.2.2 Emergency and Abnormal Operating Procedures (AOPs)

In Section 8.2.1.1 of Attachment 1 to LAR 05-04, FPLE identifies those AOPs that will require modification to incorporate a system failure of the Caldon LEFM CheckPlus™ system. These include the applicable NSSS instrument failure AOPs and BOP instrument failure AOPs, in addition to those AOPs that reference the numeric value of RTP and calorimetric values. Additionally, FPLE stated that these revisions will take place prior to implementation of the MUR power uprate. The NRC staff concurs that the necessary procedures will be revised and, therefore, finds that FPLE's response satisfies Sections VII.2.A, VII.3, and VII.4 of Attachment 1 to RIS 2002-03.

3.7.2.3 Control Room Controls, Displays, and Alarms

In Enclosure 1 to the March 24, 2006, RAI response, FPLE provided a description of the control room controls, displays, and alarms that will be in place to utilize and monitor the Caldon LEFM CheckPlus™ UFM system. Specifically, the licensee identified that the system provides input to the main plant computer system and the calculated core power level, system process data, and diagnostic data are available through the main plant computer system.

Additionally, FPLE identified alarms for LEFM trouble, LEFM datalink trouble, LEFM uninterrupted power supply trouble, and LEFM cabinet high temperature would be annunciated in the control room through the main plant computer system video alarm system. The NRC staff reviewed FPLE's description of these alarm conditions and the operator response, and finds that they provide an adequate interface with the Caldon LEFM CheckPlus™ UFM system such that operators will be able to properly respond to potential problems with the system. FPLE identified that the changes to the safety parameter display system (SPDS) would not result in a change to the layout, monitoring, or use of the SPDS. Additionally, FPLE stated that

all modifications would be completed prior to implementation of the MUR power uprate. The NRC staff finds that this satisfies Section VII.2.B of Attachment 1 to RIS 2002-03.

3.7.2.4 Control Room Plant Reference Simulator

FPLE stated that the simulator will be upgraded to match the post-MUR plant design in accordance with the controlling standard, ANSI/ANS 3.5-1998. Additionally, FPLE stated that these changes would be incorporated prior to the implementation of the MUR power uprate such that licensed and non-licensed operator training on the MUR power uprate modifications would be conducted prior to implementation. The NRC staff finds that this satisfies Sections VII.2.C and VII.3 of Attachment 1 to RIS 2002-03.

3.7.2.5 Operator Training Program

FPLE stated that the Operations Department has been integrated into the uprate process by including a representative of the Operations Department on the uprate team. Additionally, FPLE stated that training of the Operations Department staff will occur before implementation of the MUR power uprate. The NRC staff finds that this satisfies Sections VII.2.D and VII.3 of Attachment 1 to RIS 2003-03.

3.7.3 Summary

As described above, the NRC staff has reviewed the licensee's planned actions related to the human factors area and concludes that FPLE has adequately considered the impact of the proposed MUR power uprate on changes to operator actions, procedures, plant hardware, and associated training programs to ensure that operators' performance is not adversely affected by the proposed MUR power uprate. Thus, the NRC staff concludes that FPLE will continue to meet the requirements of 10 CFR 50.54 Sections (i) and (m), 10 CFR 50.120, and GDC 19.

3.8 Plant Systems

3.8.1 Regulatory Evaluation

The NRC staff's review in the area of plant systems covers the impact of the proposed MUR power uprate on (1) containment performance analyses and containment systems, (2) safe shutdown fire analyses and required systems, (3) spent fuel pool (SFP) cooling analyses and systems, (4) flooding analyses, (5) NSSS interface systems, (6) radioactive waste systems, and (7) ESF heating, ventilation, and air conditioning systems (HVAC). Additionally, the NRC staff's plant systems review will cover FPLE's plans to modify the main FW pump turbines by replacing the last stage buckets and diaphragms to reduce long-term fatigue stresses. This review verifies that FPLE's analyses bound plant operation at the proposed MUR power level and that the results of the analyses will continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate. Guidance for the NRC staff's review is contained in Chapters 3, 6, 9, 10, and 11 of the SRP and NRC RIS 2002-03, Attachment 1, Sections II, III, and VI.

3.8.2 Technical Evaluation

The NRC staff reviewed FPLE's application as it relates to the plant systems areas discussed in Section 3.8.1, and has determined that the existing NRC-approved analyses remain bounding. A summary of the areas reviewed are contained in Table 3.8.

Table 3.8
Plant Systems - Summary of Staff Review

Topic	LAR 05-04 Section	UFSAR Section(s)	Bounding Analysis (Including Reference)	NRC Approved
LOCA Analyses				
Containment Sump pH Control	Table 3.1-1, Row 3.4	6.5.2	Seabrook LAR 04-03, Attachment 1, Section 6.1.4	Yes
Containment Structures	Table 5.1-1, Row 5.15	6.2.1	Seabrook LAR 04-03, Attachment 1, Section 8.6.1	Yes
Containment Sub-Compartments	Table 5.1-1, Row 5.16	3.8.3 6.2.1.2	Seabrook LAR 04-03, Attachment 1, Section 8.6.2	Yes
Post-LOCA Containment Hydrogen Generation	Table 3.1-1, Row 3.6	6.2.5	Seabrook LAR 04-03, Attachment 1, Section 6.1.6	Yes
Long-Term LOCA Mass and Energy Release Analysis	Table 3.1-1, Row 3.32	6.2.1.3	Seabrook LAR 04-03, Attachment 1, Section 6.4.1.1	Yes
Short-Term LOCA Mass and Energy Release Analysis	Table 3.1-1, Row 3.33	6.2.1.3	Seabrook LAR 04-03, Attachment 1, Section 6.4.1.2	Yes
Fire Protection Systems				
Fire Protection Evaluation	Table 5.1-1, Row 5.17	9.5.1	Seabrook LAR 04-03, Attachment 1, Section 9.1.1	Yes

Power/Steam Systems				
Main Steam System and Steam Dump System	Table 7.1-1, Row 7.1	10.4.4	Seabrook LAR 04-03, Attachment 1, Sections 4.3.2 and 8.4.1	Yes
Condensate and FW Systems	Table 7.1-1, Row 7.5	10.4.7	Seabrook LAR 04-03, Attachment 1, Section8.4.3	Yes[3]
Emergency FW System and Condensate Storage System	Table 7.1-1, Row 7.6	9.2.6	Seabrook LAR 04-03, Attachment 1, Section 8.4.8	Yes
FW Heaters and Drains	Table 7.1-1, Row 7.10	10.4.7	Seabrook LAR 04-03, Attachment 1, Section 8.4.8	Yes
Main Condenser Evacuation System	Table 7.1-1, Row 7.8	10.4.2	Seabrook LAR 04-03, Attachment 1, Section 8.4.6	Yes
Main Condenser and Circulating Water System	Table 7.1-1, Row 7.9	10.4.5	Seabrook LAR 04-03, Attachment 1, Section 8.4.7	Yes
SG Blowdown System	Table 7.1-1, Row 7.7	10.4.8	Seabrook LAR 04-03, Attachment 1, Section 8.4.5	Yes
Extraction Steam	Table 7.1-1, Row 7.2	10.2.2.3	Seabrook LAR 04-03, Attachment 1, Section 8.4.2	Yes
Turbine System and Auxiliaries	Table 7.1-1, Rows 7.3 and 7.4	10.4.11	Seabrook LAR 04-03, Attachment 1, Section 8.3.1	Yes
Ultimate Heat Sink	Table 7.1-1, Row 7.13	9.2.5	Seabrook LAR 04-03, Attachment 1, Section 8.4.12	Yes

[3]Modifications to the FW pump turbines will not affect FW system performance as previously evaluated.

Cooling and Support Systems				
Primary Component Cooling Water System	Table 7.1-1, Row 7.15	9.2.2	Seabrook LAR 04-03, Attachment 1, Section 8.4.13.1	Yes
Service Water System	Table 7.1-1, Row 7.14	9.2.1	Seabrook LAR 04-03, Attachment 1, Section 8.4.12	Yes
Secondary Component Cooling Water	Table 7.1-1, Row 7.16	10.4.10	Seabrook LAR 04-03, Attachment 1, Section 8.4.13.2	Yes
Containment Building Spray	Table 7.1-1, Row 7.12	6.2.2	Seabrook LAR 04-03, Attachment 1, Section 8.4.10	Yes
Radioactive Waste	Table 7.1-1, Row 7.24	Chapter 11	Seabrook LAR 04-03, Attachment 1, Section 8.4.15	Yes
SFP Cooling System	Table 7.1-1, Row 7.11	9.1.3	Seabrook LAR 04-03, Attachment 1, Section 8.4.9	Yes
Heating, Ventilation, and Air Conditioning Systems				
Control Room HVAC System	Table 7.1-1, Row 7.17	9.4.1	Seabrook LAR 04-03, Attachment 1, Section 8.4.14	Yes
Fuel Storage Building HVAC System	Table 7.1-1, Row 7.18	9.4.2	Seabrook LAR 04-03, Attachment 1, Section 8.4.14	Yes
Primary Auxiliary Building HVAC System	Table 7.1-1, Row 7.19	9.4.3	Seabrook LAR 04-03, Attachment 1, Section 8.4.14	Yes
Containment Structure Heating, Cooling, and Purge System	Table 7.1-1, Row 7.20	9.4.5	Seabrook LAR 04-03, Attachment 1, Section 8.4.14	Yes

Containment Enclosure and Adjoining Areas Cooling and Ventilation System	Table 7.1-1, Row 7.21	9.4.6	Seabrook LAR 04-03, Attachment 1, Section 8.4.14	Yes
Turbine Building HVAC System	Table 7.1-1, Row 7.22	9.4.15	Seabrook LAR 04-03, Attachment 1, Section 8.4.14	Yes
Additional HVAC Systems	Table 7.1-1, Row 7.23	9.4.4, 9.4.8 through 9.4.11, 9.4.13, and 9.4.14	Seabrook LAR 04-03, Attachment 1, Section 8.4.14	Yes

3.8.3 Summary

The NRC staff has reviewed FPLE's analyses of the impact of the proposed MUR power uprate on (1) containment performance analyses and containment systems, (2) safe shutdown fire analyses and required systems, (3) SFP cooling analyses and systems, (4) flooding analyses, (5) NSSS interface systems, (6) radioactive waste systems, and (7) ESF heating, ventilation, and HVAC systems. The NRC staff has determined that the results of FPLE's analyses related to these areas will continue to meet the applicable acceptance criteria following implementation of the MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate to be acceptable with respect to the plant systems review.

4.0 STATE CONSULTATION

In accordance with the Commission' regulations, the New Hampshire and Massachusetts State officials were notified of the proposed issuance of the amendment. The State officials had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (70 FR 67748). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

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