10 CFR 50.90

CP&L

A Progress Energy Company

Serial: RNP-RA/02-0066

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

MAY 16 2002

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

REQUEST FOR TECHNICAL SPECIFICATIONS
CHANGES TO INCREASE AUTHORIZED REACTOR POWER LEVEL

Ladies and Gentlemen:

In accordance with the provisions of the Code of Federal Regulations, Title 10 (10 CFR), Part 50.90, Carolina Power and Light (CP&L) Company is submitting a request for an amendment to the Facility Operating License, including the Appendix A Technical Specifications (TS), for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. The proposed amendment would increase the authorized reactor core power level from 2300 MWt to 2339 MWt (approximately 1.7 percent).

Specifically, the following changes are requested.

- Revision of the maximum reactor core power level stated in the Facility Operating License, paragraph 3.A, and the TS 1.1 definition of "RATED THERMAL POWER (RTP)."

- Revision of the reactor core safety limit curve in TS 2.1.1, "Reactor Core SLs."

- Revision of the reference T_{avg} value in TS 3.3.1, "Reactor Protection System (RPS) Instrumentation."

- Revision of the allowable value for the "Steam Line High Differential Pressure Between Steam Header and Steam Lines" function in TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

Robinson Nuclear Plant
3581 West Entrance Road
Hartsville, SC 29550

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- Revision of the RCS pressure-temperature limit curves in TS 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

- Revision of the Required Actions in TS 3.7.1, "Main Steam Safety Valves (MSSVs)."

- Revision of the Main Feedwater Regulation Valve (MFRV) and Bypass Valve stroke time Surveillance Requirements in TS 3.7.3, "Main Feedwater Isolation Valves (MFIVs), Main Feedwater Regulation Valves (MFRVs), and Bypass Valves."

Associated changes to the TS Bases are made in accordance with the TS Bases Control Program, as described in TS 5.5.14.

The proposed change is based on recently approved changes to the requirements of 10 CFR 50, Appendix K, "ECCS Evaluation Models," that allow recovery of measurement uncertainty in the analytical margin originally required for emergency core cooling system (ECCS) evaluation models. The proposed change also reflects guidance provided by the Nuclear Regulatory Commission (NRC) in SECY-00-0057, "Final Rule: Revision of Part 50, Appendix K, 'ECCS Evaluation Models'," and Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," and NRC-approved vendor topical reports.

Attachment I provides an affidavit as required by 10 CFR 50.30(b).

Attachment II provides a description of the current condition, a description of the proposed change, a safety assessment of the proposed change, a discussion of a finding of no significant hazards, and an environmental impact determination.

Attachment III provides a markup of the affected TS pages.

Attachment IV provides retyped pages for the proposed TS.

Attachment V provides a cross-reference to RIS 2002-03 information.

Attachment VI provides a summary of the commitments made by CP&L in this submittal.

In accordance with 10 CFR 50.91(b), CP&L is providing the State of South Carolina with a copy of the proposed license amendment.

CP&L requests that the proposed change be reviewed and approved by October 7, 2002, to support the next HBRSEP, Unit No. 2, refueling outage (RO-21), which is scheduled to begin on October 12, 2002, with the amendment being implemented within 45 days of startup from RO-21. The approval date was administratively selected to allow for NRC review, but the plant does not require this amendment to allow continued safe operation.

If you have any questions concerning this matter, please contact Mr. C. T. Baucom.

Sincerely,

B. L. Fletcher III
Manager - Regulatory Affairs

RJP/rjp

Attachments:

I. Affidavit
II. Basis for the Proposed Technical Specifications Change to Increase Authorized Reactor Power Level
III. Markup of Technical Specifications Pages
IV. Retyped Technical Specifications Pages
V. Regulatory Issue Summary 2002-03 Cross-Reference
VI. Regulatory Commitment Summary

c:

Mr. H. J. Porter, Director, Division of Radioactive Waste Management (SC)
Mr. R. M. Gandy, Division of Radioactive Waste Management (SC)
Mr. L. A. Reyes, NRC Region II
Mr. R. Subbaratnam, NRC, NRR
NRC Resident Inspector, HBRSEP
Attorney General (SC)

Affidavit

State of South Carolina
County of Darlington

J. W. Moyer, having been first duly sworn, did depose and say that the information contained in letter RNP-RA/02-0066 is true and correct to the best of his information, knowledge and belief; and the sources of his information are officers, employees, contractors, and agents of Carolina Power and Light Company.

JMoyer

Sworn to and subscribed before me

This 16th day of May 2002

Curtis A. Castell
Notary Public for South Carolina

My commission expires: Oct. 10, 2010

United States Nuclear Regulatory Commission
Attachment II to Serial: RNP-RA/02-0066
107 Pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
REQUEST FOR TECHNICAL SPECIFICATIONS
CHANGES TO INCREASE AUTHORIZED REACTOR POWER LEVEL

BASIS FOR THE PROPOSED TECHNICAL SPECIFICATIONS CHANGE
TO INCREASE AUTHORIZED REACTOR POWER LEVEL

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1.0 DESCRIPTION OF CURRENT CONDITION

The H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, Facility Operating License (OL) and Technical Specifications (TS) currently authorize plant operation at a steady state reactor core power level not in excess of 2300 MWt.

The Nuclear Regulatory Commission (NRC) has approved a change to the requirements of 10 CFR 50, Appendix K, "ECCS Evaluation Models," (as revised by the Federal Register (FR) 65 FR 34913, June 1, 2000) that provides licensees with the option of maintaining the 2-percent power margin between licensed power level and the assumed power level for Emergency Core Cooling System (ECCS) evaluations, or applying a reduced margin for ECCS evaluation. For the reduced margin ECCS evaluation option, the proposed alternative reduced margin must account for uncertainties due to power level measurement instrumentation error.

By implementing the reduced margin ECCS evaluation option of 10 CFR 50, Appendix K, and through the use of more accurate measurement equipment, it is possible for licensees to achieve a modest increase in licensed power level using current NRC-approved methodologies. It has been determined for HBRSEP, Unit No. 2, that installation of improved instrumentation for feedwater flow rate, pressure, and temperature measurement, and steam line pressure measurement, provides a sufficient reduction in power level measurement uncertainty to support a request to increase the authorized reactor core power level to 2339 MWt (approximately 1.7 percent).

2.0 DESCRIPTION OF THE PROPOSED CHANGE

Carolina Power and Light (CP&L) Company is proposing that the Facility Operating License, including the Appendix A Technical Specifications, for HBRSEP, Unit No. 2, be amended to reflect an increase in the authorized reactor core power level from 2300 MWt to 2339 MWt. This change results from a reduction in the measurement uncertainty associated with feedwater flow rate, pressure, and temperature, and steam line pressure, that is made possible through the use of more accurate measurement equipment.

The specific changes required to support the requested increase in the authorized reactor core power level are as follows:

- Facility Operating License paragraph 3.A specifies a maximum steady state reactor core power level of 2300 megawatts thermal. The maximum steady state reactor core power level will be changed to 2339 megawatts thermal.

- The Technical Specification 1.1, definition for "RATED THERMAL POWER (RTP)," identifies a total reactor core heat transfer rate of 2300 MWt. The total reactor core heat transfer rate will be changed to 2339 MWt.

- The reactor core safety limit curve in TS 2.1.1, Figure 2.1.1-1, "Reactor Core Safety Limits," indicates the High Flux Trip at 118 percent of rated thermal power. The indicated High Flux Trip will be changed to 116 percent of rated power.

- The reference T_{avg} value at RTP provided in Notes 1 and 2 of TS 3.3.1, Table 3.3.1-1, "RPS System Instrumentation," is ≤ 575.4°F. This value will be changed to reflect a reference value for T_{avg} at RTP of ≤ 575.9°F.

- The Allowable Value for the "Steam Line High Differential Pressure Between Steam Header and Steam Lines - Safety Injection" function described in TS 3.3.2, Table 3.3.2-1, "Engineered Safety Feature Actuation System Instrumentation," Function 1.e, is currently ≤ 108.95 psig. This Allowable Value will be changed to ≤ 116.24 psig, and a lower bound Allowable Value of ≥ 83.76 psig that is not currently provided in the TS will be added.

- The RCS pressure-temperature limit curves in TS 3.4.3, Figure 3.4.3-1, "Reactor Coolant System Heatup Limits Applicable Up to 24 EFPY," and Figure 3.4.3-2, "Reactor Coolant System Cooldown Limitations Applicable Up to 24 EFPY," are applicable through 24 effective full power years (EFPY) of operation. These figures will be revised to reflect applicability through 23.96 EFPY.

- Required Action A.1 of TS 3.7.1, "Main Steam Safety Valves (MSSVs)," references actions that are tied to a Thermal Power of ≤ 51% RTP. This value will be changed to ≤ 50% RTP.

- The Main Feedwater Regulation Valve (MFRV) and Bypass Valve stroke time Surveillance Requirement provided in TS 3.7.3, "MFIVs, MFRVs, and Bypass Valves," is currently ≤ 30 seconds. The MFRV and Bypass Valve stroke time will be revised to ≤ 20 seconds.

- Associated changes to the TS Bases are to be made in accordance with the Technical Specifications (TS) Bases Control Program, as described in Technical Specification 5.5.14.

Associated with the request to increase the reactor core power level to 2339 MWt, CP&L also proposes continued use of topical reports identified in TS 5.6.5.b. These topical reports describe the NRC-approved methodologies that support the HBRSEP, Unit No. 2, safety analyses, including the small break and large break loss-of-coolant accident, and the main steam line break accident analyses. Many of these topical reports refer to the use of a 2-percent uncertainty applied to reactor power, consistent with Appendix K. CP&L proposes that these topical reports be approved for use consistent with this license amendment request.

3.0 SAFETY ANALYSIS

3.1 Approach

The proposed power uprate has been evaluated with respect to its impact on the Nuclear Steam Supply System (NSSS), system design and operating parameters, interfaces between the NSSS and balance of plant (BOP) systems, design transients and accidents, and systems, components, and nuclear fuel. These evaluations were performed using well-defined analysis input assumptions and parameter values, currently approved analytical techniques, operating experience, and applicable licensing criteria and standards. The scope of the evaluations, reviews, and analyses are consistent with the methodology established in Westinghouse Topical report, WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant," issued in January 1983. The methodology presented in WCAP-10263 establishes a general approach and criteria for uprate projects, including the broad categories to be addressed.

The evaluations and analyses described herein have been completed consistent with an increase in licensed power from 2300 MWt to 2339 MWt (approximately 1.7 percent). Section 3.3 of this report discusses the revised NSSS design thermal and hydraulic parameters that were modified as a result of the power uprate and that serve as the basis for the NSSS evaluations and analyses. Section 3.4 concludes that no design transient modifications are required to accommodate the revised NSSS design conditions. Sections 3.5 through 3.7 present the system (e.g., Safety Injection (SI), Residual Heat Removal (RHR), and control systems) and component (e.g., reactor pressure vessel, pressurizer, reactor coolant pumps (RCPs), steam generators, and NSSS auxiliary equipment) evaluations completed for the revised design conditions.

Section 3.8 summarizes the effect of the uprate on the BOP (secondary) systems. Section 3.9 provides an evaluation of the effects of the power uprate on the HBRSEP, Unit No. 2, electrical systems. Section 3.10 provides the results of accident analyses and evaluations performed for the loss-of-coolant accident (LOCA), and non-LOCA transients. Sections 3.10.4 and 3.10.5 summarize the containment accident analyses and evaluations and the radiological consequence evaluations. Section 3.11 contains the results of fuel-related analyses.

The results of the analyses and evaluations performed demonstrate that relevant acceptance criteria continue to be met.

No new analytical techniques have been used to support this power uprate request. However, the dose consequences associated with the fuel handling accident, Large Break LOCA, Main Steam Line Break (MSLB), Locked Rotor, Steam Generator Tube Rupture (SGTR), and Single Rod Control Cluster Assembly (RCCA) Withdrawal have been reanalyzed separately. The fuel handling accident analysis was submitted to the NRC for review and approval by letter (RNP-RA/02-0027) dated March 13, 2002. The other dose consequence analyses were submitted to the NRC for review and approval by letter (RNP-RA/02-0067) dated May 10, 2002.

A cross-reference listing of the HBRSEP, Unit No. 2, response to RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," is provided in Attachment V of this submittal.

3.1.1 General Licensing Approach for Plant Analysis Using Plant Power Level

The reactor core and NSSS thermal power are used as inputs to most plant safety, component, and system analyses. These analyses typically evaluate the core and/or NSSS thermal power in one of four ways.

In the first case, some analyses apply an explicit 2 percent increase to the initial power level to account solely for the power measurement uncertainty. It was not necessary to re-perform these analyses for the requested power uprate conditions because the sum of the increased power level (approximately 1.7 percent) and the decreased power measurement uncertainty (approximately 0.3 percent) falls within the previously analyzed conditions. The power calorimetric uncertainty calculation described in Section 3.2 indicates that with the Leading Edge Check Plus™ flow meter (LEFM) instrumentation installed, the power measurement uncertainty (based on a 95 percent probability at the 95 percent confidence level) is approximately 0.3 percent. Therefore, these analyses only need to reflect a 0.3 percent power measurement uncertainty. Accordingly the existing 2 percent uncertainty can be reallocated such that approximately 1.7 percent is applied to provide sufficient margin to address the uprate to 2339 MWt, and approximately 0.3 percent is retained in the analysis to account for power measurement uncertainty.

For the second case, some analyses employ a nominal 100% initial power level. These analyses have either been evaluated or re-performed for the increased power level. The results demonstrate that the applicable analysis acceptance criteria continue to be met under the uprated power conditions.

The third case includes analyses that were already performed at an initial condition power level in excess of the proposed 2339 MWt. For these analyses, some of the allowable margin has been used to offset the approximately 1.7 percent power uprate. Consequently, these analyses have been evaluated to confirm that sufficient analysis margin exists to envelope the proposed power uprate.

In the final case, some of the analyses are performed at zero-percent initial condition power, or do not actually model core power level. These analyses have not been re-performed since they are unaffected by reactor core power level.

3.2 Feedwater Flow Uncertainty And Energy Measurement Uncertainty Reduction

The proposed power uprate is based on reduction of power measurement uncertainty associated with the measurement of feedwater flow, feedwater pressure and temperature, and main steam pressure. Reduction of the measurement uncertainty for these parameters substantially reduces measurement uncertainty in the associated secondary calorimetric that is used to determine reactor core power level.

The current feedwater flow measurement system at HBRSEP, Unit No. 2, consists of an in-line venturi installed in the feedwater line supplying each steam generator. An analog electronic processing system provides indication, alarm, protection and control circuits input, and input to the Emergency Response Facility Information System (ERFIS) computer for calculation of the continuous on-line secondary calorimetric.

For HBRSEP, Unit No. 2, calorimetric determination of reactor core power level can be performed using either feedwater or main steam flow measurements. The uncertainty associated with determining reactor core power level using the installed main steam flow measurement equipment is 1.72 percent of RTP, and is 1.54 percent of RTP for the feedwater flow measurement equipment. The main steam flow measurements are typically used to perform secondary calorimetric calculations due to the potential for fouling of the feedwater flow measurement venturis.

The largest contributors to uncertainty using the current methods are: tracer test measurement error, and flow and temperature measurement error. Through the use of an ultrasonic feedwater flow measurement system, measurement errors resulting from tracer testing are eliminated, and the uncertainties associated with measuring flow and temperature are significantly reduced. Additional reductions in power measurement uncertainty are also made possible through improvements in feedwater and main steam pressure measurement.

HBRSEP, Unit No. 2, is installing a Caldon LEFM Check Plus™ flow measurement system. The Caldon LEFM Check Plus™ units are chordal transit time meters. These units measure the time required for an ultrasonic pulse to travel across a pipe from one transducer to another along a chordal path that is diagonal to the fluid flow. The difference in flight times for pulses traveling with and against the fluid flow is proportional to the fluid velocity. Volumetric flow is calculated from this measured fluid velocity and known measured physical dimensions of the meter.

The LEFM Check Plus™ System is comprised of pre-fabricated piping spool pieces, or meters, that are installed in each of the feedwater headers that supply the steam generators, and an Electronic Unit that is installed in the cable spreading room. Each spool piece consists of two intersecting planes of transducer pairs, and each plane has four pairs of transducers. The configuration of sensor pairs for each meter results in precise volumetric measurement, which is further documented in Caldon Topical Reports ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System," dated March 1997; and ER-157P, "Supplement to Topical

Report ER-80P: "Basis for a Power Uprate with the LEFM Check or CheckPlus System," dated October 2001.

The LEFM Check Plus™ units are also able to calculate bulk feedwater temperature with greater precision than is measured by the currently installed temperature instrumentation. Bulk feedwater temperature is determined based on a correlation between measured feedwater pressure and sound velocity. An improved feedwater pressure transmitter is being provided to further reduce feedwater temperature measurement uncertainty.

Through the use of improved methods for determining feedwater pressure and temperature, the uncertainty associated with determining feedwater density is similarly reduced. These reductions in volumetric flow rate and density uncertainty result in an overall reduction in the measurement uncertainty for feedwater mass flow rate.

The reduction in uncertainty for feedwater bulk temperature and density, and feedwater pressure, also results in a reduction in uncertainty for the feedwater enthalpy determination. Similarly, following replacement of the existing main steam pressure transmitters, the reduction in measurement uncertainty associated with main steam pressure will result in a corresponding reduction in uncertainty associated with the main steam enthalpy determination. These reductions in uncertainty for feedwater and main steam enthalpy determination result in reduced secondary calorimetric uncertainty, and ultimately, reduce the reactor core power level measurement uncertainty.

For HBRSEP, Unit No. 2, the reduction in measurement uncertainty associated with use of an ultrasonic feedwater flow measurement system, including the use of more accurate main feedwater and main steam pressure instruments and feedwater temperature determination methods, will reduce power measurement uncertainty to approximately 0.3 percent, as shown in Table 3.2-1. As a result, the existing 2-percent analysis margin reserved for power measurement uncertainty can be reallocated such that approximately 1.7 percent is applied to provide an uprate to 2339 MWt, and approximately 0.3 percent is retained in the analysis to account for reactor core power level measurement uncertainty.

3.2.1 Compliance with the NRC Safety Evaluation Report

The Caldon LEFM Check Plus™ flow measurement system at HBRSEP, Unit No. 2, complies with Caldon Topical Reports ER-80P, and ER-157P. These Topical Reports were reviewed and approved for use by the NRC as follows:

- Caldon Topical Report ER-80P was approved in an NRC SER dated March 8, 1999.
- Caldon Topical Report ER-157P was approved in an NRC SER dated December 20, 2001.

In addition to the installation requirements provided in these Topical Reports, the NRC identified the following requirements that must be addressed by licensees requesting a license amendment based on their content. HBRSEP, Unit No. 2, complies with these requirements, as described in the following criteria.

3.2.1.1 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

Implementation of the power uprate license amendment will include development of the necessary procedures and documents required for operation, maintenance, calibration, testing, and training at the uprated power level with the new LEFM system. Plant maintenance and calibration procedures will be revised to incorporate Caldon maintenance and calibration requirements prior to declaring the LEFM system operable and raising power above the current licensed power level of 2300 MWt. The incorporation of, and continued adherence to, these requirements will assure that the LEFM system is properly maintained and calibrated.

Calibration and testing of the existing feedwater temperature instrumentation, and feedwater line and steam line flow instrumentation, will continue to be periodically performed following installation of the LEFM system. While it possible that the LEFM system could be used for calibration of the feedwater line and steam line venturi flow measuring instrumentation there are currently no plans to do so at this time.

The continuing adequacy of the LEFM system will be preserved through the use of existing programs and procedures. Administrative control of software and hardware configurations will be maintained by the programs and procedures for configuration management, configuration control of plant digital systems, and the engineering change process. Corrective actions will be conducted in accordance with procedures governing the plant corrective action program and work management process. Reporting of deficiencies to the manufacturer will be conducted in accordance with procedures governing the operating experience program and NRC reporting requirements. Receipt and addressing of manufacturer deficiency reports will be performed in accordance with the operating experience program and vendor manual control program procedures.

Operability requirements for the LEFM Check Plus™ system will be contained in the HBRSEP, Unit No. 2, Technical Requirements Manual (TRM). A Technical Requirements Manual Specification (TRMS) has been drafted for inclusion in the TRM stating that the LEFM Check Plus™ flow meter must be available to perform calorimetric heat balance measurements in order to support plant operation with reactor thermal power greater than the current licensed power level of 2300 MWt. The TRMS will be incorporated into the TRM prior to operation at the uprated power level.

When the LEFM Check Plus™ system is not used to perform calorimetric heat balance measurements (i.e., due to system unavailability), plant operation will be limited to a power level of ≤ 2300 MWt, which is consistent with the measurement uncertainty of the alternate inputs used to perform the secondary calorimetric calculation, the frequency of the TS Surveillance Requirement for performing the calorimetric heat balance calculation, and the applicable TS Required Actions. The TRM limits on plant operations account for the measurement uncertainties that are associated with the instrumentation used to provide input to the most recent secondary calorimetric calculation, thereby preserving ECCS analysis limits.

3.2.1.2 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.

<u>Response to Criterion 2</u>

This Criterion is not applicable to HBRSEP, Unit No. 2. The daily calorimetric heat balance measurements are currently performed using data obtained from either the venturis in the main steam lines or the venturis in the feedwater lines. HBRSEP, Unit No. 2, is installing a new LEFM Check Plus™ system as the basis for the requested uprate. It will be installed during the next HBRSEP, Unit No. 2, refueling outage (RO-21), which is currently scheduled to begin on October 12, 2002.

3.2.1.3 Criterion 3

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 alternative approach is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

<u>Response to Criterion 3</u>

The uncertainty values used for the HBRSEP, Unit No.2, LEFM system are based on a site-specific bounding uncertainty analysis that was performed by Caldon. This evaluation is contained in Caldon Engineering Report ER-267, "Bounding Uncertainty Analysis for Thermal Power Determination at CP&L Robinson Nuclear Power Station Using the LEFM Check Plus System." The methodologies and equations used to derive the uncertainties provided in this report are consistent with those described in Caldon Topical Reports ER-80P and ER-157P.

The uncertainty calculations documented in these topical reports are based on the methods described in ASME-PTC-19.1-1985, "Measurement Uncertainty," and ISA-RP67.04, Part II, 2000, "Methodologies for the Determination of Set Points for Nuclear Safety-Related Instrumentation." The HBRSEP, Unit No.2, engineering design procedure, EGR-NGGC-0153, "Engineering Instrument Setpoints," establishes guidance for calculating instrument setpoints, and is based on ISA-S67.04, Part I, 1994, "Setpoints for Nuclear Safety-Related Instrumentation," and ISA-RP67.04, Part II, 1994.

ISA-RP67.04, Part II, 2000, states that it is equivalent to ISA-RP67.04, Part II, 1994. A comparison of the requirements of ASME-PTC-19.1-1985 to ISA-RP67.04, Part II, 2000, performed by Caldon concluded that these documents direct performance of uncertainty determinations in a similar manner. Therefore, the uncertainty methodology employed in the Caldon reports is equivalent to the plant uncertainty methodology.

3.2.1.4 Criterion 4

For plants where the ultrasonic meter (including LEFM) was not installed and flow elements calibrated to a site-specific piping configuration (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

Criterion 4 does not apply to HBRSEP, Unit No. 2. The calibration factor for the HBRSEP, Unit No. 2, spool pieces is established by tests of these spools at Alden Research Laboratory that are scheduled for performance in July 2002. These tests include a full-scale model of the HBRSEP, Unit No. 2, hydraulic geometry and flow disturbances. The calibration factor used for the LEFM Check Plus™ system at HBRSEP, Unit No. 2, is based on this testing, and the uncertainty in the calibration factor for the spools is based on a Caldon engineering report evaluating the Alden test data. The site-specific uncertainty analysis will document these analyses.

Final acceptance of the site-specific uncertainty analyses occurs after completion of the commissioning process. The commissioning process verifies bounding calibration test data, and is conducted in accordance with the Caldon "Installation and Commissioning Manual for Carolina Power & Light, Robinson Nuclear Station," dated February 2002. This commissioning process provides final positive confirmation that actual performance in the field meets the uncertainty bounds established for the instrumentation as described in Caldon Report ER-267, "Bounding Uncertainty Analysis for Thermal Power Determination at

CP&L Robinson Nuclear Power Station Using the LEFM Check Plus System." Final commissioning is expected to occur approximately two weeks after restart from the next refueling outage (RO-21), which is currently scheduled to begin on October 12, 2002.

3.3 Nuclear Steam Supply System Operating Conditions

3.3.1 Introduction

The NSSS and steam generator operating conditions are the fundamental parameters used as input in the NSSS analyses and evaluations. As part of the increase in licensed reactor core power level from 2300 MWt to 2339 MWt, it was necessary to revise these parameters. The new parameters are identified in Table 3.3-1. These parameters have been incorporated, as necessary, into the applicable NSSS system and component evaluations, and the safety analyses performed in support of the power uprate.

3.3.2 Input Parameters and Assumptions

NSSS operating conditions at the uprated power level were determined based on best estimate inputs, such as reactor coolant system (RCS) flow, core inlet flow, and projected steam generator tube plugging (SGTP) levels. These assumptions yield primary and secondary system conditions that best indicate the way the plant operates now, as well as after the power uprate is in place.

The modified power uprate conditions include an increased NSSS power level of 2348 MWt (2339 MWt core power, plus 9 MWt non-nuclear heat sources), which corresponds to the uprated power level. The effects of SGTP under power uprated conditions were analyzed assuming both zero percent SGTP, and a bounding SGTP value of 6 percent. The existing T_{avg} versus power relationship has been extended for the uprate such that T_{avg} increases by 0.5°F, from 575.4°F to 575.9°F, and T_{cold} remains unchanged from the pre-uprate value of 547.6°F for the analysis case assuming zero percent SGTP. The value for T_{avg} also increases by 0.5°F, from 575.4°F to 575.9°F for the analysis case assuming 6 percent SGTP; however, T_{cold} decreases by 0.3°F, from 547.6°F to 547.3°F. Section 3.3.3 describes the effects of these modified input assumptions on NSSS operating conditions.

3.3.3 Results of Parameter Cases

Table 3.3-1 summarizes the NSSS operating condition cases that were developed and used as the basis for the power uprate. A description of the three uprate cases follows.

Case 1 represents current full power operating conditions, with reactor power equal to 2300 MWt, the current cold leg temperature of 547.6°F, and zero percent SGTP. The plant currently has approximately 0.1 percent SGTP.

Case 2 represents NSSS operating conditions at the uprated power level (2339 MWt) assuming zero percent SGTP. Cold leg temperature remains unchanged from the current value of 547.6°F, average reactor coolant temperature increases by 0.5°F, from 575.4°F to 575.9°F, and hot leg temperature increases from 603.2°F to 604.1°F. This case results in the greatest feedwater and steam flow rates.

Case 3 represents NSSS operating conditions at the uprated power level (2339 MWt) assuming 6 percent SGTP. Cold leg temperature decreases from the current value of 547.6°F to 547.3°F, average reactor coolant temperature increases by 0.5°F, from 575.4°F to 575.9°F, and hot leg temperature increases from 603.2°F to 604.5°F. This case results in the greatest hot leg temperature and the lowest steam generator steam temperature and pressure.

For the operating condition described in Case 2, which most closely resembles the current plant condition, the power uprate to 2339 MWt results in small changes to some of the NSSS operating conditions when compared to the current operating conditions. These small changes include the following RCS operating condition changes:

- T_{hot} increases by 0.9°F
- T_{avg} increases by 0.5°F

These small changes occur since the reactor coolant average temperature (T_{avg}) is increased while cold leg temperature (T_{cold}) remains constant at the current operating condition value of 547.6°F, while increasing core power by 39 MWt, from 2300 MWt to 2339 MWt. The temperature changes reflect the additional temperature difference across the uprated core.

In addition, the uprate results in the following secondary-side operating condition changes with zero percent SGTP:

- Steam temperature decreases by 0.4°F
- Steam pressure decreases by 2.8 psig
- Steam mass flow rate increases by 1.7 percent

These small changes occur based on a calculation of the steam generator and secondary-side performance resulting from increased core power. As a result of greater power coming from the steam generator, a higher steam flow is required, resulting in a lower steam temperature and pressure.

3.3.4 Conclusions

The various NSSS analyses and evaluations described in this document use the uprated operating conditions and current design parameters appropriate for the given analytical area. The changes seen in plant parameters from the current to the uprated operating point are commensurate with the requested power increase.

3.4 Design Transients

3.4.1 Nuclear Steam Supply Design Transients

Chapter 15 of the HBRSEP, Unit No. 2, Updated Final Safety Analysis Report (UFSAR) provides a listing of existing operational and design transients based on American Nuclear Society (ANS) categories, frequency of occurrence, initial design conditions, and associated thermal-hydraulic conditions experienced by various systems and components as a result of these transients. The proposed power uprate will neither create new types of transients, nor increase the probability of occurrence of any existing design transients.

The impact of the proposed power uprate on existing design transients was evaluated to verify that the original design transients were conservatively developed with respect to the rate and extent of pressure and temperature changes during the design basis events. The most limiting normal plant transients (i.e., plant heatup/cooldown) are limited by administrative controls and/or process limits (i.e., maximum flow rate) and are therefore not impacted by the power uprate. For more severe transients (i.e., loss-of-coolant accident, steam generator tube rupture, etc.), the evaluations were initially based on 102 percent reactor power, or some other conservatively specified value. Therefore, the thermal-hydraulic transients described in the HBRSEP, Unit No. 2, UFSAR are not adversely impacted by the power uprate.

3.5 Nuclear Steam Supply Systems

This section presents the results of evaluations and analyses performed for NSSS systems and components to support the revised operating conditions previously described in Section 3.3.3. The systems addressed in this section include fluid systems and control systems. The results and conclusions of each evaluation and analysis are provided within each subsection.

3.5.1 Reactor Coolant System

The purpose of the Reactor Coolant System (RCS) is to remove heat from the core and transfer it to the secondary side of the steam generators. Major components of the RCS include: three heat transfer loops connected in parallel to the reactor pressure vessel, with each containing a reactor coolant pump; the primary side of its respective steam generator and the associated connecting piping; and the pressurizer, pressurizer relief tank, reactor vessel flange leak detection line, and attendant interfacing piping, valves, and instrumentation.

Various assessments were performed to ensure that the RCS design basis functions could be met at the uprated power conditions. The potential impact of the uprated conditions on the RCS functions is described below:

1. The core power increase will affect the total amount of heat transferred to the main steam system. This increase in heat transfer to the main steam system is accomplished through a slight increase in hot leg temperature while holding cold leg temperature constant and maintaining a constant reactor coolant mass flow rate. The ability of the NSSS to support this increase in the normal heat removal function is addressed in Sections 3.7 and 3.8.

2. During the second phase of plant cooldown, the Residual Heat Removal (RHR) System will be required to remove larger amounts of decay heat from the RCS. This function is addressed in the RHR System evaluation in Section 3.5.4.

3. The increased thermal power can change the transient response of the RCS to normal and postulated design basis events. The transient response evaluations are based on a power level of 102 percent, which bounds the proposed power uprate. The capabilities of the NSSS control and protection functions are addressed in Sections 3.5.6 and 3.10.

4. The RCS thermal design flow does not change as a result of the uprated power conditions.

5. The pressurizer relief requirements do not change as a result of the proposed power uprate. Therefore, the following parameters are also not affected:

   - Pressurizer relief tank sizing and setpoints

   - Pressurizer relief valve sizing and discharge piping pressure drop

   - Pressurizer relief valve inlet pressure drop

   - Pressurizer surge line pressure drop

3.5.2 Safety Injection System

The function of the Safety Injection (SI) System is to remove the stored energy and fission product decay heat from the reactor core following a LOCA. The SI System also provides negative reactivity following an MSLB. The system is designed such that fuel rod damage, to the extent that would impair effective cooling of the core, is prevented.

The "active" part of the SI System consists of the high-pressure safety injection (HPSI) pumps, the low-pressure safety injection (LPSI) pumps, the refueling water storage tank (RWST), and the associated valves, instrumentation, and piping. The active portion of the SI System (injection pumps) injects borated water from the RWST into the reactor vessel following a break in either the RCS or steam system piping to cool the reactor and prevent an uncontrolled return to criticality.

The "passive" portion of the SI System includes the SI accumulator vessels that are connected to each of the RCS cold leg pipes. Each SI accumulator contains borated water under nitrogen pressure, and automatically injects into the RCS when pressure in the RCS drops below the operating pressure of the accumulators.

SI System operation is described in two phases; the injection phase, and the recirculation phase. The injection phase provides emergency core cooling and additional negative reactivity immediately following a spectrum of accidents, including LOCA and MSLB, by providing prompt delivery of borated water to the reactor vessel. The recirculation phase provides long-term post-accident cooling by recirculating water from the containment sump.

During normal operation the SI System is not required to operate. Thus, during normal operation there will be no impact on the SI System due to the power uprate. Additionally, the slight increase in RCS stored energy and decay heat resulting from the power uprate is well within the available capability of the SI System for response to design basis events.

Evaluation of SI System performance during LOCA and non-LOCA events is provided in Section 3.10. Based on the results of these evaluations, the SI System has been determined to be acceptable under power uprate conditions.

3.5.3 Chemical and Volume Control System

The Chemical and Volume Control System (CVCS) provides for boric acid addition, chemical additions for corrosion control, reactor coolant cleanup and degasification, reactor coolant makeup, reprocessing of water letdown from the RCS, and reactor coolant pump (RCP) seal injection. During plant operation, letdown flow from the RCS cold leg flows through the shell side of the regenerative heat exchanger and then through the letdown orifices. The regenerative heat exchanger reduces the temperature of the reactor coolant and the letdown orifices reduce the pressure.

The cooled, low-pressure water leaves the containment and enters the Auxiliary Building. A second temperature reduction occurs in the tube-side of the non-regenerative letdown heat exchanger followed by a second pressure reduction due to the low-pressure letdown valve. After passing through one of the mixed bed demineralizers, where ionic impurities are removed, coolant flows through the reactor coolant filter and enters the volume control tank (VCT).

In the assessment of CVCS operation at revised RCS operating temperatures, the maximum expected RCS T_{cold} must be less than the applicable CVCS design temperature to support functional operability of the system and its components. Additionally, the maximum expected RCS T_{cold} must be less than or equal to the heat exchanger design inlet operating temperature to ensure that heat exchanger design operating conditions remain bounding. The value for RCS T_{cold} will not change as a result of the power uprate. The results of these assessments indicate that the CVCS will not be adversely impacted by the power uprate.

Resizing of the CVCS equipment is not required to support the proposed power uprate. Evaluation of required boric acid concentrations is performed as part of the normal reload safety evaluation process. There will be no change to letdown and makeup requirements as a result of the proposed power uprate. The slight increase in N-16 activity that will occur at uprated conditions will have a negligible effect on letdown/excess letdown line delay time requirements.

3.5.4 Residual Heat Removal System

The Residual Heat Removal (RHR) System is designed to remove sensible and decay heat from the core and to reduce the temperature of the RCS during the second phase of plant cooldown. As a secondary function, the RHR System is used to transfer refueling water between the RWST and the refueling cavity at the beginning and end of refueling operations.

The RHR System consists of two heat exchangers, two RHR pumps, and associated piping, valves, and instrumentation. The RHR heat exchangers are of a shell and U-tube type. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell.

During plant shutdown and refueling, reactor coolant is drawn from the hot leg of RCS loop #2 by the RHR pumps, discharged through the tube side of the RHR heat exchangers, and returned to the RCS using the three cold legs. A flow control valve located in the bypass line around the RHR heat exchangers regulates the total RHR System flow. A second flow control valve regulates the flow rate through the RHR heat exchangers, and consequently the cooldown rate of the RCS.

There is no specific cooldown rate capability requirement for the RHR System, although there is a general performance expectation that the RHR System is capable of cooling the RCS to 140°F within 20 hours of reactor shutdown, and separately, that it is capable of cooling the RCS to Cold Shutdown within 72 hours under Appendix R conditions. Depending on the availability of various system components, the plant cooldown rate can vary. The increased decay heat generation rate resulting from the power uprate will cause a slight increase in the time at which the RHR System heat removal rate will match the decay heat production rate, thereby increasing plant cooldown time. This increase will not adversely impact the ability of the RHR System to cool the RCS to 140°F within 20 hours of reactor shutdown, or achieve Cold Shutdown conditions within 72 hours under Appendix R conditions, and the impact of the power uprate on the RHR System is therefore not significant.

3.5.5 Spent Fuel Pool Cooling and Purification System

The Spent Fuel Pool Cooling and Purification System provides a means for safely storing spent fuel assemblies with adequate shielding and cooling capacity to prevent any release of radioactivity to the environment. The system also provides spent fuel pool and refueling water purification to maintain water clarity for fuel handling and to minimize activity.

The spent fuel pool cooling loop consists of two 100 percent capacity pumps, heat exchanger, filter, demineralizer, associated valves, and instrumentation. Two pumps, one operating and the other as a backup, draw water from the pool to circulate through the heat exchanger prior to being returned to the pool.

The Technical Requirements Manual (TRM) specifies that the water temperature in the spent fuel pool shall be maintained below 150°F. If this temperature is exceeded, the TRM requires that fuel assemblies be moved back into the Containment Vessel to reduce heat load in the spent fuel pool.

The proposed power uprate will cause a slight increase in the decay heat load of the fuel assemblies. However, the method of maintaining spent fuel pool water temperature within specification will not be affected by the power uprate. The time to boil following a loss of spent fuel pool cooling will be slightly reduced, but will remain sufficiently long to allow operators adequate time to implement alternative cooling measures. Additionally, the type and amount of contaminants that must be removed by the purification system are not expected to increase significantly. Therefore, the Spent Fuel Pool Cooling and Purification System will not be significantly affected by the proposed power uprate.

3.5.6 NSSS Transient Control Systems and Components

The current HBRSEP, Unit No. 2, design includes the ability to tolerate certain rapid changes in plant load without generating a reactor trip or engineered safety features system actuation. These operational transients include: step load changes of 10%, ramp load changes of 5% per minute, and a 50% load rejection.

The ability of the plant to respond to load changes is governed by the following systems or attributes:

- Rod control system – which defines the speed at which core thermal power can be changed. This includes the control rod worth versus insertion characteristics, T_{avg} measurements, actuation setpoints, control system gains and lead-lag time constants.

- Feedwater control and steam dump systems - which govern how quickly the heat can be removed from the RCS. The feedwater control system attributes include the MFRV flow coefficient (Cv) versus stroke characteristics, valve motor characteristics, steam and feedwater flow measurements, setpoints, and control system gains and time constants. The steam dump system includes the valve setpoints, valve capacity, actuation times, and control system gains and time constants.

- Pressurizer pressure control system - which affects the primary system pressure response. This includes the pressurizer spray valve Cv characteristics, valve opening time, control system gains and time constants, number of operational pressurizer heater elements, and spray and heater setpoints. An anticipatory pressurizer Power Operated Relief Valve (PORV) actuation feature is provided by a lead/lag circuit at a setpoint less than the reactor trip setpoint.

- RCS fluid mass – which influences how quickly primary pressure and temperature will change as a result of a load change. RCS fluid mass will decrease slightly as a result of the power uprate since pressurizer level will be maintained at the current full-power pressurizer level and T_{avg} is increasing from 575.4°F to 575.9°F. The increase in T_{avg} will cause RCS fluid mass to decrease approximately by the T_{avg} density ratio, or 0.1 percent.

For a given absolute change in plant load (assuming no changes to the rod control, feedwater control, steam dump, and pressurizer pressure control systems), the rates of change in primary pressure and temperature will be slightly more pronounced because of the reduction in primary fluid mass. Because the change in RCS fluid mass is small, the pressure/temperature response rate change is viewed as insignificant.

It is concluded that the uprate will not have a significant effect on the ability of the plant to tolerate certain operational transients.

3.5.7 Low Temperature Overpressure Protection

The Low Temperature Overpressure Protection System (LTOPP) is designed to protect the RCS from overpressure events and low temperature brittle fracture when RCS temperature is below 350°F.

Changes to full-power operating parameters, such as NSSS power, do not impact the LTOPP system analyses. Additionally, the 10 CFR 50, Appendix G, curves and reference temperature values will not change as a result of the power uprate, although the applicable RCS pressure-temperature limit curves of TS 3.4.3, Figures 3.4.3-1 and 3.4.3-2, will be re-designated from 24 effective full power years (EFPY) of operation to 23.96 EFPY. The power uprate will not require any changes to the Low Temperature Overpressurization Protection system analyses, equipment, or instrumentation setpoints.

3.5.8 Plant Protection Systems

The plant protection systems at HBRSEP, Unit No. 2, include the Reactor Protection System (RPS) and Engineered Safety Feature Actuation System (ESFAS).

3.5.8.1 Reactor Protection System

The Reactor Protection System (RPS) monitors plant parameters related to safe operation of the reactor and initiates signals to trip the reactor under specified conditions. The RPS consists of the sensors, logic, and other equipment necessary to monitor selected NSSS parameters and containment conditions, and to effect safe and reliable control rod insertion (reactor trip) if any monitored condition or combination of monitored conditions approach specified safety system settings.

The proposed power uprate does not impact or modify any RPS hardware. However, due to power uprate-related changes in plant parameters, some instrumentation will need to be recalibrated and rescaled. The impact of the power uprate on RPS setpoints is discussed in Section 3.10.1.1.

3.5.8.2 Engineered Safety Feature Actuation System

The Engineered Safety Feature Actuation System (ESFAS) consists of the sensors, logic, and other equipment necessary to monitor selected parameters in the RCS, Main Steam System, Containment, and auxiliary systems, and to generate signals to actuate the necessary safety systems and support systems.

The proposed power uprate does not impact the safety or operational functions of the ESFAS hardware. The power uprate will necessitate some scaling and setpoint calibration revisions to reflect changes to plant operating parameters. Additionally, a change to an Allowable Value in TS 3.3.2, Table 3.3.2-1, "Engineered Safety Feature Actuation System Instrumentation," is required for Function 1.e (Safety Injection—High Steam Line Differential Pressure Between the Steam Header and Steam Lines). The revised Allowable Value reflects installation of more accurate main steam line pressure transmitters and provides a lower bound value that did not previously exist.

The impact of the power uprate on ESFAS instrumentation and setpoints is discussed in Section 3.10.1.2.

3.6 Nuclear Steam Supply Components

3.6.1 Reactor Coolant System Loss-of-Coolant Accident Forces Evaluation

The HBRSEP, Unit No. 2, UFSAR discusses protection against dynamic effects associated with the postulated rupture of piping. As discussed in the UFSAR, a double-ended guillotine break need not be postulated as a design basis event for defining structural loads or for requiring installation of pipe whip or jet impingement devices since it is assumed that the RCS piping would leak a detectable amount well in advance of any crack growth that could result in a break (e.g., Leak-Before-Break).

Evaluations have concluded that the Leak-Before-Break characteristics of the RCS piping will remain valid under power uprate operating conditions. The continued applicability of the Leak-Before-Break concept for operating conditions that bound the power uprate operating conditions was also demonstrated by Westinghouse evaluations performed in support of License Renewal and documented in WCAP-15628, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the H. B. Robinson Unit 2 Nuclear Power Plant for the License Renewal Program."

For ECCS lines (i.e., Safety Injection, Accumulator, RHR) that are attached to the RCS cold leg piping, the operating conditions, and consequently the pipe whip/jet impingement loads and break locations, are not changed by the proposed power uprate. For ECCS lines that are attached to the RCS hot leg piping, the increase in thermal stress will be insignificant. Therefore, the current pipe whip/jet impingement loads are not changed and no new break locations are postulated.

The effect of the power uprate on piping attached to the RCS and the associated supports is negligible. Evaluation of the dynamic forces associated with a rupture of a Main Steam or Feedwater line inside containment determined that because secondary pressures decrease slightly under power uprate conditions, the current pipe whip/jet impingement loads are bounding and no new break locations are postulated.

The subcompartment pressurization loads used in the design of the plant were based on the effects of a double-ended guillotine LOCA. The subcompartment loadings resulting from this analysis are bounding for any subcompartment pressurization loadings that would have to be considered using the Leak-Before-Break methodology. An evaluation of the impact of the power uprate on containment subcompartment pressurization loadings concluded that subcompartment pressurization due to a LOCA in the RCS cold legs remains bounded by the current analysis. The subcompartment pressurization due to a LOCA in the RCS hot legs will increase slightly, but the change is considered negligible.

Under power uprate conditions the current pipe whip/jet impingement loads remain bounding or have a negligible increase. Since the change in subcompartment pressurization and jet impingement loads due to the power uprate are negligible, and since the Leak-Before-Break evaluation eliminates the need to postulate a double-ended guillotine break for structural loads, the stress analyses presented in the HBRSEP, Unit No. 2, UFSAR remain bounding for power uprate conditions.

3.6.2 Reactor Coolant System Component Assessments

Assessments were performed to demonstrate the ability of RCS components to perform their design basis functions at the revised design conditions resulting from the proposed power uprate. The potential impact of the uprated design conditions on RCS components is provided in the following subsections.

3.6.2.1 Reactor Coolant System Stress and Fatigue

As shown in Table 3.6-1, the values for T_{hot} and T_{cold} under uprated power conditions are bounded by the revised RCS design condition values. Consequently, the uprated power operating conditions are bounded by the existing RCS design conditions, and the effect of operation at uprated power on thermal expansion loads and thermal stresses are also bounded by the existing design. The proposed power uprate will not adversely affect RCS component stress and fatigue.

The 10 CFR 50, Appendix G, curves and reference temperature values will not change as a result of the power uprate. However, the RCS pressure-temperature limit curves of TS 3.4.3, Figures 3.4.3-1 and 3.4.3-2, will be re-designated from 24 effective full power years (EFPY) of operation to 23.96 EFPY to reflect a conservative projection of the increase in neutron fluence associated with the power uprate. This projection will ensure that the requirements of 10 CFR 50, Appendix G, "Fracture Toughness Requirements," will continue to be met following the proposed power uprate.

There will be no impact to the Reactor Vessel coupon withdrawal and inspection frequency as a result of the power uprate.

3.6.2.2 Reactor Vessel Internals and Supports

As shown in Table 3.6-1, the temperature changes resulting from the proposed power uprate are bounded for the reactor vessel internals by the revised RCS design conditions. The effects of flow on the reactor vessel internals were evaluated. Flow rates will not increase under uprated power conditions and will have no effect on the reactor vessel internals. Similarly, the effects of increased fluence due to the power uprate were also determined to have a negligible effect on the material properties of the reactor vessel internals.

The temperature changes resulting from the proposed power uprate are also bounded by the revised RCS design conditions for the reactor vessel supports, as shown in Table 3.6-1. The effects of increased fluence have a negligible impact on the reactor vessel supports.

The power uprate will also not have a significant impact on the chemistry conditions in the RCS and will therefore not impact the compatibility of reactor vessel materials.

3.6.2.3 Leak-Before-Break

The Leak-Before-Break concept applies known mechanisms for flaw growth to piping designs with assumed through-wall flaws, and is based on the ability of plant equipment to detect leaks in the RCS. Leak-Before-Break evaluations demonstrate that postulated flaws producing detectable leakage will not become unstable when subjected to extreme loads. One of the load components used in Leak-Before-Break evaluations is thermal expansion. Thermal expansion loads resulting from uprate conditions are bounded by existing analyses, and are therefore not impacted.

3.6.2.4 Postulated Break Loadings

Postulated breaks in piping can affect RCS loads by both internal pressure differentials across components and pressure loadings external to components. The external pressure loadings, or subcompartment cavity pressure loadings, can also result in loading of the subcompartment walls.

The pressure forces resulting from pipe breaks can be affected by changes in saturation pressure and density of the process fluid. The pressure waves internal to the RCS are a function of the initial process fluid temperature and the saturation pressure. At the time of the break, the pressure will drop to the saturation pressure. At lower temperatures the saturation pressure is lower, resulting in an increase in the pressure wave magnitude due to increased process fluid density.

To evaluate the effect of changes in pressure loadings due to postulated breaks, the difference between current RCS design conditions and uprated conditions are compared. As shown in the Table 3.6-1, the uprated T_{cold} operating condition is greater than the current RCS design conditions, and the uprated T_{hot} operating condition is slightly less than the current RCS design value. The relative effects of these changes on density and the difference between process fluid pressure and P_{sat} have been evaluated and determined to be negligible.

The effects of these changes in reactor coolant temperature, saturation pressure, and density are considered negligible. Therefore, the effects of pressure loadings resulting from postulated pipe breaks on the RCS and subcompartment walls are also negligibly impacted.

3.6.2.5 Pressurizer

The power uprate will not require any physical changes to the pressurizer. As discussed in Section 3.6.2.1, a structural evaluation of RCS components concluded that operation at uprated power conditions was bounded by RCS design conditions and will not impact the stress or fatigue of any RCS components, including the pressurizer. Thermal expansion loads on the RCS supports (including the pressurizer) are also bounded by the existing analyses.

Operating temperature, pressure, and level will not change under power uprate conditions. The fluid mass in the RCS will decrease as a result of the power uprate since pressurizer level will be maintained at the current, full-power pressurizer level and T_{avg} is increasing from 575.4°F to 575.9°F. The increase in T_{avg} will cause RCS fluid mass to decrease by approximately the T_{avg} density ratio, or 0.1 percent. The rates of change in primary pressure and temperature will be slightly more pronounced because of this reduction in primary fluid mass. However, the change in RCS fluid mass is small, and the pressure/temperature response rate change is viewed as insignificant.

The pressurizer will not be adversely impacted by the power uprate.

3.6.2.5.1 Pressurizer Surge Line Thermal Stratification

The pressurizer surge line experiences pressurizer temperature at the pressurizer nozzle, and RCS hot leg temperature at the hot leg nozzle. Stratification can occur during in-surges and out-surges as the hotter pressurizer water passes over the colder water in the RCS hot leg. This temperature difference causes thermal bending moments and peak stresses in the pressurizer surge line.

To evaluate the effects of surge line thermal stratification, the change in the difference between pressurizer temperature and RCS hot leg temperature was compared for both current and uprated power conditions. The resulting difference between pressurizer temperature and RCS hot leg temperature changes by less than 1 percent and is considered negligible. Pressurizer surge line thermal stratification will not be affected by the power uprate.

3.6.2.5.2 Pressurizer Spray Line Thermal Stratification

The pressurizer spray line experiences pressurizer temperature at the pressurizer nozzle, and RCS cold leg temperature at the cold leg nozzle. Spray line thermal stratification can occur due to pressurizer spray and auxiliary spray actuations. For HBRSEP, Unit No. 2, the potential for spray line stratification is minimized since a small amount of spray flow is constantly maintained through the spray bypass valves during power operations. Additionally, there is no change in auxiliary spray temperature, and RCS cold leg temperature will not change due to the power uprate and will remain within the revised RCS design conditions with 6 percent SGTP (Table 3.6-1). Consequently, pressurizer spray line thermal stratification will not be impacted by the power uprate.

3.6.2.6 Reactor Coolant System Attached Primary Piping and Supports

The RCS piping and associated supports can be affected by temperature changes in the RCS, and also by thermal stratification in the RCS piping caused by valve leakage and turbulent penetration. As discussed in Section 3.6.2.1, RCS temperature changes resulting from the power uprate are bounded by existing analyses and will therefore have negligible effects on piping attached to the RCS.

The HBRSEP, Unit No. 2, design does not include high pressure sources of cold water that are connected to the RCS. Consequently, there is no potential for cold water leakage through valves causing thermal fatigue of RCS piping. However, even if valve leakage could potentially allow cold water to seep into hot lines, the temperature changes resulting from the power uprate are small (Table 3.3-1). The potential effects of valve leakage will not change appreciably as a result of the power uprate.

Turbulent penetration of RCS water into attached piping can also lead to thermal fatigue. RCS temperatures and flows (Table 3.3-1) will not change significantly as a result of the proposed power uprate, such that the potential for, and effects of, turbulent penetration during uprated power operations are not affected.

The effect of the power uprate on the attached RCS primary piping and supports is negligible.

3.6.2.7 Control Rod Drive Mechanisms

RCS operating conditions following the power uprate are well within the design conditions of the Control Rod Drive System. An evaluation of the increase in the heat load to the Control Rod Drive Mechanism (CRDM) coils due to the power uprate-related increase in T_{hot} determined that CRDM coil temperature could increase by approximately 1.3°F.

As described in the HBRSEP, Unit No. 2, UFSAR, the forced air cooling along the outside of the CRDM coil stacks maintains a coil temperature of approximately 392°F, and the design operating temperature of the coils is 450°F. Therefore, an increase in the coil temperature of 1.3°F will continue to provide significant margin to the design temperature.

The Rod Position Indication (RPI) coils are located above the CRDM coils and therefore are less affected by the increase in T_{hot}. Furthermore, the forced cooling air flow passes across the RPI coils before reaching the CRDM coils. Consequently, CRDM and RPI coil temperatures will not be adversely impacted by the power uprate.

3.6.2.8 Reactor Vessel/ Control Rod Drive Mechanism Head Cracking Issues

Primary Water Stress Corrosion Cracking (PWSCC) of the CRDM nozzles in the reactor vessel head has been observed in some Pressurized Water Reactors (PWRs). There are several factors that can contribute to PWSCC in these nozzles, one of which is temperature. As shown in Table 3.3-1, RCS T_{hot} will increase as a result of the power uprate by approximately 0.9°F with zero percent SGTP, and by approximately 1.3°F with 6 percent SGTP. Temperature increases of these magnitudes will not significantly increase the PWSCC susceptibility of the CRDM nozzles.

Consequently, PWSCC of the CRDM nozzles will not be impacted by the increase in RCS T_{hot} resulting from the power uprate. An inspection of the CRDM nozzles is planned for the next refueling outage (RO-21).

3.6.2.9 Reactor Coolant Pumps and Motors

As discussed in Section 3.6.2.1, a structural evaluation of RCS components concluded that operation at uprated power conditions was bounded by existing RCS design conditions and will not impact the stress or fatigue of any RCS components, including the Reactor Coolant Pumps (RCPs). There is no change in the RCP flow, NPSH available, or RCP power requirements due to the power uprate. Consequently, the power uprate has no impact on the ability of the RCPs to provide circulation of reactor coolant.

Normal RCS flow and RCP motor power will not change under power uprate conditions, since T_{cold} will not change as a result of the power uprate. RCS flow will decrease as the percentage of plugged steam generator tubes increases, and is reduced by approximately 1 percent at an SGTP of 6 percent. Consequently, the Reactor Coolant Low Flow Trip and RCP Underfrequency Trip will not change significantly under power uprate conditions, and the relationship between the RCP Undervoltage Trip and the Reactor Coolant Low Flow Trip will not be affected. Additionally, since the normal RCS flow and RCP motor power will not change significantly, the margin between the normal operating voltage on the 4KV buses and the trip setpoint will not be affected.

The existing evaluation of RCP flywheel missiles is dependent upon the maximum overspeed of the turbine generator in combination with postulated crack size and crack growth in the flywheel, none of which are impacted by the power uprate.

The proposed power uprate will not impact the RCPs or RCP motors.

3.6.2.10 Steam Generators

The power uprate will not require any physical changes to the steam generators. As discussed in Section 3.6.2.1, a structural evaluation of RCS components concluded that operation at uprated power conditions was bounded by RCS design conditions and will not impact the stress or fatigue of any RCS components, including the steam generators. Thermal expansion loads on the RCS supports (including the steam generator supports) are also bounded by the existing analyses.

Operation at higher power levels will typically yield higher fluid velocities, higher vapor void fractions, a slight reduction in recirculation ratio, and possibly less margin to fluid-elastic stability. The largest velocity increases will occur on the hot side of the upper bundle U-bend tube region where there is higher quality boiling, or on the bundle periphery.

A three-dimensional steady-state model of the steam generators at their nominal and uprated conditions was performed to evaluate U-bend gap velocities using the ATHOS 3-D thermal hydraulics code (ATHOS3). ATHOS3 is a homogeneous thermal hydraulics code designed to provide detailed primary and secondary temperatures, directional flow velocity, bulk fluid quality, void fraction, and recirculation ratios in U-tube recirculating steam generators of various types. The code predictions are based on solving continuity, momentum, and energy transport equations using a finite difference control volume method in conjunction with empirical correlations. The two-phase flow conditions are calculated based on an algebraic slip model.

The ATHOS3 study was performed assuming zero percent and 6 percent SGTP. To account for uncertainty in tube plugging scenarios, an additional multiplier of 1.05 and 1.08 was applied to the gap velocity results in the uprated zero percent and 6 percent SGTP cases, respectively. The HBRSEP, Unit No. 2, steam generators currently have approximately 0.1 percent SGTP.

The results of the ATHOS3 study concluded that the changes in gap velocities resulting from the proposed power uprate are small, with an average increase of approximately 2 percent. The general operational results also indicated that the power uprate impact would be small. The predicted drop in recirculation ratio is less than 2 percent, indicating that the changes in fluid velocities and steam qualities would be negligible.

Steam flow from the steam generators will increase from the current value of approximately 3.37×10^6 lbm/hr with a feedwater temperature of 440°F, to 3.43×10^6 lbm/hr with a feedwater temperature of 440°F under power uprate conditions. An evaluation of the impact of the higher steam flow on Flow-Induced Vibration (FIV) of the steam generator tubes concluded that the steam generator tubes were stable under power uprate conditions, and that the wear mechanism will not change from "normal" wear to the more severe "impact" wear. It was also determined that there is no significant impact on steam generator tube plug wear due to the power uprate.

Steam pressure in the steam generators will decrease from 809.0 psig to 806.2 psig, and the steam temperature will decrease from 521.8°F to 521.4°F under power uprate conditions. In addition, the average heat transfer from the reactor core to each steam generator will increase from 766.7 MWt (2.616×10^9 BTU/hr) to 779.7 MWt (2.660×10^9 BTU/hr). Analyses demonstrate that the steam generators are capable of transferring this amount of heat in addition to the heat from the RCPs and other non-nuclear heat sources.

The higher flow rates and reduced steam density result in higher steam velocities and potentially higher moisture carryover from the steam generators. The integrity of the steam generator tubes under uprated power conditions will continue to be verified through periodic inspections and measurements performed in accordance with the Steam Generator Program. This program meets the requirements of Nuclear Energy Institute (NEI) guideline

NEI-97-06, "Steam Generator Program Guidelines," and provides a balance of prevention, inspection, evaluation and repair, and leakage monitoring.

The increase in steam velocity will also result in an increase in steam hammer loads during transients such as turbine stop valve closure for components located downstream of the steam generators. The change in steam hammer loads has been evaluated and determined to be negligible.

The current level of SGTP is very close to zero percent (approximately 0.1 percent). The impact of the current SGTP rate is reflected in both the current parameters and the parameters following power uprate. A projection of the parameters after power uprate with 6 percent SGTP was also performed that demonstrated continued acceptability of the steam generators to support plant operations and analyses at the uprated power level.

3.6.2.11 Fuel Assembly Structural Integrity

The Framatome-ANP High Thermal Performance/Intermediate Flow Mixer (HTP/IFM) fuel design was evaluated for HBRSEP, Unit No. 2, to determine the impact of the proposed power uprate on fuel assembly structural integrity. The evaluation included consideration of changes to the reactor coolant system temperatures and reactor power used in the mechanical design calculations to reflect uprated power conditions. The analyses demonstrate that the mechanical design criteria for fuel rod and fuel assembly design continue to be met under power uprate conditions. Consequently, the proposed power uprate does not increase operating and transient loads and will not adversely affect the fuel assembly functional requirements. There is no impact on the LOCA structural evaluation for the fuel assemblies since the LOCA loads are not affected by the power uprate. Similarly, there is no impact on the seismic structural evaluation for the fuel assemblies since the response of the plant is not affected by the power uprate.

Fuel assembly structural integrity is not affected by the proposed power uprate, and the normal, seismic, and LOCA evaluations of the HTP/IFM fuel design for HBRSEP, Unit No. 2, remain applicable.

3.7 NSSS/BALANCE OF PLANT INTERFACE

The following BOP systems were reviewed to assess their adequacy and performance relative to the uprated operating and design conditions.

3.7.1 Main Steam System

At 100 percent RTP, normal steam generator operating pressure is currently approximately 810 psia. Normal steam generator operating pressure at 100 percent RTP will decrease slightly under uprated operating conditions, and will remain within the system design pressure with operating margin.

The following subsections summarize the evaluation of the major Main Steam (MS) System components relative to the revised power uprate conditions. The major MS System components include the steam generator Main Steam Safety Valves (MSSVs), Steam Generator (SG) Power Operated Relief Valves (PORVs), Main Steam Isolation Valves (MSIVs), Main Steam Check Valves, Main Steam Dump Valves, and the Moisture Separator Reheater (MSR) Safety Valves.

3.7.1.1 Main Steam Safety Valves

The MSSVs must have sufficient capacity so that main steam pressure does not exceed 110 percent of design pressure (the maximum pressure allowed by the ASME B&PV Code) for any anticipated transients.

The continued acceptability of the MSSVs under power uprate conditions is established by the accident analysis detailed in Chapter 15 of the UFSAR. These accident analyses model the installed MSSVs, including the pressure drop from the steam generators to the MSSVs, during various plant transients. The analysis is performed at 102 percent of current RTP, and demonstrates that peak Main Steam system pressure is maintained below 110 percent of the system design pressure under uprated power conditions. Consequently, the MSSVs have sufficient capacity to support the proposed power uprate.

Required Action A.1 of TS 3.7.1 will be revised to require that Thermal Power be reduced to < 50% RTP in the event of an inoperable MSSV, instead of the current requirement to reduce Thermal Power to < 51% RTP. The current value of 51% is based on 100 percent steam flow at a Thermal Power of 2300 MWt, and is provided to limit primary side heat generation such that overpressurization of the secondary side is precluded. The reduction in power level associated with TS Required Action A.1 reflects the effective increase in steam flow associated with a given percent Thermal Power level due to the power uprate, and the relief capabilities of the remaining operable MSSVs.

3.7.1.2 Steam Generator Power Operated Relief Valves

The Steam Generator (SG) Power Operated Relief Valves (PORVs) provide a means for decay heat removal and plant cooldown by discharging steam to the atmosphere whenever the MSIVs are closed, or when the condenser is not available. Under such circumstances, the SG PORVs, in conjunction with the Auxiliary Feedwater (AFW) System, permit the plant to be cooled down from the PORV pressure setpoint, to the point where the Residual Heat Removal (RHR) system can be placed in service. The SG PORV setpoint is below the lowest MSSV setpoint.

During plant cooldown the SG PORVs are either automatically or manually controlled. When in the automatic mode, each SG PORV controller automatically compares steam line pressure to the pressure setpoint, which is manually set by the plant operator. The SG

PORV automatic setpoint can be lowered as desired to conduct a cooldown and/or to remain at nominal hot standby temperature and pressure.

The SG PORVs are designed to provide a means of decay heat removal and plant cooldown following a loss of condenser vacuum during full power operation. As a result, functional capacity requirements for the SG PORVs are limited by the SG steaming rate that can be maintained by AFW System flow into the SGs. AFW System capacity and flow rate requirements are not impacted by the proposed power uprate. Consequently, it is concluded that the SG PORVs have sufficient capacity to support the proposed power uprate.

The SG PORVs also operate in conjunction with the Steam Dump System to increase the ability of the plant to respond to transient events, such as a secondary load rejection or turbine runback. As a result of the power uprate, rescaling of the setpoint modules for the SG PORVs will be necessary to ensure that the SG PORVs open prior to exceeding steam dump capacity.

The power uprate will not adversely impact the function or adequacy of the SG PORVs.

3.7.1.3 Main Steam Isolation Valves and Main Steam Isolation Bypass Valves

The MSIVs are located outside containment and downstream of the MSSVs. The MSIVs function to isolate forward steam flow to the main steam header. Isolation of main steam flow in the event of reverse flow is provided by the Main Steam Check Valves. The MSIVs are required to close within 5 seconds following receipt of a closure signal.

The ability of the MSIVs to close within the required time is not affected by the power uprate. The MSIVs are held open by pneumatic pressure that is released upon receipt of a valve closure signal. The MSIVs are swing-disc check valves. Based on the valve construction, increased steam flow acts to assist closure of the MSIVs. As a result, the increased steam flow and pressure drop across the MSIVs associated with operation under uprated power conditions will not affect the ability of the MSIVs to fully close within the required time.

The MSIV Bypass Valves are used to warm the main steam lines and equalize pressure across the MSIVs prior to opening the MSIVs. The MSIV Bypass Valves perform their function at no-load and low power conditions where the operating conditions will not be affected by the power uprate. Consequently, the proposed power uprate will have no impact on operation of the MSIV Bypass Valves.

3.7.1.4 Main Steam Check Valves

The Main Steam Check Valves are located downstream of the MSIVs and prevent reverse flow in the main steam lines in the event of an upstream main steam line break. The ability of the main steam check valves to close will not be adversely impacted by the proposed power uprate. The power uprate related increase in main steam forward steam flow and velocity will tend to increase the force holding the Main Steam Check Valve discs out of the flow stream and against the stop, which is the preferred operating position and is not detrimental to valve operation. The dynamic loading on the Main Steam Check Valves will also not be adversely affected under main steam line break conditions.

3.7.1.5 Main Steam Dump Valves and Steam Dump Control System

To reduce the probability of unwarranted reactor trips, the Main Steam Dump System is designed with the capacity to reject approximately 39 percent of the uprated full load main steam flow. A total of approximately 56 percent of the uprated full load main steam flow can be rejected when the Main Steam Dump System is operated in conjunction with the steam generator (SG) Power Operated Relief Valves (PORVs).

This capacity provides a means to absorb limited load rejections on the turbine-generator that occur more rapidly than the reactor power level can be reduced. In conjunction with the SG PORVs and the MSSVs, the Main Steam Dump System is designed to prevent the MS System from exceeding its design pressure during all phases of operation. The Main Steam Dump System also provides a means for removal of stored heat and decay heat from the RCS during heatup and cooldown.

The increase in steam flow under power uprate conditions will result in a slight increase in required steam dump capacity. The installed steam dump capacity will remain unchanged. However, there is sufficient steam dump capacity currently installed to satisfy Main Steam Dump System requirements under uprated power conditions.

The Steam Dump Control System is actuated by coincidence of a large load change (as sensed by main turbine 1st stage pressure), and a high error signal between T_{avg} and a programmed reference value (T_{ref}) for reactor coolant system average temperature. Predominantly as a result of replacement of the high-pressure main turbine rotor, but also partially due to the power uprate, the pressure verses power relationship for main turbine 1st stage pressure will change. This change will necessitate rescaling of the main turbine 1st stage pressure input to the steam dump controls.

Operation of the Steam Dump Control System will not be adversely impacted by the power uprate.

3.7.2 Feedwater System

The Main Feedwater (FW) System, in conjunction with the Condensate and Heater Drains and Vent Systems, must automatically maintain steam generator water level within a programmed band during steady state operations and limited transients and events. The major components of the FW system are the MFRVs, Main Feedwater Header Section Valves, Feedwater Bypass Valves, and the Main Feedwater Pumps.

The range of revised NSSS performance parameters under power uprate conditions results in a nominal feedwater volumetric flow increase of approximately 2-3 percent during full power operations. The higher feedwater flow rate has an impact on system pressure drop, which will increase as a result of the power uprate. The system has been evaluated to accommodate the increased pressure drop and flow requirements for the proposed power uprate.

3.7.2.1 Main Feedwater Regulation Valves

The MFRVs are located outside containment and have the dual function of modulating feedwater flow during normal modes of operation and isolating FW System flow after receipt of a Safety Injection (SI) signal, or other automatic closure signal.

Modification of the MFRV trim (i.e., valve internals components) is necessitated as a result of the power uprate in order to meet increased feedwater flow rate requirements. The replacement valve trim is designed for a normal operating flow of 3.46×10^6 lbm/hr, and for a maximum flow with the MFRV in the failed open position of less than 4.2×10^6 lbm/hr. These flow characteristics meet both the normal operating flow requirements under power uprate conditions, and the maximum allowable flow assumed in the containment analyses.

The safety related function of isolating the FW System flow following receipt of an SI signal, or other automatic closure signal, is not impacted by the proposed power uprate. The stroke time requirement for MFRV closure provided in TS Surveillance Requirement 3.7.3.1 is revised from ≤ 30 seconds to ≤ 20 seconds to ensure that the analyzed containment pressure during postulated accidents remains below the allowable limit.

No modifications are being made to the Main Feedwater Pumps that will result in an increase in line pressure as the MFRVs fully close. The FW System pressure upstream of the MFRVs will decrease during normal power operation due to increased feedwater flow through the pumps. The MFRV design temperature will bound the maximum feedwater temperature.

3.7.2.2 Main Feedwater Bypass Valves

The Main Feedwater Bypass Valves are located outside containment and have the dual function of modulating feedwater flow during start-up and low power operation, and isolating the FW System after receipt of an SI signal or other automatic closure signal.

The ability of the Main Feedwater Bypass Valves to modulate feedwater flow during start-up and low power operation, and to isolate the FW System following receipt of an SI signal, or other automatic closure signal, will not be impacted by the power uprate. However, the stroke time requirement for Main Feedwater Bypass Valve closure provided in TS Surveillance Requirement 3.7.3.1 is revised from ≤ 30 seconds to ≤ 20 seconds to ensure that the calculated containment pressure during postulated accidents remains below the allowable limit.

The maximum expected differential pressure in the FW System is not impacted by the power uprate, and no modifications are being made to the Main Feedwater Pumps that result in an increase in line pressure as the Main Feedwater Bypass Valves fully close. The FW System pressure upstream of the Main Feedwater Bypass Valves will decrease during normal power operation due to increased feedwater flow through the pumps. The Main Feedwater Bypass Valve design temperature will bound the maximum feedwater temperature.

3.7.2.3 Main Feedwater Header Section Valves

The Main Feedwater Header Section Valves are motor-operated valves that are located upstream of the MFRVs. The Main Feedwater Header Section Valves are required to isolate the FW System following receipt of an SI signal. The maximum expected differential pressure that the valves are required to close against will not be impacted by the power uprate, and there are no equipment modifications to the FW System that would increase the line pressure as the Main Feedwater Header Section Valves fully close. The Main Feedwater Header Section Valve design temperature bounds the maximum feedwater temperature, as identified in the heat balance.

3.7.2.4 Main Feedwater Pumps

Two, 50 percent capacity Main Feedwater Pumps are installed to feed the steam generators. Each FW pump is driven by one 6,000 hp electric motor. Following replacement of the MFRV trim, the two existing FW pumps will continue to have sufficient capacity to support the proposed power uprate. Under power uprate operating conditions, the FW pumps will operate within the preferred operating range of flows, and will not exceed their net positive suction head (NPSH) requirements or the nameplate horsepower rating of the FW pump motors.

3.7.3 Auxiliary Feedwater System

The Auxiliary Feedwater (AFW) System supplies feedwater to the secondary side of the steam generators when the normal feedwater system is not available, thereby maintaining the steam generator heat sink. The AFW System provides an alternate source of feedwater to the steam generators during normal unit startup, hot standby, and cooldown operations, and functions as an Engineered Safety Feature (ESF) System.

The AFW System is required to remove decay heat during transients and accidents. The minimum flow requirements for the AFW System are dictated by the accident analyses that have been performed at 102 percent of current rated thermal power.

The required AFW initiation time for the 10 CFR 50, Appendix R event has been evaluated for the power uprate. As discussed in Section 4.4, the required time for initiation of AFW flow under uprated power conditions, as specified in RNP Document No. FPP-RNP-300, Rev. 6, "10 CFR 50 Appendix R Section III.G Safe Shutdown Component/Cable Separation Analysis," will not be affected for the Appendix R event. The AFW initiation time continues to provide sufficient available time for the operators to actuate the AFW System.

As discussed in Section 4.3, the power uprate will result in a slight increase in the total amount of water that must be provided from the Condensate Storage Tank (CST) for the Station Blackout (SBO) event. The increase in total water volume is less than 1.3 percent in each case, and well within the administratively controlled, available CST volume.

It is concluded that the proposed power uprate will not adversely impact the safety or operational functions of the AFW System or the CST.

3.7.4 Steam Generator Blowdown System

The Steam Generator Blowdown System is used in conjunction with the chemical addition and sampling systems to maintain the chemical composition of the steam generator shell-side water within the specified limits. The blowdown system also controls the buildup of solids in the bottoms of the steam generators.

The blowdown flow rates required during plant operation are based on chemistry control and tubesheet sweep requirements to control the buildup of solids. The rate of addition of dissolved solids to the secondary system is a function of condenser leakage and the quality of secondary water makeup. The rate of generation of particulates is a function of erosion and corrosion within the secondary systems.

The present range of NSSS operating parameters permits a decrease in steam pressure from no load to full load of 216 psig (i.e., from 1025 psig to 809 psig). This is significant since the inlet pressure to the Steam Generator Blowdown System varies proportionally with operating steam pressure.

Neither condenser leakage nor secondary water makeup quality will be impacted by the proposed power uprate. Consequently, the blowdown rate required to address dissolved solids will not be impacted. The potential for erosion and corrosion will theoretically increase due to increased secondary system flow rates resulting from the uprated power conditions; however, the overall effect of these minor increases in secondary system fluid velocities is not expected to alter erosion and corrosion rates appreciably. For the blowdown flow control valves, based on the revised range of NSSS parameters associated with the power uprate, the operating range of these valves is adequate to preclude any impact on blowdown flows due to the power uprate.

As a result, the required blowdown to control secondary chemistry and particulates will not be significantly impacted by the proposed power uprate.

3.7.5 Component Cooling Water System

The Component Cooling Water (CCW) System provides an intermediate cooling loop for removing heat from reactor plant auxiliary systems and transferring it to the Service Water (SW) System during plant operation, plant shutdown, and the post-accident recovery period.

During normal operation, one CCW pump and one CCW heat exchanger have sufficient capacity to transfer the design heat load from the components served. With a maximum allowable SW temperature of 99°F, the CCW System is designed to supply 105°F water to components cooled during normal operation, and 125°F water to components cooled during normal or post-accident cooldown. The power uprate-related increase in spent fuel pool (SFP) cooling load will not adversely impact the CCW System, since the maximum SFP temperature will remain controlled within the Technical Requirements Manual.

The power uprate will result in a small increase in heat loads for the CCW System. CCW System heat removal requirements for the Residual Heat Removal (RHR) heat exchangers and other CCW System loads at the uprated power conditions are bounded by existing analyses. The peak CCW supply temperature during a design basis accident at power uprate conditions remains less than the 125°F allowable temperature. Additionally, the CCW System will continue to remove the required heat loads under normal operating conditions without exceeding the design temperature limit of 105°F. No changes or modifications to the CCW System are required to support the power uprate.

3.7.6 Service Water System

The Service Water (SW) System provides cooling water from Lake Robinson to various safety-related and non-safety-related components during power operation, plant shutdown, and the post-accident recovery period. Heat rejection to the SW System will increase slightly during normal operation as a result of the proposed power uprate. However, the SW System design pressure and temperature will not be exceeded. Additionally, SW flow demands will increase during normal operation as a result of increased turbine-generator heat loads. The increased heat load and flow demands are within the design capacities of the SW System pumps, heat exchangers, and control valves.

The timing and conduct of system alignments by operators during normal startup, standby, and cooldown will not be affected by the power uprate. Letdown and decay heat loads are manually controlled during these major evolutions. Thus, minor increases in primary and secondary system stored energy and the increase in decay heat load will not translate into a perceivable increase in SW flow requirements or primary system cooldown and heatup times.

HBRSEP, Unit No. 2, Technical Specification 3.7.8, "Ultimate Heat Sink," establishes Required Actions and Completion Times for periods when the SW inlet temperature exceeds the maximum allowed temperature of 97°F, including provisions to operate with SW inlet temperature up to 99°F if equipment conditions are favorable. Although the SW System will experience slightly higher heat loads during normal operation as a result of the power uprate, the existing system will continue to satisfy its normal and accident functions without requiring modifications to the system.

3.7.7 Ultimate Heat Sink

Plant waste heat is rejected to the Lake Robinson Ultimate Heat Sink (UHS) during normal operation and accident conditions via the open-cycle Circulating Water System and SW System. Water is taken directly from the lower end of the lake through a submerged inlet to an intake structure, and pumped through an underground conduit system for plant use. Water is discharged back to the lake near its upper end through a 4.2 mile long discharge canal.

The waste heat to the UHS during normal power operation will increase in approximately the same proportion as the requested power uprate of approximately 1.7 percent. The expected increase in average UHS temperature is less than 0.2°F following the power uprate. CP&L is subject to the monitoring requirements of the Environmental Protection Agency, as delineated under the National Pollutant Discharge Elimination System (NPDES) program. The current NPDES permit specifies limitations on discharge temperature, dam release temperature, and Circulating Water System flow. These discharge limits will not change as a result of the proposed power uprate, and HBRSEP, Unit No. 2, will continue to comply with the NPDES permit.

At the minimum allowable level of 218 ft. mean sea level (MSL), the UHS provides a 22-day supply of cooling water to the SW System pumps for accident mitigation under worst case local meteorological conditions. The existing design basis analysis of record for the UHS remains bounding with respect to the minimum available coolant inventory and level following power uprate.

3.8 BALANCE OF PLANT SYSTEMS

3.8.1 Heat Balance

Heat balance calculations were performed at an NSSS power level of 2309 MWt (2300 MWt core power, plus 9 MWt non-nuclear heat sources) and 2348 MWt (2339 MWt core power, plus 9 MWt non-nuclear heat sources). The BOP systems were evaluated using the data from these heat balances.

3.8.2 Main Feedwater System

The FW System supplies heated feedwater to the steam generators during steady-state operations and limited transients and events, and maintains steam generator water level. Steam generator level is maintained by positioning of the Main Feedwater Regulation Valve (MFRV) in the feedwater line to each steam generator.

Under the proposed power uprate, feedwater flow rate and velocity, and feedwater system pressure drop, will increase during full power operations, but will remain within acceptable limits for the components.

3.8.3 Feedwater Heater System

HBRSEP, Unit No. 2, is equipped with two parallel feedwater heating strings, each consisting of six feedwater heaters. The feedwater heater strings each consist of five low-pressure heaters and one high-pressure heater that are supplied with extraction steam for heating. The Feedwater Heater System pressure, temperature, and flow rates will change slightly at power uprate conditions.

The feedwater heaters were evaluated based on a revised heat balance reflecting power uprate conditions. These operating parameters were then compared to the design guidelines of the Heat Exchanger Institute (HEI). The HEI guidelines are derived from general experience based on equipment and demonstrated long-term operation, and provide some latitude for applying the recommended values to prevent unnecessary inspections for calculated values that only marginally exceed the recommended value.

These evaluations identified some feedwater heater components that marginally exceed HEI guidelines. In each case, the identified components marginally exceed HEI guidelines at the current power level, and will exceed the guidelines to a slightly higher degree as a result of the power uprate. For these feedwater heater components, operating experience at the current power level indicates that there are no resultant adverse effects. Based on the slight change in feedwater heater parameters, and favorable experience with these feedwater heater components under current operating conditions, it is expected that the feedwater heaters will not be adversely impacted by the power uprate.

Many of the feedwater heater components that were evaluated to exceed HEI guidelines have been recently measured and inspected under the provisions of the FAC Program. Measurement and inspection of the remaining feedwater heater components that exceed HEI guidelines, and that are composed of materials that are sensitive to FAC, will be performed prior to operating at uprated power conditions.

These feedwater heater component inspection and measurement results are entered into the Flow Accelerated Corrosion (FAC) Program models in accordance with the provisions of the FAC Program, as described in Section 4.6.5, for evaluation of continued component adequacy and inspection frequency.

Additionally, the feedwater heaters are equipped with tube-side (heaters 1 through 6) and shell-side (heaters 3 through 6) relief valves to protect against overpressure. The tube-side relief valves were evaluated with respect to the HEI guidelines and found to be acceptable for operation at uprated power conditions. The shell-side relief valves were also evaluated with respect to the HEI guidelines, but were determined to have insufficient capacity. Replacement of the shell-side feedwater heater relief valves with relief valves that meet HEI guidelines will be completed prior to operating at uprated power conditions.

3.8.4 Condensate System

The primary function of the Condensate System is to supply preheated condensate, via the feedwater heater trains, to the suction of the FW pumps. During normal operation the Condensate System must automatically maintain sufficient pressure at the FW pump suction to prevent cavitation. The major component of the Condensate System that supports the NSSS is the Condensate Pumps.

The range of revised NSSS performance parameters due to the proposed power uprate results in a nominal Condensate System volumetric flow increase of approximately 2-3 percent, and a slight increase in system pressures and temperatures. The higher Condensate System flow has an impact on system pressure drop, which will increase under uprated power conditions. The Condensate System has been evaluated to accommodate the system pressure drop for power uprate conditions. At the increased flow, the Condensate Pumps are able to provide the Net Positive Suction Head (NPSH) required at the FW pumps with substantial margin.

3.8.5 Condensate Polishing System

The Condensate Polishing System will experience increased condensate mass flow rate as a result of the proposed power uprate. The increased flow rate will also result in increased pressure drop across the system, decreased inlet pressure, and a negligible change in inlet temperature. Both the inlet pressure and inlet temperature remain well below maximum allowable values following the power uprate.

The condensate mass flow rate will increase due to the power uprate, which results in a corresponding increase in service velocity (flux) for the ion exchange resins. Polisher flux will remain below the maximum service velocity for the ion exchange resins. The increased system pressure drop resulting from the increased condensate mass flow rate will also remain below the maximum allowable pressure drop for the system.

3.8.6 Main Condenser and Condenser Vacuum System

The heat load and steam flow to the Main Condenser under power uprate conditions will result in a slight increase in condenser backpressure. The increased backpressure and flow are within acceptable limits and are bounded by the condenser design. The Condenser Vacuum System will also be adequate to support operations at the uprated power level. No modifications will be required to either the Main Condenser or Condenser Vacuum System.

3.8.7 Moisture Pre-Separator and Moisture Separator Reheater

The Moisture Pre-Separators and Moisture Separator Reheaters (MSRs) will experience slight increases in steam flows, drain flows, and operating pressures and temperatures under power uprate conditions. These increases are acceptable and continue to provide sufficient margin to design values. Additionally, the installed relief capacity of the Moisture Separator Reheater (MSR) safety valves is sufficient to accommodate the required MSR relief valve capacity at power uprate conditions. Consequently, the Moisture Pre-Separators and MSRs are adequate to support operations at the uprated power level and no modifications will be required to the Moisture Pre-Separators or MSRs.

3.8.8 Extraction Steam System

The Extraction Steam System transmits steam from the high-pressure and low-pressure main turbines to the shell side of the feedwater heaters. During normal operation, steam from the high-pressure turbine is used to heat feedwater flowing through the 5^{th} and 6^{th} point feedwater heaters, and steam from the low-pressure turbine is used to heat feedwater flowing through the 1st through 4^{th} point feedwater heaters.

Extraction steam flow will increase by approximately 2 percent as a result of the proposed power uprate for the 6^{th} point feedwater heater. System piping and components (e.g., non-return valves) were reviewed and found to be acceptable for operation at the uprated conditions.

3.8.9 Heater Drains and Vent System

The Heater Drains and Vent System and associated equipment were evaluated to ensure the ability of the system to function under power uprate conditions. Heater drain design parameters were reviewed and compared against power uprate conditions. Based on this review, it is determined that acceptable margins exist for operation at uprated power conditions. These evaluations concluded that post-power uprate pressures and temperatures remain bounded by the existing design for the Heater Drains System and its components. The heater drain control valves have been analyzed and have sufficient margin to operate at the increased flow rates that will result from the power uprate.

3.8.10 Circulating Water System

The Circulating Water (CW) System is an open-loop cooling system that provides cooling water for the main condenser of the turbine-generator unit. The cooling water is taken from and discharged to Lake Robinson.

CW System flow will remain essentially unchanged following power uprate. The increased levels of rejected heat resulting from increased turbine exhaust flow at uprated conditions will increase the CW outlet temperature by approximately 0.2°F and will cause a slight but acceptable increase in main condenser backpressure. The increase in CW outlet temperature due to the increased heat load is bounded by the CW System design and can be accommodated by Lake Robinson. No modifications to the CW System or its components are required for the proposed power uprate.

3.8.11 Main Turbine

The capability of the Main Turbine to perform at uprated power conditions has been evaluated. This evaluation included consideration of the throttle valves, high-pressure and low-pressure turbines, and associated auxiliary equipment (i.e., lube oil cooling, non-return valves, electro-hydraulic control, etc.), including the Moisture Separator Reheaters (MSRs) and MSR relief valves. The evaluation also considered the effect of power uprate on the turbine missiles analysis. The Main Turbine and auxiliary equipment components are adequate to support operation at the uprated power level without requiring equipment modifications. Additionally, the turbine missiles analysis remains bounding under uprated power conditions.

3.8.12 Turbine Component Cooling Water

Cooling water for secondary-side components, including main turbine auxiliary systems and components, is provided by the turbine building loop of the Service Water (SW) System. The SW System removes heat from secondary-side plant auxiliary systems and discharges it to Lake Robinson. The SW System is discussed in more detail in Section 3.7.6.

The main turbine auxiliary and secondary cooling water loads were evaluated at power uprate conditions. The power uprate will result in small increases to system heat load for some systems and will necessitate adjustments to cooling water flow to some components. However, the SW and secondary system designs bound these changes.

3.8.13 Balance of Plant Piping and Supports

3.8.13.1 Balance of Plant Structural Analysis

An evaluation of piping systems associated with Main Steam, Extraction Steam, Feedwater, high-pressure Heater Drains, and Condensate was performed to determine the effects of the proposed power uprate. The evaluation concluded that these piping systems remain acceptable. The piping systems will continue to satisfy the design basis requirements in accordance with the applicable design basis criteria under the temperature, pressure, and flow rate effects that will result from power uprate conditions. The HBRSEP, Unit No. 2, plant piping and related support systems will remain within code allowable stress limits.

3.8.13.2 Feedwater Hydraulic Analysis

There are no specific commitments for HBRSEP, Unit No. 2, related to water hammer. The piping configuration for the Main Feedwater System is consistent with the Westinghouse recommended configuration for feedwater connections to the steam generator to reduce the effects of water hammer. The steam generator piping configuration is not modified by the proposed power uprate.

3.8.13.3 Feedwater Thermal Stratification

A finite element analysis of the 16-inch Main Feedwater System piping to determine the effects of thermal flow stratification was previously performed. The proposed power uprate does not impact this analysis because the initial temperature for the feedwater piping and the temperature of the cold auxiliary feedwater used in the analysis bound the uprated power conditions.

3.8.13.4 Main Steam Isolation Valve/Main Steam Check Valve Trip Analysis

The worst case for the MSIV/Main Steam Check Valve Trip internal stress analysis has been determined to occur at 53 percent of current licensed power. Consequently, the proposed power uprate will not impact the effects of a MSIV/Main Steam Check Valve Trip.

3.8.14 Balance of Plant Instrument and Control Systems

The BOP instrumentation was evaluated to determine the impact of the proposed power uprate. Main turbine 1st stage pressure will increase following the next refueling outage as a result of a plant modification to replace the high-pressure turbine rotor, and to a lesser degree, operation at uprated power conditions. The increase in main turbine 1st stage pressure will necessitate rescaling of the related instrumentation and replacement of some indicators. Additionally, the changes in power train parameters will necessitate retuning of the Main Feedwater Control System.

3.9 ELECTRICAL SYSTEMS

3.9.1 Main Generator and Auxiliaries

The Main Generator nameplate rating is 854.1 MVA at 22 kV, when operating at 60 Hz, with 75 psig hydrogen pressure, and a 0.9 power factor.

The gross electrical output of the Main Generator varies depending on seasonal fluctuations. For 2001, these variations ranged between a peak gross generator output of approximately 736 MWe (summer) and 767 MWe (winter). The gross generating capacity of the main generator is estimated to increase by approximately 20 MWe as a result of the combined effects of the power uprate and replacement of the high-pressure turbine rotor.

The approximate increase in main generator gross generating capacity attributable solely to the power uprate is determined by the ratio of the uprated NSSS power level (2339 MWt, plus 9 MWt non-nuclear sources), and the current NSSS power level (2300 MWt, plus 9 MWt non-nuclear sources). Consequently, the increase in gross generating capacity due to the power uprate will be approximately 12.4 MWe (summer) and 13.0 MWe (winter).

Main Generator gross power generation capability was evaluated for uprated power conditions at 780 MWe (summer) and 800 MWe (winter), with a maximum gross generator output of 800 MWe and 250 MVAR (the MVAR metering scale limit). This output level corresponds to approximately 838 MVA at a 0.954 PF. This value is within the Main Generator nameplate rating, and is within the generator reactive capability curve at 68 psig generator hydrogen pressure. Main Generator capacity is adequate to support the power uprate and the generator will continue to be operated to produce power within its reactive capability curve (i.e., VAR, VA, MWe).

The generator exciter has a nameplate rating of 4700 kW, 550 V, 8545 amps. An evaluation of excitation requirements for operation at 800 MWe with a 0.93 PF indicates that the required excitation capability is 4240 kW, 525 V, 7355 amps, which is well within the generator exciter capability.

HBRSEP, Unit No. 2, is equipped with an isophase bus to deliver electrical power from the generator to the transmission system. The isophase bus is rated at 25,000 amps, with forced air-cooling provided. Normal loadings on the isophase bus range from 19,500 to 21,500 amps, and are not expected to exceed 22,000 amps due to the proposed power uprate. The isophase bus capacity is adequate for power uprate operation without modification.

Additional evaluations were performed for the rotor winding, stator winding, stator core main body, stator core end pack, parallel rings, hydrogen coolers, and main leads, bushings, and flexible connectors. These evaluations concluded that these components are adequate for operation at uprated power conditions.

An evaluation of main generator protective relay settings did not identify any required changes to equipment or protective relay settings.

The Main Generator and associated support systems are not adversely affected by the power uprate.

3.9.2 Power Conversion System

The Power Conversion System consists of the components between the output of the main generator and the transmission grid outside of HBRSEP, Unit No. 2. The main components of the Power Conversion System include the main transformer and the electrical equipment in the HBRSEP, Unit No. 2, switchyard.

These components have been evaluated in two studies: 1) "Evaluation of Transmission Parameters for Robinson Generator #2 Power Uprate," and 2) "Assessment of the Effect of Power Uprate on Generator No. 2 Protective Relaying." These studies were performed by the CP&L Transmission Department, and concluded that the existing Power Conversion Systems are adequately sized for the maximum plant electrical output following the power uprate.

3.9.3 Auxiliary Power System

The Auxiliary Power System includes the 4160V, 480V, vital 120V AC and 125V DC electrical systems, and the Start-up and Unit Auxiliary Transformers.

The Main Feedwater Pumps and Condensate Pumps will operate with slightly increased flow under uprated power conditions, which will increase their motor brake horsepower requirements. The motors for these pumps have been evaluated and it has been determined that the Main Feedwater Pump motors will remain within their nameplate motor rating.

The required motor shaft horsepower for the Condensate Pumps is currently above the nameplate motor rating, but well within the 1.15 service factor rating. The Condensate Pumps will continue to operate above the nameplate motor rating under uprated power conditions, and will remain within the 1.15 service factor rating. Sufficient margin will continue to exist in the service factor for the Condensate Pump motors, and there will be no adverse impact as a result of the power uprate. The effect of the change in motor load for the Condensate Pumps on voltage drops and terminal voltages at other loads supplied by the same bus and transformer is negligible.

The Heater Drain Pumps and Service Water Pumps will also have slightly increased flow due to the proposed power uprate; however, the actual motor brake horsepower requirements for these pumps will be reduced compared to current motor brake horsepower requirements.

Since the Main Feedwater Pump, Condensate Pump, and Heater Drain Pump and Service Water Pump motor loads are below their nameplate horsepower rating (or service factor rating), and either negligibly increased, or reduced, the current evaluations for the associated motors and busses are still valid for the proposed power uprate.

Loads on the 120V AC and 125V DC systems will not increase due to the power uprate since these systems feed components that have power requirements that do not vary with plant operating power level.

3.9.4 Standby Power System

The Standby Power System consists of three diesel generators: Emergency Diesel Generator A, Emergency Diesel Generator B, and the Dedicated Shutdown Diesel Generator. The proposed power uprate will not add or remove loads from the Standby Power System, and the electrical loads that will be changed by the power uprate are not fed from standby power. As a result, the Standby Power System will not be impacted by the proposed power uprate.

3.9.5 Grid Stability

CP&L has performed a grid stability analysis to support the proposed power uprate. The grid system analysis was performed using a bounding generator output value of 810 MWe gross, which is in excess of the maximum expected post-uprate generator output, as discussed in Section 3.9.1. Use of the higher value is a conservative assumption for performing stability analysis, since higher output tends to reduce generator stability.

The analysis included simulated disturbances that exceeded those normally required by the North American Electric Reliability Council (NERC) Planning Standards. The simulated disturbance was a three-phase fault with delayed clearing.

The results of the grid stability analysis indicate that there is no adverse effect on grid stability, and that both HBRSEP, Unit No. 2, and the area transmission system (including local area generating units) will not be adversely affected by the power uprate. Additionally, the power uprate will not adversely impact the availability of offsite power for HBRSEP, Unit No. 2, auxiliary loads in the event of a unit trip.

3.10 NUCLEAR STEAM SUPPLY SYSTEM ACCIDENT EVALUATION

3.10.1 Reactor Protection System and Engineered Safety Feature Actuation System Setpoints

As described in Sections 3.10.1.1 and 3.10.1.2, plant protection for HBRSEP, Unit No. 2, is provided by the Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS).

The RPS initiates reactor shutdown based on the values of selected plant parameters, thereby protecting against violating core fuel design limits and the Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (AOOs), and to assist the Engineered Safety Features in mitigating accidents.

The RPS protection and monitoring systems are designed to assure safe operation of the reactor. This is achieved, in part, through specification of Limiting Safety System Settings (LSSS) for the parameters monitored by the RPS. These LSSS are reflected in the HBRSEP, Unit No. 2, Technical Specifications (TS) as Allowable Values.

The ESFAS initiates necessary safety systems based on the values of selected plant parameters to protect the core fuel design limits and RCS pressure boundary, and to mitigate accidents.

The instrument setpoint methodology utilized by HBRSEP, Unit No. 2, includes consideration of instrument channel uncertainties to ensure that automatic protective action will occur to protect against the most severe abnormal situation without exceeding analytical safety limits.

Instrument channel uncertainties are the combination of error effects that are inherent with instrument channel components. The methods used to derive the Nominal Trip Setpoints and Allowable Values reflected in the TS are not affected by the power uprate.

The effect of the proposed power uprate on process measurement uncertainty for the RPS and ESFAS instrument channels is discussed below.

3.10.1.1 Reactor Protection System Setpoints

A list of automatic reactor trips, means of actuation, and the coincident circuit requirements is given in the TS Bases, and is summarized below. Certain reactor trip channels are automatically bypassed at low power where they are not required for safety. Nuclear Source Range and Intermediate Range Trips are specifically provided for protection at low power or subcritical operation. For higher power operations, these trips are bypassed by manual action.

Constraints that are selected for various core parameters to assure that departure from nucleate boiling ratio (DNBR) is not less than the safety limit are determined by a digital code which mathematically correlates the nuclear and thermal hydraulic properties of the primary system. The trip settings remain more restrictive than the core safety limits, and are used in the protection system to provide suitable margin for measurement and instrument errors. The setpoints and allowable values for the automatic RPS functions are listed in TS Table 3.3.1-1.

The proposed power uprate does not affect the design, function, physical configuration, or actuation logic of the RPS. Process parameters will remain within the current range of the instrumentation. There will be no impact on the failure modes and effects evaluations, the single failure analyses, or the interactions between control systems and the RPS.

The design basis functions of the RPS, and the impact on these functions due to the power uprate, are described as follows based on how each RPS function performs in response to the analyzed transients.

The proposed power uprate will necessitate a change to the TS value of T_{avg} used to calculate the Overtemperature ΔT (OTDT) and Overpower ΔT (OPDT) Trip setpoints, and the update of several instrument calculations to reflect uprated power conditions. However, it is concluded that the power uprate will not impact the safety or operational functions of the RPS.

Power Range Neutron Flux

The High Nuclear Flux Trip (Power Range Neutron Flux Trip) is generated when two of the four power range channels read above the trip setpoint. There are two independent trip settings, a high and a low setting. The high trip setting provides protection during normal power operation. The low setting provides protection during startup, and can be manually bypassed when two out of the four power range channels read above approximately 10 percent power (P-10). Three out of the four channels below 10 percent automatically reinstates the trip. The high trip setting is always active. This trip function is credited in the UFSAR Chapter 15 accident analysis, and the nominal trip setpoint is provided in TS Table 3.3.1-1.

The UFSAR Chapter 15 analyses that utilize the High Nuclear Flux (Power Range) Trip - High function were performed at 102 percent of the current licensed power limit of 2300 MWt, or 2346 MWt, in order to reflect power measurement determination uncertainty. As a result of improved secondary flow measurement accuracy and better heat balance determination capabilities, the proposed 100 percent power level will be 2339 MWt, and the Chapter 15 analyses that utilize the High Nuclear Flux (Power Range) Trip - High function will continue to be performed at 2346 MWt.

The current analysis value (analytical limit) for the overpower trip is 118 percent. The thermal power associated with this analytical limit will not change as a result of the power uprate; however, it will be re-designated as 116 percent as a result of the power uprate (e.g., $(1.18*2300)/2339 \cong 1.16$). The re-designated limit will continue to provide sufficient margin between the analysis value and the nominal TS Trip Setpoint value, including the associated instrument uncertainty. It will be necessary to update instrument calculations for the High Nuclear Flux (Power Range) Trip - High Trip function to reflect the increased power level. There will be no change to the TS value and no effect on the RPS trip setpoint as a result of the power uprate.

As a result of this change in the analytical limit for the Power Range Neutron Flux Trip, the High Flux Trip indicated on TS Figure 2.1.1-1, "Reactor Core Safety Limits," will be revised from the current value of 118 percent of rated power to reflect the revised analytical value of 116 percent of rated power. This change will ensure that the overpower limits assumed in the thermal-hydraulic design analysis for the nuclear fuel are maintained.

Similarly, the analysis value for the High Nuclear Flux (Power Range) Trip - Low function will be reduced from 35 percent to 34.4 percent. The proposed power uprate will not necessitate hardware changes or TS impacts for the Power Range Neutron Flux trips.

Intermediate Range Neutron Flux

The High Nuclear Flux (Intermediate Range) Trip is generated when one out of the two intermediate range channels reads above the trip setpoint. This trip provides protection during reactor startup, and can be manually bypassed if two out of four power range channels are above approximately 10 percent (P-10). Three out of four channels below this value automatically reinstate the trip. The Intermediate Range channels (including detectors) are separate from the power range channels.

The Intermediate Range Trip and permissive are not credited in the UFSAR Chapter 15 accident analyses. Consequently, maintaining the existing setpoints will not affect the Chapter 15 accident analyses. The nominal trip setpoint is provided in TS Table 3.3.1-1. There is no change required as a result of the power uprate.

Source Range Neutron Flux

The High Nuclear Flux (Source Range) Trip is generated when one of the two Source Range channels reads above the trip setpoint. This RPS function provides protection during reactor startup and can be manually bypassed when one of two Intermediate Range channels reads above the P-6 setpoint value, which is set to provide approximately one decade of overlap between the source range and intermediate range flux instrumentation. The Source Range Neutron Flux RPS function is automatically reinstated when both Intermediate Range channels decrease below the P-6 setpoint value. This trip is also bypassed by two out of four power range channels are above the P-10 setpoint, and can be reinstated below P-10 by manual operator action. The trip point is set between the Source Range cutoff power level and the maximum Source Range power level.

The source range instrumentation is used during startup and shutdown and during low power operation. The nominal trip setpoint is provided in TS Table 3.3.1-1. There is no impact on the Source Range Trip due to the power uprate.

Overtemperature ΔT

The OTDT Trip protects the core against departure from nucleate boiling (DNB) by generating a reactor trip on coincidence of two out of the three signals, with one set of temperature measurements per loop. The setpoint is continuously calculated for each loop. This trip is used in the Chapter 15 analysis, and the equation used to calculate the trip setpoint is provided in TS Table 3.3.1-1.

The OTDT reactor trip equation contains two terms that are affected by the power uprate. These terms are Reference T_{avg} (currently 575.4° F), used as a comparison temperature to operating T_{avg}, and ΔTo (currently rated at 2300 MWt power). The proposed power uprate will not necessitate hardware changes for the OTDT Trip other than instrument rescaling. However, the value for T_{avg} provided in TS Table 3.3.1-1, Note 1, will be revised from 575.4 °F to 575.9 °F to reflect the change in power uprate operating conditions, as shown in Table 3.3-1 of this submittal.

Overpower ΔT

The OPDT Trip protects against excessive power level (fuel rod rating protection) and fuel rod linear heat generation rate (LHGR) by generating a reactor trip on a coincidence of two out of the three signals, with one set of temperature measurements per loop. The setpoint is continuously calculated for each loop. The equation used to calculate the trip setpoint is provided in TS Table 3.3.1-1.

The OPDT reactor trip equation contains two terms that are affected by the power uprate. These terms are Reference T_{avg} (currently 575.4° F), used as a comparison temperature to operating T_{avg}, and ΔTo (currently rated at 2300 MWt power). The proposed power uprate will not necessitate hardware changes for the OPDT Trip other than instrument rescaling. However, the value for T_{avg} provided in TS Table 3.3.1-1, Note 2, will be revised from 575.4 °F to 575.9 °F to reflect the change in power uprate operating conditions, as shown in Table 3.3-1 of this submittal.

Pressurizer Pressure – High

The High Pressurizer Pressure Trip limits the range of required protection from the OTDT Trip and protects against RCS overpressure. The reactor is tripped on coincidence of two out of the three high pressurizer pressure signals. This trip is used in the UFSAR Chapter 15 analysis, and the nominal trip setpoint is provided in TS Table 3.3.1-1.

The power uprate will have no impact on the ability of the High Pressurizer Pressure Trip function. Each accident for which the trip is credited to respond has been analyzed based on a power level of 102 percent of RTP. Since this uprate remains bounded by the existing analyses, The High Pressurizer Pressure Trip function will continue to be adequate.

Pressurizer Pressure - Low

The Low Pressurizer Pressure Trip protects against excessive core steam voids that could lead to DNB. A lead-lag filter is applied to the pressure signal to compensate for transient pressure overshoot. The circuit trips the reactor on coincidence of two out of the three low pressurizer pressure signals. This trip is blocked when three of the four power range channels and both turbine first stage pressure channels read below approximately 10 percent power (P-10 and P-7, respectively). This trip is used in Chapter 15 analysis, and the nominal trip setpoint is provided in TS Table 3.3.1-1.

The power uprate will have no impact on the ability of the Low Pressurizer Pressure Trip function. Each accident for which the trip is credited to respond has been analyzed based on a power level of 102 percent RTP, except the steam line break (SLB) event. For the SLB event, hot zero power is bounding. Since this proposed uprate remains bounded by the existing analyses, the Low Pressurizer Pressure Trip function will continue to be adequate.

Pressurizer Water Level - High

The High Pressurizer Water Level Trip is provided as a backup to the High Pressurizer Pressure Trip. The coincidence of two out of the three high pressurizer water level signals trips the reactor. This trip is blocked when three of the four power range channels and both turbine first stage pressure channels read below approximately 10 percent power (P-10 and P-7, respectively).

The power uprate will have no impact on the High Pressurizer Water Level Trip function. This trip is a backup to the High Pressurizer Pressure Trip, and the analysis for the High Pressurizer Pressure Trip bounds the power uprate. The nominal trip setpoint is provided in TS Table 3.3.1-1.

Reactor Coolant Flow Low

The Low Reactor Coolant Flow Trips (e.g., Reactor Coolant Flow - Low (single loop and two loop), RCP Breaker Position (single loop and two loop), Undervoltage Reactor Coolant Pumps, and Underfrequency Reactor Coolant Pumps) protect the core from DNB during a low flow or loss of coolant flow event. These trips are used in the UFSAR Chapter 15 accident analysis, and their applicable nominal trip setpoints are provided in TS Table 3.3.1-1. The means of sensing low flow and a loss of coolant flow accident are as follows.

- Measured low flow or loss of forced coolant flow (actuated by the coincidence of two out of three signals for any reactor coolant loop).

  The loss of flow in any two loops causes a reactor trip above approximately 10 percent (P-7). Above 40 percent power (P-8), the loss of flow in any loop causes a reactor trip.

- Monitored electrical supply to the RCP consisting of: breaker position, undervoltage, and underfrequency (indirectly via RCP breaker trip).

The power uprate will have no impact on the ability of the Low Reactor Coolant Flow Condition Trip function. The events for which the trip are credited to respond have been analyzed based on a power level of 102 percent RTP. Since this uprate remains bounded by the existing analyses, the Low Reactor Coolant Flow Trip function will continue to be adequate.

Steam Generator Level - Low-Low

The Low-Low Steam Generator Level Trip generates a trip signal to protect the SG in the case of a sustained steam/feedwater flow mismatch of insufficient magnitude to cause a flow mismatch reactor trip. The trip is actuated on two out of the three low-low level signals in any SG. This trip is used in the UFSAR Chapter 15 accident analysis, and the nominal trip setpoint is provided in TS Table 3.3.1-1.

The power uprate will have no impact on the ability of the Low–Low Steam Generator Level Trip function. Each accident for which the trip is credited to respond has been analyzed based on a power level of 102 percent of rated thermal power. Since this uprate remains bounded by the existing analyses, the Low–Low Steam Generator Level Trip function will continue to be adequate.

Steam Generator Level - Low, Coincident with Steam Flow/Feedwater Flow Mismatch

The Steam/Feedwater Flow Mismatch Trip protects the reactor from a sudden loss of heat sink. The trip is actuated by a steam/feedwater flow mismatch (one out of two) in coincidence with low water level (one out of two) in any SG.

Feedwater and main steam flows will increase by approximately 1.7 percent as a result of the uprate. It will be necessary to revise calculations for the Steam/Feedwater Flow Mismatch Trip to reflect the new nominal flow rates for feedwater and steam flow. The nominal trip setpoint is provided in TS Table 3.3.1-1. The proposed power uprate will not necessitate hardware changes or require TS changes for this trip function.

Turbine Trip-Low Auto-Stop Oil Pressure

The Low Auto-Stop Oil Pressure Trip function protects the reactor from a sudden loss of its heat sink by anticipating loss of heat removal capabilities of the secondary system following a main turbine trip. The coincidence of two out of the three pressure switches trips the reactor. This trip is blocked when turbine first stage pressure is below approximately 10 percent power (P-7).

The Turbine Trip-Low Auto-Stop Oil Pressure Trip function is not directly credited to mitigate any UFSAR Chapter 15 events, nor is it used to support the containment analysis. Therefore, there is no analysis value. The nominal trip setpoint is provided in TS Table 3.3.1-1. The power uprate will not change the EHC fluid pressure and will have no impact on the Auto-Stop Oil Pressure setpoint.

Turbine Trip-Turbine Stop Valve Closure

The Turbine Stop Valve Closure Trip function protects the reactor from a sudden loss of its heat sink by anticipating loss of heat removal capabilities of the secondary system following a main turbine trip. The coincidence of two out of two turbine stop valve limit switches indicating the stop valves are closed trips the reactor. This trip is blocked when turbine first stage pressure is below approximately 10 percent power (P-7). There is no nominal trip setpoint provided in the TS for this function. Neither turbine stop valve position, nor indication of turbine stop valve position are altered by the power uprate.

Safety Injection Input from Engineered Safety Feature Actuation System

The SI input from ESFAS Trip ensures that if a reactor trip has not already been generated, the ESFAS automatic initiation logic will initiate a reactor trip upon any signal that initiates SI. This trip is based on the safety injection relay contacts being either open or closed. There is no nominal trip setpoint provided in the TS for this function.

Neither the relay position, nor indication of ESFAS relay position, is altered by the power uprate.

Reactor Protection System Permissives

Permissives provide block and unblock signals, as appropriate for plant conditions. No changes are required to the permissives to support the proposed power uprate (i.e., percentage values will not change). However, some instrument rescaling and calibration will be required for the main turbine 1st stage pressure input to the P-7 permissive due to replacement of the high-pressure turbine rotor, and to a lesser degree, higher steam flows resulting from the power uprate. Utilizing the current permissive values will have no impact on the Chapter 15 accident analyses.

3.10.1.2 Engineered Safety Features Actuation System Setpoints

Engineered Safety Features (ESF) and associated support system actuation is provided by the Engineered Safety Features Actuation System (ESFAS) to mitigate the consequences of analyzed events. The impact of the proposed power uprate on these functions and the analyzed events are summarized as follows.

With the exception of the main steam pressure transmitters, which are being replaced to reduce secondary calorimetric measurement uncertainty, the proposed power uprate does not modify or change ESFAS instrumentation or components. The main steam pressure transmitters provide input to the SI initiation ESFAS functions for "Steam Line High Differential Pressure Between Steam Header and Steam Lines" and "High Steam Flow in Two Steam Lines With Steam Line Pressure - Low." The replacement transmitters are more accurate than the current transmitters. Consequently, the uncertainty for these ESFAS functions will decrease and the current Technical Specification Nominal Trip setpoint will not be impacted

The power uprate will necessitate changes to the TS Allowable Value for the "Steam Line High Differential Pressure Between Steam Header and Steam Lines - Safety Injection" actuation function. This function is not utilized to mitigate any UFSAR Chapter 15 events, but is used in the containment analysis. The current TS Allowable Value for this function is ≤ 108.95 psig. The Allowable Value for TS Table 3.3.2-1, Function 1.e, will be revised to reflect an upper bound of ≤ 116.24 psig and add a lower bound of ≥ 83.76 psig.

In addition to the TS change, it will be necessary to update several ESFAS instrument calculations to reflect uprated power conditions. Also, some instrument rescaling and calibration will be required due to replacement of the high-pressure turbine rotor, and to a lesser degree, higher steam flows resulting from the power uprate.

It is concluded that the power uprate will not impact the safety or operational functions of the ESFAS.

Containment High Pressure

The proposed power uprate will not change the analysis values used in the UFSAR Chapter 15 accident analysis, or the analysis values used in the containment analysis. Additionally, the power uprate will not alter the limiting environmental conditions at the pressure transmitters (which are located outside containment), alter environmental conditions for the Hagan electronics, or alter the calibration of the equipment. Therefore, there is no impact to the instrument uncertainty for this function.

Since the analytical values and the uncertainty do not change, neither the TS Nominal Trip Setpoint, nor the TS Allowable Value for the Containment High Pressure Safety Injection function will be impacted as a result of the power uprate. The TS limits on containment pressure will ensure that the Containment High Pressure setpoint is sufficiently above normal containment pressure during operation.

Pressurizer Low Pressure

The proposed power uprate will not change the analysis values used in the UFSAR Chapter 15 accident analysis, or the analysis values used in the containment analysis.

Additionally, the power uprate will not alter the limiting environmental conditions at the pressure transmitters, alter environmental conditions for the Hagan electronics, or alter the calibration of the equipment. Therefore, there is no impact to the instrument uncertainty for this function.

Since the analytical values and the uncertainty do not change, neither the TS Nominal Trip Setpoint, nor the TS Allowable Value for the Pressurizer Low Pressure Safety Injection function will be impacted as a result of the power uprate. Normal pressurizer pressure will not change under power uprate conditions; therefore, the margin between the normal operating point and the trip setpoint will not change.

Steam Line High Differential Pressure Between Steam Header and Steam Lines

This function is not utilized to mitigate any UFSAR Chapter 15 events.

As previously discussed, the power uprate will necessitate changes to the TS Allowable Value for the "Steam Line High Differential Pressure Between Steam Header and Steam Lines - Safety Injection" actuation function. The current TS Allowable Value for this function is ≤ 108.95 psig. Containment response evaluations show that both an overly low value of the actuation setpoint and an overly high value of the actuation setpoint are adverse. The current Allowable Value only provides an upper value for Steam Line High Differential Pressure.

The Allowable Value for TS Table 3.3.2-1, Item 1e, "Steam Line High Differential Pressure Between Steam Header and Steam Lines," is revised to reflect an upper bound of ≤ 116.24 psig, and add a lower bound of ≥ 83.76 psig. These values reflect the reduction in instrument uncertainty associated with replacement of the main steam pressure transmitters to support the proposed power uprate, and the need to provide a lower bound for the "Steam Line High Differential Pressure Between Steam Header and Steam Lines" function. Addition of the lower bound Allowable Value is required due to changes in plant operating characteristics resulting from the power uprate. Since the replacement transmitters are more accurate, the instrument uncertainty for this ESFAS function will decrease and the current Technical Specification Nominal Trip setpoint will not be impacted.

The power uprate will cause the pressure in the main steam lines and in the main steam header to decrease slightly due to the decrease in the steam generator dome pressure. Also, the pressure drop between the steam line pressure transmitter location and the main steam header will increase slightly due to the higher steam flow rates. The current differential pressure between the main steam lines and the main steam header is 20 to 30 psig and will only increase by 1 to 2 psig. Therefore, the differential pressure will remain well below the Steam Line High Differential Pressure setpoint, and sufficient margin will be maintained between the normal operating point and the ESFAS setpoint.

High Steam Flow in Two Steam Lines

The proposed power uprate will not change the analysis values used in the UFSAR Chapter 15 accident analysis.

The current TS Nominal Trip setpoint is a ΔP corresponding to 37.25 percent full steam flow below 20 percent load, and increasing linearly from 37.25 percent steam load at 20 percent power to 109 percent full steam flow at 100 percent load. The TS Allowable Value is a ΔP corresponding to 41.58 percent full steam flow below 20 percent load, and increasing linearly from 41.58 percent full steam flow at 20 percent power to 110.5 percent full steam flow at 100 percent load, and a ΔP corresponding to 110.5 percent full steam flow above 100 percent load.

As a result of the higher steam flows resulting from the power uprate, the High Steam Flow instrument setpoint will be increased. However, the TS Nominal Trip setpoint will not be increased since it is expressed in percent full steam flow. The 100 percent power steam generator pressure used in the uncertainty calculations bounds the power uprate values. Therefore, any impact on instrument uncertainty will be minor.

The UFSAR Chapter 15 analysis value has sufficient margin to accommodate the higher setpoint, and the containment analysis has been updated to reflect the power uprate-related increase in full power steam flow at the High Steam Flow setpoint. Instrument calculation revisions to reflect the new full power steam flow value from the power uprate will also be required.

Additionally, turbine 1st stage pressure will be affected by both the power uprate, and more significantly by replacement of the high-pressure turbine rotor. Turbine 1st stage pressure provides the turbine load input to the High Steam Flow Setpoint equation. Consequently, instrument calculation revision, rescaling of the turbine 1st stage pressure input to the High Steam Flow setpoint to reflect the new pressure verses power relationship, and replacement of some Control Room indicators will be required to support the power uprate. By increasing the High Steam Flow setpoint to match the increase in 100 percent power steam flow the current operating margin will be preserved.

Low T_{avg}

This function is not utilized to mitigate any UFSAR Chapter 15 events, and is not utilized in the containment analysis. Therefore, there is no analysis value.

The power uprate will not impact the hot zero power (HZP) T_{avg}; therefore, the Low T_{avg} Safety Injection signal will not be impacted. The margin between the normal operating point and the ESFAS setpoint will be maintained since the power uprate is not changing the HZP T_{avg} value.

High Steam Flow in Two Steam Lines With Steam Line Pressure - Low

The proposed power uprate will not change the analysis values used in the UFSAR Chapter 15 accident analysis, or the analysis values used in the containment analysis.

The main steam line pressure transmitters are being replaced with more accurate models. As a result, the instrument uncertainty for this ESFAS function will decrease, and neither the TS Nominal Trip setpoint, nor the TS Allowable Value will be impacted by the transmitter replacement or power uprate. Instrument calculation revisions will be required to reflect installation of the new transmitters.

The power uprate will cause the pressure in the main steam lines to decrease slightly due to the decrease in the steam generator dome pressure. However, the pressure will remain well above the Steam Line Low Pressure setpoint and sufficient margin will be maintained between the normal operating point and the ESFAS setpoint.

Containment High-High Pressure

The proposed power uprate will not change the analysis values used in the UFSAR Chapter 15 accident analysis, or the analysis values used in the containment analysis. Additionally, the power uprate will not alter the limiting environmental conditions at the pressure transmitters, which are located outside containment, alter environmental conditions for the electronics, or alter the calibration of the equipment. Therefore, there is no impact on the instrument uncertainty for this function.

Since the analytical values and the uncertainty do not change, neither the TS Nominal Trip Setpoint, nor the TS Allowable Value for Containment High-High Pressure actuation of the Containment Spray, Phase B Isolation, or Steam Line Isolation functions will change as a result of the power uprate. The TS limits on containment pressure will ensure that the Containment High-High Pressure setpoint is sufficiently above the normal containment pressure during operation.

Pressurizer Low Pressure Interlock

This function is not utilized to mitigate any Chapter 15 events, and is not utilized in the containment analysis. Therefore, there is no analysis value.

The Pressurizer Low Pressure Interlock allows the Pressurizer Low Pressure Safety Injection feature to be manually blocked in order to cooldown and depressurize the plant. The power uprate will not alter pressurizer temperature, pressure, or level. Consequently, neither the TS Nominal Trip Setpoint, nor the TS Allowable Value for the Pressurizer Low Pressure Interlock function will need to be changed. The margin between the normal operating point and the interlock setpoint will not change due to the power uprate since the normal pressurizer pressure will not change as a result of the power uprate.

Low T_{avg} Interlock

This function is not utilized to mitigate any Chapter 15 events, and is not utilized in the containment analysis. Therefore, there is no analysis value.

The Low T_{avg} Interlock allows the Low T_{avg} Safety Injection signal to be manually blocked in order to cooldown the plant. The power uprate will not impact the HZP T_{avg}; therefore, the Low T_{avg} Interlock signal will not be impacted. Consequently, neither the TS Nominal Trip Setpoint, nor the TS Allowable Value for the Low T_{avg} Interlock signal will need to be changed. The margin between the normal operating point and the interlock setpoint is maintained, since the power uprate is not changing the HZP T_{avg} value.

3.10.2 Emergency Core Cooling System Performance

The HBRSEP, Unit No. 2, Emergency Core Cooling (ECCS) performance analysis consists of three analyses: loss-of-coolant accidents (e.g., Large Break Loss-of-Coolant (LBLOCA) and Small Break Loss-of-Coolant (SBLOCA)), MSLB, and post-LOCA Long Term Cooling (LTC). With the exception of the MSLB, these analyses were performed at 2346 MWt, or 102 percent power. The MSLB analysis was performed for the most limiting case, at HZP.

As allowed by the recent revision to 10 CFR 50, Appendix K, paragraph 1.A, this license amendment request proposes to increase licensed core power by approximately 1.7 percent to 2339 MWt, by reducing the power measurement uncertainty to less than or equal to 0.3 percent. As a result of this reduction in power measurement uncertainty, the value of the proposed licensed core power level plus the maximum power measurement uncertainty is unchanged at 2346 MWt. The proposed values for licensed core power level and power measurement uncertainty comply with the revised requirements of 10 CFR 50, Appendix K, paragraph 1.A.

As a consequence of the proposed power uprate, reactor vessel average temperature is slightly higher than the nominal value used in the current LBLOCA analysis (0.5 °F). A nominal value is used in the LBLOCA analysis since reactor vessel average temperature is not a significant parameter for this accident. The change in reactor vessel average temperature has a negligible effect on peak cladding temperature. The reactor vessel average temperature used in the SBLOCA analysis will remain bounding under power uprate conditions.

The effect of the power uprate on the Hot Full Power (HFP) MSLB was evaluated and it was concluded that the HFP event would not become more limiting than the HZP event. Therefore, there will be no impact on the ability the ECCS to mitigate the thermal-hydraulic consequences of a MSLB due to the power uprate, and the ECCS will continue to be capable of mitigating a MSLB.

Since there is no change to the power level used in the ECCS performance analysis, or significant changes to other inputs to the analysis as a result of the proposed power uprate, there are no significant changes to the HBRSEP, Unit No. 2, ECCS performance analysis.

3.10.3 Updated Final Safety Analysis Report Transients and Accidents

This section summarizes the dispositions for the LOCA and non-LOCA events evaluated in Chapter 15 of the HBRSEP, Unit No. 2, UFSAR, and their supporting analyses and calculations. Changes in plant parameters that result from an increase in core power of up to 1.7 percent were systematically reviewed relative to the Cycle 21 analyses of record. The Cycle 21 analyses of record bound the power uprate, except for the 10 CFR 50.46 analysis of the LBLOCA (UFSAR 15.6.5). The effect of the power uprate on the LBLOCA are judged to be negligible as described herein.

HBRSEP, Unit No. 2, has reanalyzed the dose consequences associated with the Fuel Handling Accident (FHA), LBLOCA, MSLB, Locked Rotor, Steam Generator Tube Rupture (SGTR), and Single Rod Control Cluster Assembly (RCCA) Withdrawal. The FHA analysis was submitted to the NRC for review and approval by letter (RNP-RA/02-0027) dated March 13, 2002. The remainder of these dose consequence analyses were submitted to the NRC for review and approval by letter (RNP-RA/02-0067) dated May 10, 2002.

Plant Parameter Changes

The disposition of events was based on the following plant parameters for the power uprate:

- The uprated nominal core power is 2339 MWt and the vessel average temperature is 575.9°F.

- The power measurement uncertainty after the power uprate is 0.3 percent, such that the total core thermal power, including the uncertainty, is no greater than 2346 MWt.

- The analysis value for the Power Range High Flux Trip (high setting) setpoint is 116 percent of 2339 MWt. Also, the analysis value for the Power Range High Flux Trip (low setting) setpoint is 34.4 percent.

- The RCS total flow rate is unchanged from the current Technical Specification minimum value of 97.3×10^6 lbm/hr.

- The maximum analysis value for main feedwater flow is increased from 4.2×10^6 lbm/hr, to a bounding value of 4.8×10^6 lbm/hr for Chapter 15 analyses, and 4.3×10^6 for UFSAR Chapter 6 containment evaluations.

- The reference T_{avg} in the Overtemperature ΔT (OTDT) and Overpower ΔT (OPDT) Trips will change to be consistent with the vessel T_{avg} under power uprate conditions.

- The HZP temperature is unchanged from the current value of 547°F.

The following table compares the UFSAR Chapter 15 safety analysis boundary conditions for the current and uprated power levels:

	Current	**1.7 Percent Power Uprate**	**Difference**
Uncertainty Adjusted Thermal Power	2346 MWt	2346 MWt	+0 MWt
Vessel Average Temperature	575.4°F	575.9°F	+0.5°F
Minimum RCS Flow Rate (Technical Specification)	97.3×10^6 lbm/hr	97.3×10^6 lbm/hr	No change

With respect to events that challenge the departure from nucleate boiling (DNB) acceptance criterion, the primary effect of the power uprate will be a slight increase in the vessel average temperature. Based on sensitivities for the high thermal performance (HTP) DNB correlation, it is estimated that the effect of this change will be a decrease in DNB ratio (DNBR) of about 1 to 2 percent. In general, the additional margin afforded by application of the approved statistical DNBR methodology is much larger than the estimated DNBR decrease for the power uprate.

The changes to thermal-hydraulic properties due to the change in vessel temperature will have a negligible effect on the timing of reactor trips, including the OTDT, OPDT, and high flux trips.

The fuel centerline melt limit and the peak linear heat generation rate (LHGR) analyses will not be significantly impacted by the power uprate.

3.10.3.1 Increase In Heat Removal by the Secondary Side

<u>Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature</u> (UFSAR 15.1.1)

The Excess Load event (UFSAR 15.1.3) bounds the consequences of the decrease in feedwater temperature event. The NRC found this assessment to be acceptable in a Safety Evaluation associated with Technical Specifications (TS) Amendment No. 87, dated November 7, 1984. None of the changes associated with the proposed power uprate will impact the relative severity of this event.

<u>Feedwater System Malfunctions that Result in an Increase in Feedwater Flow</u> (UFSAR 15.1.2)

For the inadvertent opening of one feedwater regulation valve at power, the consequences are bounded by the Excess Load event (UFSAR 15.1.3). For the inadvertent opening of one feedwater regulation valve at startup, the consequences are bounded by the Uncontrolled

Control Bank Withdrawal from a Subcritical or Low Power Condition (UFSAR 15.4.1). The NRC found the assessment that the Excess Load event bounded the overcooling response of the decrease in feedwater temperature event to be acceptable in a Safety Evaluation associated with TS Amendment No. 87, dated November 7, 1984. In this same Safety Evaluation, the NRC found the assessment that the rod withdrawal event bounded the reactivity insertion response of the decrease in feedwater temperature event to be acceptable.

For the hot full power (HFP) initiated events, the heat load resulting from the increase in feedwater regulation valve flow to the analysis value of 4.8×10^6 lbm/hr continues to be bounded by the Excess Load event. Likewise, the change in feedwater regulation valve flow rate will not invalidate the disposition for an event initiated from HZP. Therefore, the Uncontrolled Control Bank Withdrawal from a Subcritical or Low Power Condition initiated at beginning-of-cycle (BOC) bounds an increase in feedwater flow initiated at HZP and end-of-cycle (EOC).

Increase in Steam Flow (Excess Load)
(UFSAR 15.1.3)

This event is analyzed to assess the challenge to the DNBR criterion. The analysis of record modeled a core power level of 2346 MWt.

The methodology used in the analysis of record to assess the challenge to the DNBR criterion and thermal-hydraulic effects was ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, May 1992. This methodology was approved for use at HBRSEP, Unit No. 2, by the NRC in a Safety Evaluation associated with Technical Specifications (TS) Amendment No. 154, dated December 12, 1994.

Thus, the analysis of record is bounding for the power uprate.

Inadvertent Opening of a Steam Generator Relief or Power Operated Relief Valve
(UFSAR 15.1.4)

The Inadvertent Opening of a Steam Generator Relief or Safety Relief Valve event is bounded by the Excess Load event (UFSAR 15.1.3) at power. After reactor trip, it is bounded by the MSLB event (UFSAR 15.1.5). The NRC found this assessment to be acceptable in a Safety Evaluation associated with TS Amendment No. 87, dated November 7, 1984. Changes associated with the proposed power uprate will not impact the relative severity of this event.

Main Steam Line Break
(UFSAR 15.1.5)

This event is analyzed to assess the challenge to the radiological dose criterion resulting from fuel failure due to penetration of DNBR and/or fuel centerline melt limits. The methodology used in the analysis of record to determine the extent of fuel failure due to penetration of DNBR and/or fuel centerline melt limits for use in the radiological dose consequence analysis was EMF-84-093(P)(A), Revision 1, "Steam Line Break Methodology for PWRs," Siemens Power Corporation, February 1999. This methodology was approved for use at HBRSEP, Unit No. 2, by the NRC in a Safety Evaluation associated with TS Amendment No. 188, dated August 3, 2000. The limiting minimum departure from nucleate boiling (MDNBR) and peak LHGR cases in the analysis of record are initiated from HZP. None of the changes proposed for the power uprate will affect HZP conditions; thus, the limiting HZP cases are not impacted by the power uprate.

The analyses of record HFP cases were initialized with the following conditions: core thermal power of 2346 MWt; mass flow rate of 97.3×10^6 lbm/hr; and, a vessel average temperature of 577.5°F. The moderator feedback and the worth of the assumed stuck rod primarily drive this event. Moderator feedback is a function of the Core Operating Limits Report (COLR) moderator temperature coefficient (MTC) and the initial RCS average temperature. The COLR MTC is not changed by the proposed power uprate changes. Comparing the average temperatures from the analysis and that for the power uprate (i.e., 575.9°F) indicates that the value used in the analysis of record is conservatively higher (i.e., yielding more moderator feedback). The stuck rod worth will only be minimally affected by the power uprate. The Doppler feedback from the uprate power level compared to the analysis of record will not significantly affect this event. The HFP cases in the analysis of record will not become limiting as a result of the power uprate.

Changes to the MFRV trim will not be significant since main feedwater is isolated shortly after event initiation. Also, the analysis of record conservatively assumes that three times the initial main feedwater flow is directed to the faulted steam generator until isolation occurs. This flow rate significantly exceeds the maximum capacity of the feedwater regulation valves. Thus, the analysis of record is not impacted by the changes in the Main Feedwater System.

The radiological consequences of this event have been reanalyzed utilizing an Alternative Source Term (AST) in accordance with 10 CFR 50.67. The analysis was performed in accordance with Regulatory Guide (RG) 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," July 2000. This analysis assumed a core thermal power level of 2346 MWt and bounds the proposed power uprate. The analysis was submitted for NRC review by letter (RNP-RA/02-0067) dated May 10, 2002. Following approval of the AST submittal, the changes associated with the proposed power uprate will have no impact on the MSLB accident radiological dose consequences.

3.10.3.2 Decrease In Heat Removal by the Secondary Side

Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow
(UFSAR 15.2.1)

The plant has no main steam pressure regulator whose failure or malfunction could cause a steam flow transient. Therefore, this event is not applicable.

Loss of External Electric Load
(UFSAR 15.2.2)

This event is analyzed to evaluate the challenge to peak primary and secondary side pressure and DNBR criteria.

The methodology used in the analysis of record to assess the challenge to peak primary and secondary pressure and the DNBR criterion was ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, May 1992. This methodology was approved for use at HBRSEP, Unit No. 2, by the NRC in a Safety Evaluation associated with Technical Specifications (TS) Amendment No. 154, dated December 12, 1994.

The small increase in vessel average temperature will insignificantly impact the thermal properties of the primary and secondary sides. The timing of reactor trips will be negligibly affected by the power uprate. Thus, the peak primary and secondary side pressures are not adversely impacted by the power uprate conditions and the analysis of record remains bounding.

Relative to DNBR, the small increase in vessel average temperature will insignificantly impact the thermal properties of the primary and secondary sides. Thus, the DNBR analysis of record is bounding for the power uprate.

The evaluation of this event considered changes to the MFRV trim resulting from the power uprate. The MFRV trim modification will not adversely impact the UFSAR Chapter 15 analysis of this event.

Turbine Trip
(UFSAR 15.2.3)

The analysis of record for the Loss of External Load Event (15.2.2) was evaluated in a manner that bounds this event. The NRC found this assessment to be acceptable in a Safety Evaluation associated with TS Amendment No. 87, dated November 7, 1984.

Loss of Condenser Vacuum and Other Events Resulting in Turbine Trip
(UFSAR 15.2.4)

This event is bounded by the Loss of External Load event (UFSAR 15.2.2). The NRC found this assessment to be acceptable in a Safety Evaluation associated with TS Amendment No. 87, dated November 7, 1984. The power uprate changes will not impact the relative severity of this event.

Inadvertent Closure of Main Steam Isolation Valves
(UFSAR 15.2.5)

Inadvertent Closure of the Main Steam Isolation Valves is bounded by the Loss of External Load event (15.2.2). The NRC found this assessment to be acceptable in a Safety Evaluation associated with TS Amendment No. 87, dated November 7, 1984. Changes associated with the proposed power uprate will not impact the relative severity of this event.

Loss of Non-Emergency AC Power to the Station Auxiliaries
(UFSAR 15.2.6)

This event is comprised of two situations, the first situation involves a loss of external loads with continued availability of station auxiliaries (i.e., primary coolant pumps and MFW pumps), and the second involves a turbine trip with coincident loss of the primary coolant pumps and FW pumps.

For the case where external electrical loads are lost, but station auxiliaries are maintained, this event is bounded by the Loss of External Load event (UFSAR 15.2.2). The NRC found this assessment to be acceptable in a Safety Evaluation associated with TS Amendment 87, dated November 7, 1984.

For the case where the turbine trips resulting in loss of primary coolant and MFW pumps, this event is bounded in the short-term by the Loss of Forced Reactor Coolant Flow event (UFSAR 15.3.1). In the long-term, this event is bounded by the Loss of Normal Feedwater (UFSAR 15.2.7).

This assessment was provided to the NRC in report XN-NF-83-72, Revision 2, Supplement 1, "H. B. Robinson Unit 2 Cycle 10 Safety Analysis Report, Revision 2 Disposition of Chapter 15 Events," Exxon Nuclear Corporation, July 1984. This report was reviewed by the NRC during preparation of the Safety Evaluation associated with TS Amendment No. 87, dated November 7, 1984. Changes associated with the proposed power uprate will not impact the relative severity of this event.

Loss of Normal Feedwater
(UFSAR 15.2.7)

This event is analyzed to assess the challenge to the RCS pressure criterion. The analysis of record modeled a core power level of 2346 MWt.

The methodology used in the analysis of record to assess the challenge to the RCS pressure criterion was ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, May 1992. This methodology was approved for use at HBRSEP, Unit No. 2, by the NRC in a Safety Evaluation associated with TS Amendment No. 154, dated December 12, 1994.

Decay heat and the capacities of the AFW System primarily drive this event. The small increase in vessel average temperature will insignificantly impact the thermal properties of the primary and secondary sides. Further, the timing of reactor trip will not be impacted. The progression of this event and the capacity of the AFW System to remove decay heat will be negligibly impacted by the power uprate. The changes in the main feedwater valves will not adversely impact this event. Thus, the analysis of record bounds the power uprate.

Feedwater System Pipe Break
(UFSAR 15.2.8)

This event is bounded by the MSLB (UFSAR 15.1.5). The NRC found this assessment to be acceptable in a Safety Evaluation associated with TS Amendment No. 87, dated November 7, 1984. Changes resulting from the power uprate will not impact the relative severity of this event.

3.10.3.3 Decrease In Reactor Coolant System Flow Rate

Loss of Forced Reactor Coolant Flow
(UFSAR 15.3.1)

This event is analyzed to assess the challenge to the DNBR criterion. The analysis of record modeled a core power level of 2346 MWt.

The methodology used in the analysis of record to assess the challenge to the DNBR criterion was ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, May 1992. This methodology was approved for use at HBRSEP, Unit No. 2, by the NRC in a Safety Evaluation associated with TS Amendment No. 154, dated December 12, 1994.

The small increase in vessel average temperature will insignificantly impact the thermal properties of the primary and secondary sides. Further, the timing of reactor trip is not impacted. Thus, the analysis of record is bounding for the power uprate.

The evaluation of this event considered changes to the MFRV trim resulting from the power uprate. The MFRV trim modification will not adversely impact the UFSAR Chapter 15 analysis of this event.

Reactor Coolant Pump Shaft Seizure (Locked Rotor)
(UFSAR 15.3.2)

This event is analyzed to assess the challenge to the radiological dose criterion resulting from fuel failure due to penetration of the DNBR limit. The analysis of record modeled a core power level of 2346 MWt.

The methodology used in the analysis of record to determine the extent of fuel failure due to penetration of the DNBR limit for use in the radiological dose consequence analysis was ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, May 1992. This methodology was approved for use at HBRSEP, Unit No. 2, by the NRC in a Safety Evaluation associated with TS Amendment No. 154, dated December 12, 1994.

For Cycle 21, eight fuel assemblies are predicted to fail as a result of this event. Since all fuel rods within the eight affected assemblies are conservatively assumed to fail, the small changes in core operating conditions will not impact the fuel failure estimate. Thus, the analysis of record is bounding for the power uprate.

The evaluation of this event considered changes to the MFRV trim resulting from the power uprate. The MFRV trim modification will not adversely impact the UFSAR Chapter 15 analysis of this event.

The radiological consequences of this event have been reanalyzed utilizing an Alternative Source Term (AST) in accordance with 10 CFR 50.67. The analysis was performed in accordance with Regulatory Guide (RG) 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," July 2000. This analysis assumed a core thermal power level of 2346 MWt and bounds the proposed power uprate. The analysis was submitted for NRC review by letter (RNP-RA/02-0067) dated May 10, 2002. Following approval of the AST submittal, the changes associated with the proposed power uprate will have no impact on the Locked Rotor event radiological dose consequences.

Reactor Coolant Pump Shaft Break
(UFSAR 15.3.3)

This event is bounded by the Locked Rotor event (UFSAR 15.3.2). The NRC found this assessment to be acceptable in a Safety Evaluation associated with TS Amendment No. 87, dated November 7, 1984. Changes resulting from the power uprate will not impact the relative severity of this event.

3.10.3.4 Reactivity and Power Distribution Anomalies

Uncontrolled Rod Cluster Control Assembly (RCCA) Bank Withdrawal from Subcritical or Low Power
(UFSAR 15.4.1)

This event is analyzed to assess the challenge to the DNBR and fuel centerline melt criteria. The methodology used in the analysis of record to assess the challenge to the DNBR and fuel centerline melt criteria was ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, May 1992. This methodology was approved for use at HBRSEP, Unit No. 2, by the NRC in a Safety Evaluation associated with TS Amendment No. 154, dated December 12, 1994.

The HZP temperature is unchanged and the analytical value Power Range High Neutron Flux Trip (low setting) setpoint is proportionally reduced. The progression of the transient is not impacted by the proposed changes for the power uprate. The MDNBR and fuel centerline temperature analyses will not be impacted.

The evaluation of this event considered changes to the MFRV trim resulting from the power uprate. The MFRV trim modification will not adversely impact the UFSAR Chapter 15 analysis of this event.

Uncontrolled Control Rod Assembly Bank Withdrawal at Power
(UFSAR 15.4.2)

This event is analyzed to assess the challenge to the DNBR criterion. The analysis of record modeled a core power level of 2346 MWt.

The methodology used in the analysis of record to assess the challenge to the DNBR criterion was ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, May 1992. This methodology was approved for use at HBRSEP, Unit No. 2, by the NRC in a Safety Evaluation associated with TS Amendment No. 154, dated December 12, 1994.

The power uprate will have a negligible effect on the timing of reactor trips. Thus, the analysis of record is bounding for the power uprate.

The evaluation of this event considered changes to the MFRV trim resulting from the power uprate. The MFRV trim modification will not adversely impact the UFSAR Chapter 15 analysis of this event.

Withdrawal of Single Full Length RCCA
(UFSAR 15.4.3.1)

This event is analyzed to assess the challenge to the radiological dose criterion resulting from fuel failure due to penetration of DNBR and/or fuel centerline melt limits. The analysis of record modeled a core power level of 2346 MWt.

The methodology used in the analysis of record to determine the extent of fuel failure due to penetration of DNBR and/or fuel centerline melt limits for use in the radiological dose consequence analysis was ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, May 1992. This methodology was approved for use at HBRSEP, Unit No. 2, by the NRC in a Safety Evaluation associated with TS Amendment No. 154, dated December 12, 1994.

For Cycle 21, three fuel assemblies are predicted to fail as a result of this event. Since all fuel rods within the three affected assemblies are conservatively assumed to fail, the small changes in core operating conditions for the power uprate will not impact the fuel failure estimate. Thus, the analysis of record is bounding for the power uprate.

The evaluation of this event considered changes to the MFRV trim resulting from the power uprate. The MFRV trim modification will not adversely impact the UFSAR Chapter 15 analysis of this event.

The radiological consequences of this event have been reanalyzed utilizing an Alternative Source Term (AST) in accordance with 10 CFR 50.67. The analysis was performed in accordance with Regulatory Guide (RG) 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," July 2000. This analysis assumed a core thermal power level of 2346 MWt and bounds the proposed power uprate. The analysis was submitted for NRC review by letter (RNP-RA/02-0067) dated May 10, 2002. Following approval of the AST submittal, the changes associated with the proposed power uprate will have no impact on the Withdrawal of a Single Full Length RCCA event radiological dose consequences.

Static Misalignment of a Single RCCA
(UFSAR 15.4.3.2)

This event is analyzed to assess the challenge to the DNBR and fuel centerline melt criteria. The analysis of record modeled a core power level of 2346 MWt.

The methodology used in the analysis of record to assess the challenge to the DNBR and fuel centerline melt criteria was ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, May 1992. This methodology was approved for use at HBRSEP, Unit No. 2, by the NRC in a Safety Evaluation associated with TS

Amendment No. 154, dated December 12, 1994. Thus, the analysis of record is bounding for the power uprate.

The evaluation of this event considered changes to the MFRV trim resulting from the power uprate. The MFRV trim modification will not adversely impact the UFSAR Chapter 15 analysis of this event.

Dropped RCCA and RCCA Bank
(UFSAR 15.4.3.3)

This event is analyzed to assess the challenge to the DNBR and fuel centerline melt criteria. The analysis of record modeled a core power level of 2346 MWt.

The methodology used in the analysis of record to assess the challenge to the DNBR and fuel centerline melt criteria was ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, May 1992. This methodology was approved for use at HBRSEP, Unit No. 2, by the NRC in a Safety Evaluation associated with TS Amendment No. 154, dated December 12, 1994. Thus, the analysis of record is bounding for the power uprate.

The evaluation of this event considered changes to the MFRV trim resulting from the power uprate. The MFRV trim modification will not adversely impact the UFSAR Chapter 15 analysis of this event.

Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature
(UFSAR 15.4.4)

The TS do not permit operation with less than three primary coolant pumps during power operation. Therefore, analysis of this event is unnecessary.

Recirculation Loop at Incorrect Temperature or Flow Controller Malfunction
(UFSAR 15.4.5)

The plant has neither primary loop isolation valves nor means to control primary flow. Therefore, this event is not applicable.

Chemical Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant
(UFSAR 15.4.6)

This event is analyzed to assess the challenge to the time-to-criticality criteria. During power operation, the consequences of this event are bounded by those of the rod withdrawal events (UFSAR 15.4.1 and 15.4.2). During operation in other modes, the operator has sufficient time to respond to and mitigate the event. The NRC found this assessment to be acceptable in a Safety Evaluation associated with TS Amendment No. 87, dated

November 7, 1984. Since the challenge to acceptance criteria is most significant for modes other than Mode 1, the changes in core operating parameters will not impact this event.

Inadvertent Loading and Operation of a Fuel Assembly into the Improper Location (UFSAR 15.4.7)

This event is analyzed to assess the challenge to the radiological dose criterion resulting from fuel failure due to penetration of DNBR and/or fuel centerline melt limits. The analysis of record modeled a core power level of 2346 MWt.

The methodology used in the analysis of record to determine the extent of fuel failure due to penetration of DNBR and/or fuel centerline melt limits for use in the radiological dose consequence analysis was XN-NF-84-73(A), Revision 5, "Exxon Nuclear Methodology for PWRs: Analysis of Chapter 15 Events," Exxon Nuclear Corporation, October 1990. Approval of this methodology for use at HBRSEP, Unit No. 2, was acknowledged by the NRC in a Safety Evaluation associated with TS Amendment No. 141, dated July 15, 1992. Thus, the analysis of record is bounding for the power uprate.

The evaluation of this event considered changes to the MFRV trim resulting from the power uprate. The MFRV trim modification will not adversely impact the UFSAR Chapter 15 analysis of this event.

Spectrum of Rod Cluster Control Assembly Ejection Accidents (UFSAR 15.4.8)

This event is analyzed to assess the challenge to the radiological dose criterion resulting from fuel failure due to penetration of DNBR and/or fuel centerline melt limits. The analysis of record modeled a core power level of 2346 MWt.

The methodologies used in the analysis of record to determine the extent of fuel failure due to penetration of DNBR and/or fuel centerline melt limits for use in the radiological dose consequence analysis are as follows:

- ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, May 1992. This methodology was approved for use at HBRSEP, Unit No. 2, by the NRC in a Safety Evaluation associated with TS Amendment No. 154, dated December 12, 1994.

- XN-NF-78-44(NP)(A), "A Generic Analysis of the Control Rod Ejection Transient for PWRs," Exxon Nuclear Corporation, February 1979. Approval of this methodology for use at HBRSEP, Unit No. 2, was acknowledged by the NRC in a Safety Evaluation associated with TS Amendment No. 141, dated July 15, 1992.

The power uprate will have a negligible effect on the timing of reactor trips. Thus, the analysis of record is bounding for the power uprate.

The evaluation of this event considered changes to the MFRV trim resulting from the power uprate. The MFRV trim modification will not adversely impact the UFSAR Chapter 15 analysis of this event.

Spectrum of Rod Drop Accidents
(UFSAR 15.4.9)

This event is not applicable to pressurized water reactors such as HBRSEP, Unit No. 2.

3.10.3.5 Increase In Reactor Coolant System Inventory

Inadvertent Operation of Emergency Core Cooling System
(UFSAR 15.5.1)

This event is not applicable for HBRSEP, Unit No. 2, because the shutoff head of the high pressure safety injection pumps is less than the reactor trip setpoint pressure, and therefore, cannot increase the primary inventory during power operation.

CVCS Malfunction that Increases Reactor Coolant Inventory
(UFSAR 15.5.2)

For refueling and startup, this event is bounded by the Decrease in Boron Concentration event (UFSAR 15.4.6). For at-power conditions, it is bounded by the Uncontrolled RCCA Bank Withdrawal at Power event (UFSAR 15.4.2). The pressurizer PORVs mitigate the pressurization. The NRC found this assessment to be acceptable in a Safety Evaluation associated with TS Amendment No. 87, dated November 7, 1984. Changes resulting form the power uprate will not impact the relative severity of this event.

3.10.3.6 Decreases In Reactor Coolant Inventory (UFSAR 15.6)

Inadvertent Opening of a Pressurizer Safety or Power Operated Relief Valve
(UFSAR 15.6.1)

This event is analyzed to assess the challenge to the DNBR criterion. The analysis of record modeled a core power level of 2346 MWt.

The methodology used in the analysis of record to assess the challenge to the DNBR criterion was ANF-89-151(P)(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," Advanced Nuclear Fuels Corporation, May 1992. This methodology was approved for use at HBRSEP, Unit No. 2, by the NRC in a Safety Evaluation associated with TS Amendment No. 154, dated December 12, 1994. Thus, the analysis of record is bounding for the power uprate.

The evaluation of this event considered changes to the MFRV trim resulting from the power uprate. The MFRV trim modification will not adversely impact the UFSAR Chapter 15 analysis of this event.

Small Break Loss-of-Coolant Accidents
(UFSAR 15.6.2)

This event is analyzed to assess the challenge to the 10 CFR 50.46 criteria, particularly peak cladding temperature and oxidation limits. The analysis of record was initialized with the following conditions: core thermal power of 2346 MWt; mass flow rate of $100.3x10^6$ lbm/hr; and vessel average temperature of 578.1°F. The analysis was performed using the methodologies contained in XN-NF-82-49(P)(A), Rev. 1, "Exxon Nuclear Company Evaluation Model - EXEM PWR Small Break Model," Siemens Power Corporation, April, 1989; and XN-NF-82-49(P)(A), Rev. 1, Supplement 1, "Exxon Nuclear Company Evaluation Model - Revised EXEM PWR Small Break Model, Siemens Power Corporation, Dec. 1994. These methodologies were acknowledged by the NRC as approved for use at HBRSEP, Unit No. 2, in the Safety Evaluations associated with TS Amendment 141, dated July 15, 1992; and TS Amendment 154, dated December 12, 1994.

The initial RCS average temperature used in the analysis is over 2°F higher than the value for the power uprate. A higher initial RCS temperature is slightly more limiting due to a higher saturation pressure, which limits the injection of high pressure safety injection flow. Since the conditions used in the analysis of record bound the conditions of the power uprate, the existing analysis bounds the power uprate.

The evaluation of this event considered changes to the MFRV trim resulting from the power uprate. The MFRV trim modification will not adversely impact the UFSAR Chapter 15 analysis of this event.

Steam Generator Tube Rupture (SGTR)
(UFSAR 15.6.3)

The Inadvertent Opening of a Pressurizer Safety or Power Operated Relief Valve event (UFSAR 15.6.1) bounds the non-radiological consequences of this event. The NRC found this analysis of the SGTR event to be acceptable in a Safety Evaluation associated with TS Amendment No. 87, dated November 7, 1984. Changes resulting from the power uprate will not impact the relative severity of the non-radiological consequences of this event.

The radiological consequences of this event have been reanalyzed utilizing an Alternative Source Term (AST) in accordance with 10 CFR 50.67. The analysis was performed in accordance with Regulatory Guide (RG) 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," July 2000. This analysis assumed a core thermal power level of 2346 MWt and bounds the proposed power uprate. The analysis was submitted for NRC review by letter (RNP-RA/02-0067) dated May 10, 2002. Following approval of the AST submittal, the changes associated with the proposed power uprate will have no impact on the SGTR analysis.

Spectrum of Boiling Water Reactor (BWR) Steam Piping Failures Outside Containment (UFSAR 15.6.4)

This event is not applicable to HBRSEP, Unit No. 2.

Loss-of-Coolant Accidents (UFSAR 15.6.5)

This Loss-of-Coolant event is analyzed to assess the challenge to the 10 CFR 50.46 criteria, particularly peak cladding temperature and oxidation limits. The current 10 CFR 50, Appendix K analysis was performed assuming a core power level of 2346 MWt, a vessel average temperature of 575.4°F, and an RCS mass flow rate of 97.3×10^6 lbm/hr. The LBLOCAs are analyzed using the methodologies contained in "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," EMF-2087(P)(A), Siemens Power Corporation, June 1999; and EMF-2286(P), "H. B. Robinson Unit 2 Extended Transfer to Cold Leg Recirculation Following a LBLOCA," Siemens Power Corporation, November 2000.

With respect to the power uprate conditions, the vessel average temperature is slightly higher (0.5°F) than for the current analysis. The methodology uses a nominal vessel average temperature since it is not a significant parameter for LBLOCA. The 0.5°F increase in vessel average temperature affects peak cladding temperature negligibly. The analysis of record thus remains applicable for the power uprate. Additionally, the switchover to recirculation analysis remains bounding following the power uprate.

The evaluation of this event considered changes to the MFRV trim resulting from the power uprate. The MFRV trim modification will not adversely impact the UFSAR Chapter 15 analysis of this event.

The radiological consequences of this event have been reanalyzed utilizing an Alternative Source Term (AST) in accordance with 10 CFR 50.67. The analysis was performed in accordance with Regulatory Guide (RG) 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," July 2000. This analysis assumed a core thermal power level of 2346 MWt and bounds the proposed power uprate. The analysis was submitted for NRC review by letter (RNP-RA/02-0067) dated May 10, 2002. Following approval of the AST submittal, the changes associated with the proposed power uprate will have no impact on the LOCA radiological dose consequences.

3.10.3.7 Radioactive Releases From A Subsystem Or Component

Radioactive Waste Gas System Leak or Failure
(UFSAR 15.7.1)

The radiological consequences of this event have been reanalyzed utilizing an Alternative Source Term (AST) in accordance with 10 CFR 50.67. The analysis was performed in accordance with Regulatory Guide (RG) 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," July 2000. This analysis assumed a core thermal power level of 2346 MWt and bounds the proposed power uprate. The analysis was submitted for NRC review by letter (RNP-RA/02-0067) dated May 10, 2002. Following approval of the AST submittal, the changes associated with the proposed power uprate will have no impact on the radiological dose consequences of this event.

Liquid Waste System Leak or Failure
(UFSAR 15.7.2)

UFSAR 15.7.2 states that the administrative controls imposed, combined with the safety features built into the equipment, provide a high degree of assurance against an accidental release of waste liquids. The proposed power uprate will not alter the administrative controls or the design or configuration of the Liquid Waste System; therefore, the conclusion of UFSAR 15.7.2 is not impacted.

Postulated Radioactivity Release Due to Liquid Tank Failure
(UFSAR 15.7.3)

UFSAR 15.7.3 references the Diffusion of Short-Term Releases evaluation in UFSAR 2.4.6. The maximum permissible releases of radioactive materials are evaluated in UFSAR 2.4.6.3 based on both individual nuclides and a "typical mixture." Review of the original FSAR shows that the "typical mixture" was based on the Reactor Coolant System Equilibrium Activities listed in UFSAR Table 11.1.1-2, corrected for decay during processing. Compliance with the current RCS specific activity limits of TS 3.4.16 will ensure that the RCS Equilibrium Activities listed in UFSAR Table 11.1.1-2 will remain bounding under power uprate conditions. Additionally, compliance will also continue to be assured under the more restrictive TS 3.4.16 RCS specific activity limits proposed in the AST submittal letter (RNP-RA/02-0067) dated May 10, 2002. Therefore, this event is not impacted by the changes associated with the proposed power uprate.

Design Basis Fuel Handling Accidents
(UFSAR 15.7.4)

The radiological consequences of this event have been reanalyzed utilizing an AST in accordance with 10 CFR 50.67. The analysis was performed in accordance with Regulatory Guide (RG) 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," July 2000. This analysis assumed a core thermal power level of 2346 MWt and bounds the proposed power uprate. The analysis was submitted for NRC review by letter dated March 13, 2002 (RNP-RA/02-0027). Following approval of the AST submittal, the changes associated with the proposed power uprate will have no impact on the FHA radiological dose consequences.

Spent Fuel Cask Drop Accidents
(UFSAR 15.7.5)

The radiological consequences of the Spent Fuel Cask Drop Accident are based on the fuel assembly burnup, enrichment, and cooling time limits established by the Certificate of Compliance for the cask. The proposed power uprate does not alter the cask license limits; therefore, this event is not impacted by the power uprate.

Spent Fuel Pit Water Loss
(UFSAR 15.7.6)

The proposed power uprate will not alter the configuration of the Spent Fuel Pool, and therefore will not alter the conclusions of the UFSAR 15.7.6 evaluation of Spent Fuel Pool water loss.

Control Room Habitability
(UFSAR 6.4.4.2 and 15.6.5)

The effect of the power uprate on Control Room Habitability due to non-radiological threats (i.e., toxic chemical releases, volatile chemicals) has been evaluated. The proposed power uprate will have no impact on the sources of these threats or the ability to detect and respond to indications of smoke or toxic/noxious chemicals.

The ability of the Control Room Habitability system to mitigate the radiological consequences of various events was reanalyzed during AST development. This reanalysis assumed a core thermal power level of 2346 MWt and bounds the proposed power uprate. The reanalysis was provided in CP&L letters dated March 13, 2002 (RNP-RA/02-0027), and May 10, 2002 (RNP-RA/02-0067). Following approval of these AST submittals, the changes associated with the proposed power uprate will have no impact on control room habitability.

3.10.4 Radiological Consequences

HBRSEP, Unit No. 2, has reanalyzed the radiological dose consequences associated with the Fuel Handling Accident (FHA), Large Break LOCA, MSLB, Locked Rotor, Steam Generator Tube Rupture (SGTR), and Single Rod Control Cluster Assembly (RCCA) Withdrawal utilizing an Alternative Source Term (AST). The analyses were performed in accordance with Regulatory Guide (RG) 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Plants," July 2000. The FHA analysis was submitted to the NRC for review and approval by letter dated March 13, 2002 (RNP-RA/02-0027). The remainder of these dose consequence analyses were submitted to the NRC for review and approval by letter (RNP-RA/02-0067) dated May 10, 2002. The reanalysis of these events assumed a core thermal power level of 2346 MWt and bounds the proposed power uprate.

3.10.5 Containment Performance

The containment analyses discussed in UFSAR 6.2.1 were performed at a power level of 2346 MWt (102 percent). However, due to slight changes in RCS conditions as well as the increased Feedwater Regulation Valve (FRV) flow capacity, the containment response to a LOCA and a MSLB under the proposed power uprate conditions was re-evaluated. The evaluation determined that there are no changes in the plant parameters following the proposed uprate that would change the results or conclusions of the LOCA analysis presented in UFSAR 6.2.1.

The scope of MSLB evaluations performed to determine the impact of changes associated with the power uprate was limited to HFP cases, since the HZP cases would not be affected by these changes. The results of the analyses showed that the HZP Steam Line Check Valve failure case continues to produce the limiting containment pressure response with a peak pressure of 41.85 psig; therefore, the changes associated with the proposed power uprate will not cause a postulated pipe failure in containment (LOCA or MSLB) to exceed the containment design pressure (42 psig).

3.10.6 Anticipated Transient without Scram

In compliance with 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transient Without SCRAM (ATWS) Events for Light-Water-Cooled Nuclear Power Plant," ATWS mitigation circuitry has been incorporated into the HBRSEP, Unit No. 2, design. The purpose of the Anticipated Transient Without Scram (ATWS) System is to automatically initiate a turbine trip and AFW start under conditions indicative of an ATWS and a loss of feedwater.

The ATWS Mitigation Actuation Circuitry System (AMSAC) actuates AFW and a main turbine trip upon detection of low SG level coincident with main turbine load greater than 40 percent. There are time delays associated with each of the parameter measurements for initiation—approximately 360 seconds for turbine load, and approximately 25 seconds for initiation of AFW and main turbine trip, when enabled.

The ATWS mitigation system has been reviewed with respect to the proposed power uprate. Some instrument rescaling and calibration, and programming of the AMSAC controls, will be required for the main turbine 1st stage pressure input to the AMSAC due to replacement of the high-pressure turbine rotor, and to a lesser degree, higher steam flows resulting from the power uprate. There are no other impacts on ATWS System as a result of changes associated with the proposed power uprate.

3.10.7 Environmental Qualification

The HZP Steam Line Check Valve failure MSLB case continues to produce the limiting containment pressure response and component temperatures under uprated power conditions. Therefore, the current containment pressure and temperature profiles will remain bounding for the proposed power uprate.

For the LBLOCA, there are no changes in the plant parameters following the proposed uprate that would change the results or conclusions of the LOCA analysis presented in UFSAR 6.2.1.

Changes to the MFRV trim were evaluated and found to be not significant, since feedwater is isolated shortly after event initiation. The analysis of record conservatively assumes that main feedwater flow is directed to the faulted steam generator until feedwater isolation occurs. Thus, the analysis of record is not impacted by changes in the Main Feedwater System.

An evaluation has shown that the steam generator blowdown operating conditions will be slightly reduced by the changes associated with the proposed power uprate. Therefore, the current temperature and pressure profiles for the pipe penetration gallery will remain bounding. There is no impact on equipment located in this area as a result of the proposed power uprate.

The impact of the power uprate on radiological doses to EQ equipment was also evaluated. The inventory of radioisotopes released from the core following a design basis accident (DBA) LOCA was determined by assuming a limiting peak fuel assembly burnup and a limiting core average burnup. The power levels required to achieve these assumed burnup values are well above both the current and the proposed power levels.

In addition to the inventory of radioisotopes released from the core, the evaluation considered the contribution of the normal RCS radioactivity to the DBA LOCA source term. Compliance with the current RCS specific activity limits of TS 3.4.16 will ensure that the RCS equilibrium activities listed in UFSAR Table 11.1.1-2 will remain bounding under uprate conditions. Additionally, compliance will also be assured under the more restrictive TS 3.4.16 RCS specific activity limits proposed in the AST submittal letter (RNP-RA/02-0067) dated May 10, 2002. Based on this, the post-accident radiation environments in containment and in areas containing sump recirculation equipment will not be impacted by the proposed power uprate.

The impact of the uprate on normal operational doses to EQ equipment has been evaluated. The approximate 1.7 percent increase in the normal operating dose is less than 0.3 percent of the total EQ dose. This increase is considered negligible because it is less than the round-off in the doses. This assessment is supported by the fact that following a previous power uprate from 2200 MWt to 2300 MWt in July 1979, the average yearly dose was not significantly different than the average doses prior to the uprate. Furthermore, the qualification testing performed for HBRSEP, Unit No. 2, components bounds the expected increase in radiation levels as a result of the uprate.

For areas inside of containment that are not directly exposed to radiation from the reactor vessel, radiation from N-16 decay in the reactor coolant loops, or areas outside of containment, the normal operational doses are due to the radionuclide concentrations in the reactor coolant. Compliance with the current RCS specific activity limits of TS 3.4.16 will ensure that the RCS activities will remain bounding under power uprate conditions. Additionally, compliance will also be assured under the more restrictive TS 3.4.16 RCS specific activity limits proposed in the AST submittal letter (RNP-RA/02-0067) dated May 10, 2002.

Based on these evaluations, the proposed power uprate will have no adverse impact on the pressure, temperature, radiation, or chemical environments used in the environmental qualification of equipment. No EQ component replacement intervals or EQ component maintenance activities require revision based on the changes associated with the proposed power uprate.

3.10.8 Flooding

Protection from flooding is afforded by features of HBRSEP, Unit No. 2, that are not affected by the changes associated with the proposed power uprate. These features include: the physical relationship between plant grade and lake elevation, condenser circulating water piping location, and auxiliary building construction that does not contain any watertight spaces.

No piping or pump modifications for the HBRSEP, Unit No.2, water systems are necessitated by the proposed power uprate. Therefore, the leakage conditions with maximum flood potential (i.e., pipe break with pump run out) for the high volume water systems (service water, circulating water, component cooling water, fire protection water, etc.) are not impacted by the proposed power uprate. Therefore, the changes associated with the proposed power uprate do not impact flooding.

3.11 Nuclear Fuel

This section summarizes the evaluations performed to determine the effect of the proposed power uprate on the nuclear fuel. The nuclear fuel review for the power uprate to 2339 MWt evaluated the mechanical and thermal-hydraulic fuel design.

3.11.1 Fuel Core Design

The effects of power uprate conditions on nuclear fuel core design were analyzed using the current fuel cycle (Cycle 21). These analyses concluded that the power uprate would not necessitate a change to the physical design of the reactor core, fuel assemblies, or core components.

The design criteria used in the current mechanical design analysis for the fuel rods and fuel assemblies will continue to be satisfied under uprate power conditions to a peak assembly average exposure of 57,000 MWD/MTU, and a peak rod exposure of 62,000 MWD/MTU.

The mechanical design analysis also evaluated overpower conditions of up to 118 percent of the uprated power level (i.e., 2760 MWt). The mechanical parameters evaluated at these conditions included: fuel rod internal hydriding; cladding collapse; cladding stress and strain; fuel rod mechanical fracture; fuel densification and swelling; fuel assembly stress, strain, and loading limits; fuel rod fatigue; fuel rod fretting wear; fuel rod oxidation, hydriding, and crud buildup; fuel rod bow; fuel rod and fuel assembly axial growth; fuel rod internal pressure; fuel assembly liftoff; fuel assembly handling loads; and fuel assembly structural deformation.

The power uprate will not result in changes to current nuclear design methodologies or limits. Core design and power distribution control will continue to be performed using the approved methodologies listed in HBRSEP, Unit No. 2, TS 5.6.5.b.

Evaluation of the fuel rod/fuel assembly thermal-hydraulic analyses, core safety limits, and UFSAR Chapter 15 safety analyses was also performed for the current fuel cycle (Cycle 21) at the uprated core power level of 2339 MWt.

The evaluation of events that challenge the DNBR acceptance criterion concluded that the primary effect of the power uprate will be a slight increase in the vessel average temperature which degrades the DNBR by 1-2 percent. However, the application of the approved statistical DNBR methodology as allowed by TS 5.6.5.b, Reference 19, would more than offset the reduction in DNBR.

The evaluation also concluded that changes to thermal-hydraulic properties due to the change in RCS temperature will have a negligible effect on the timing of reactor trips, including the OPDT and OTDT trips, and high flux reactor trips. Additionally, the peaking factors ($F_q=2.46$, $F\Delta H=1.80$) used in the current analyses for the OPDT and OTDT trips will remain valid under power uprate conditions, and the current OPDT and OTDT setpoints and Core Safety Limit Lines, as revised to indicate re-designation of the high flux trip for power uprate conditions, will remain valid under power uprate conditions.

Finally, the evaluation concluded that the fuel centerline melt limit analyses will not be significantly impacted by the power uprate, and that rod bow penalties would be bounded by the analysis of record, and that the results of the limiting cases for the UFSAR Chapter 15 Analyses of Record either remain bounding or the impact is negligible.

Table 3.2-1

Two-Sigma Secondary Calorimetric Power Measurement Uncertainty Components

Parameter	LEFM Check Plus™ Uncertainty (%)
A. Feedwater Mass Flow, LEFM[2,4]	±0.2828
B. Feedwater Temperature[1]	+0.0433/-0.0428
C. Main Steam Pressure[1]	+0.0104/-0.0095
D. Main Steam Pressure, SG Blowdown[1]	±0.0005
E. Blowdown Flow[1]	±0.0034
F. CPC Factor[3]	±0.0391
G. Total RMS Uncertainty	+0.296/-0.295

Notes:
1. Instrument uncertainty per loop (3 loops).
2. Includes uncertainty due to hydraulics (profile factor), geometry (dimensions, alignment, thermal expansion), feedwater density and enthalpy correlations (feedwater pressure measurement and temperature determination), and time measurements (transit times and non-fluid delay).
3. Core Power Correlation (CPC) due to various heat gains and losses, calculated to equal to 30.6764×10^6 btu/hr.
4. This input to the uncertainty calculation is not multiplied by the number of feedwater loops, since the three loops are perturbed in the same manner, consistent with the bounding uncertainty calculation performed by the manufacturer.

$$\text{Uncertainty}_{\text{LEFM}} = [(B)^2+(C)^2*3+(D)^2*3+(E)^2*3+(F)^2*3+(H)^2]^{1/2}$$

Table 3.3-1

NSSS Operating Conditions for Current and Power Uprate Cases

Parameter		Case 1	Case 2	Case 3
		Current Full Power Operating Condition	Uprated Full Power Operating Conditions	
			0% SGTP	6% SGTP
Core Thermal Power (MWt)		2300	2339	2339
Non-Nuclear RCS Heat Sources (MWt)		9	9	9
Total Thermal Power (MWt)		2309	2348	2348
Tcold (°F)		547.6	547.6	547.3
Thot (°F)		603.2	604.1	604.5
Tavg (°F)		575.4	575.9	575.9
Tsteam (°F)		521.8	521.4	518.5
Psteam (psig)		809.0	806.2	785.8
Core Outlet Temperature (°F)		606.5	607.5	607.9
RCS Mass Flow Rate (lbm/hr)		106.48×10^6	106.48×10^6	105.28×10^6
RCS Volumetric Flow Rate (gpm)		282,349	282,349	279,039
Feedwater/Steam Loop Flow Rate (lbm/hr) – (3 loops)	Tfw = 435 °F	10.05×10^6	10.22×10^6	10.21×10^6
	Tfw = 440 °F	10.12×10^6	10.29×10^6	10.28×10^6
	Tfw = 445 °F	10.19×10^6	10.36×10^6	10.35×10^6

Table 3.6-1

Uprated NSSS Operating Conditions versus RCS Design Conditions
(Original and Revised)

Parameter	Uprated Operating Conditions		RCS Design Conditions[1] (original)	RCS Design Conditions[2] (revised)
	0% SGTP	6% SGTP		
Thot (°F)	604.1	604.5	610.9	604.6
Tavg (°F)	575.9	575.9	--	--
Tcold (°F)	547.6	547.3	554.7	546.1

Notes:
1. RCS design conditions from original reactor vessel, steam generators, and pressurizer equipment specifications.
2. RCS design conditions from replacement steam generator equipment specification.

4.0 MISCELLANEOUS

4.1 Affected Plant Systems

4.1.1 Control Room

A Control Room alarm is added due to installation of the LEFM Check Plus™ System to indicate conditions that could adversely affect availability of the LEFM Check Plus™ system instrumentation. Operator response to this alarm will be provided within an approved Annunciator Panel Procedure (APP). The APP will specify the actions required upon a loss of LEFM Check Plus™ System instrumentation.

Plant parameters displayed in the control room will experience minor changes due to the power uprate. Those parameters that are determined to be outside of their existing indicating bands are addressed within design packages that include plant changes, such as span and scaling, due to the proposed power uprate.

4.1.2 Simulator

The proposed power uprate will require changes to the simulator to ensure that it continues to accurately reflect plant status and physical appearance (hardware), and simulation of plant response (software). These changes will range from simple modifications to process temperatures and flow rates, to plant responses to accidents and transients.

Hardware and software changes to the simulator are implemented through plant approved change processes. The hardware and software changes, including changes involving plant process computer inputs, which affect operator performance, will be completed prior to operation at the uprated power level. Simulator revalidation is performed in accordance with ANSI/ANS-3.5-1998, Section 4.4, "Simulator Testing."

4.2 Operating Procedures and Operator Actions, Training, and Simulator

4.2.1 Operating Procedures (Abnormal/Normal), Emergency Operating Procedures, and Off-Normal Procedures

The power uprate will lead to minor changes in several plant parameters, including the 100 percent value for rated thermal power, reactor coolant system delta temperature, 1st stage turbine pressure, turbine governor valve positions, steam generator pressure, and main steam and feedwater flows. Therefore, the proposed power uprate is expected to have a limited effect on the manner in which the operators control the plant during normal operations, and transient and emergency conditions.

Necessary changes to Normal and Abnormal Operating Procedures, Emergency Operating Procedures, and Off-Normal Procedures will be made in accordance with the plant-approved process for plant modifications and will be in place prior to use at the uprated power conditions.

Administrative controls to prevent operation above the licensed power level will continue to be provided in Plant Operating Manual procedures. These controls include a statement that steady-state reactor power shall be maintained at or below the licensed power level, and management expectations to minimize temporary operation above the licensed power level. Plant operating procedures will be revised to indicate that temporary operation above the licensed power level shall be limited to approximately 0.3 percent of RTP, consistent with the reduced power measurement uncertainty prior to operation at uprated power levels.

4.2.2 Operator Training and Simulator

Simulator training will be provided on power uprate-related changes to the plant that affect operator performance prior to operating at uprated power levels. Training will be also be provided on power uprate-related changes to plant procedures that affect operator performance prior to their use at the uprated power conditions. Changes to the training simulator that are made necessary due to the proposed power uprate are performed consistent with ANSI/ANS-3.5-1998, and the simulator will be validated in accordance with ANSI/ANS-3.5-1998, Section 4.4, "Simulator Testing." Modification and revalidation of the training simulator to ensure that it continues to accurately reflect plant status, physical appearance, and simulation of plant response is discussed further in Section 4.1.2.

4.3 Station Blackout Event

The HBRSEP, Unit No. 2, response to the station blackout rule, 10 CFR 50.63, "Loss of All Alternating Current Power," is contained in the "Station Blackout (SBO) Coping Analysis (Document 8S19-P-101)." The Station Blackout (SBO) Coping Analysis was developed using the guidance provided in NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," and complies with the intent of NRC Regulatory Guide 1.155, "Station Blackout."

HBRSEP, Unit No. 2, is classified as an alternate AC plant that is required to cope for 8 hours following an SBO event. During the first hour of the SBO event it is assumed that there is no available AC power. After the first hour, and for the remainder of the 8-hour event, AC power is available for dedicated shutdown equipment (i.e., charging pump, service water pump, component cooling water pump, battery chargers, etc.) from the Dedicated Shutdown Diesel Generator.

During the first hour of the SBO event, decay heat removal is provided using Auxiliary Feedwater (AFW). As a result of the power uprate-related increase in decay heat rate, there will be a coincident increase in condensate inventory requirements from the Condensate Storage Tank (CST) during the SBO event. The current condensate volume required to remove decay heat during the first hour of an SBO event is 23,000 gallons (with no cooldown), and 38,840 gallons is required to achieve a cooldown rate 100°F/hr during the first hour.

Under power uprate conditions, the required condensate volume for decay heat removal during the first hour of the SBO event will be 23,270 gallons (with no cooldown), and 39,340 gallons will be required to achieve a cooldown rate 100°F/hr during the first hour. These inventory requirements are substantially less than the 62,000 gallon administrative limit for CST volume, and no additional actions are required to accommodate the increase in condensate volume requirements.

The proposed power uprate will have no impact on the methodology or implementation of the HBRSEP, Unit No. 2, SBO Coping Analysis, and consequently will not adversely impact the ability of the plant to achieve and maintain safe shutdown conditions. The existing system designs are adequate to accommodate the increased decay heat removal associated with operation at higher power levels.

4.4 10 CFR 50, Appendix R, Safe Shutdown

10 CFR 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," requires that fire protection be provided for structures, systems, and components required for safe shutdown. The HBRSEP, Unit No. 2, safe shutdown analysis and methodology is described in plant document FPP-RNP-300, Rev. 6, "10 CFR 50 Appendix R Section III.G Safe Shutdown Component/Cable Separation Analysis."

In order to satisfy the Appendix R requirements for ensuring adequate core cooling, and also to maintain a minimum inventory in the steam generators, it is currently necessary that the AFW System be initiated within a specified time. Based on evaluation of the power uprate, the current AFW System initiation time is not impacted.

The proposed power uprate will have no impact on the methodology or implementation of the HBRSEP, Unit No. 2, Appendix R safe shutdown analysis, and consequently will not adversely impact the ability of the plant to achieve and maintain safe shutdown conditions. The existing system designs are adequate to accommodate the increased decay heat removal associated with operation at higher power levels.

4.5 Safety-Related Valves

4.5.1 Generic Letter 89-10, "Safety Related Motor-Operated Valve Testing and Surveillance"

Implementation of the HBRSEP, Unit No. 2, Generic Letter 89-10 Program will be unaffected by the proposed power uprate. The Generic Letter 89-10 Program was created in response to Generic Letter 89-10, and Generic Letter 96-05, "Periodic Verification of

Design-Basis Capability of Safety-Related Motor-Operated Valves," and ensures that active safety-related motor-operated valves (MOVs) are capable of performing their required functions when subjected to design basis conditions during both normal operation and abnormal events. The program provides for the testing, inspection, and maintenance of these MOVs.

The HBRSEP, Unit No. 2, Generic Letter 89-10 Program currently includes 58 valves. The proposed power uprate will not change this population of valves. Additionally, the proposed power uprate will not require increasing the existing maximum opening and closing differential pressures for valves in the program.

4.5.2 Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety Related Operated Gate Valves"

Implementation of the HBRSEP, Unit No. 2, Generic Letter 95-07 Program will be unaffected by the proposed power uprate. The Generic Letter 95-07 Program was created in response to Generic Letter 95-07 to ensure that safety-related, power-operated (e.g., motor-operated, air-operated, and hydraulically-operated) gate valves are evaluated for susceptibility to pressure locking and thermal binding, and to ensure that susceptible valves are capable of performing safety-related functions.

The proposed power uprate will not change the population of valves in the Generic Letter 95-07 Program. Most of the valves in the Generic Letter 95-07 Program that are susceptible to pressure locking or thermal binding have already been modified such that the power uprate will have no impact. Two sets of valves (SI injection valves SI-870A and B, and pressurizer PORV block valves RC-535 and RC-536) have not been modified. For these valves, the mechanism responsible for causing pressure locking or thermal binding is not impacted by the proposed power uprate.

4.5.3 Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions"

A review of existing evaluations performed to satisfy Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design Basis Accident Conditions," was performed to determine whether the proposed power uprate would adversely affect any of the previous conclusions related to containment integrity. The review examined the potential for overpressurization of safety-related, water-filled, isolable piping sections located inside containment, and also considered the potential for water hammer and two-phase flow issues associated with the containment fan cooler units.

The conclusions reached in the existing Generic Letter 96-06 documentation and evaluations will remain valid under the proposed power uprate. The calculated post-LOCA containment analysis used in the Generic Letter 96-06 evaluation was performed assuming operation at 102 percent of current licensed power and therefore remains bounding. Additionally, water hammer and two-phase flow issues are bounded by the TS limit on initial Service Water System temperature of 99°F, which is lower than the initial SW temperature of 100°F assumed in the Generic Letter 96-06 evaluation.

4.5.4 Air Operated Valves

The HBRSEP, Unit No. 2, Air Operated Valve (AOV) Program has been developed to detail a systematic approach for design evaluation, testing, and maintenance of AOVs to verify that AOVs providing active safety-related functions are capable of performing as required. The only power uprate-related impact to the AOV Program involves changes to the MFRV trim and stroke time, as discussed in Section 3.7.2.1.

The AOV Program was reviewed to determine if the power uprate would impact the program valve population or any valve differential pressure or stem thrust requirements. With the exception of the MFRVs, the proposed power uprate will not add, delete, or modify any program valves, and will not impact any valve differential pressure or stem thrust requirements.

For the MFRVs, these valves will remain in the program following the power uprate. However, thrust values may change as a result of the trim modification. Thrust values for these valves are revised in the AOV Program as necessary to reflect changes identified during post-modification testing in accordance with the plant design change control process.

4.5.5 Check Valves

The proposed power uprate will have no impact on the Check Valve Program. The HBRSEP, Unit No. 2, Check Valve Program incorporates industry experience, plant-specific experience, and programs to enhance check valve reliability. The program defines methods for performing reviews and establishing necessary Preventative Maintenance frequencies.

The power uprate will not add, delete, or modify any check valves in the program population. Therefore, the valve population will be unchanged. The check valve test conditions and performance criteria are related to fluid velocity. Check valve concerns are related to low velocity conditions that can result in excessive check valve cycling. The power uprate will not result in any flow velocities decreasing below the minimum velocities specified in the Check Valve Program for any safety-related, swing-type check valve greater than 2 inches. Additionally, the power uprate will not alter the physical orientation of any valves. As a result, the proposed power uprate will not impact the effects associated with design installation (i.e., proximity to bends, tees, valves, strainers, etc) of check valves in the program population.

4.6 Affected Plant Programs

The proposed power uprate has the potential to affect programs that are developed and implemented at HBRSEP, Unit No. 2, to demonstrate that topical areas comply with various design and licensing requirements. The plant programs listed in Table 4.6-1 were reviewed to determine the impact due to power uprate.

The review results are summarized in Tables 4.6-1, and indicate whether the affected program will need to be updated as a result of the power uprate. The review identified several programs that are affected by the power uprate. Established change processes will capture changes to the affected programs.

4.6.1 10 CFR 50, Appendix J, Primary Containment Leakage Rate Testing Program

The proposed power uprate will not impact the HBRSEP, Unit No. 2, Appendix J Testing Program. The Appendix J Testing Program was developed in accordance with the requirements of 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." The program is implemented by procedures that provide program instructions and establish the plant position on the requirements of Appendix J to 10 CFR 50. The Appendix J Testing Program consists of a schedule for performing Type A, B, and C tests for leak testing the containment and related systems, and components that penetrate the containment pressure boundary. The proposed power uprate will not modify any containment penetrations or the containment. Additionally, the power uprate will not change the calculated peak containment pressure upon which the leak testing conditions and acceptance criteria are based.

A separate license amendment request for a one-time extension to the current 10-year test interval for Type A leakage rate testing, in accordance with 10 CFR 50, Appendix J, Option B, was submitted to the NRC for review and approval by letter dated March 26, 2002 (RNP-RA/02-0028). This amendment request is unrelated to the proposed power uprate request and does not alter the conclusions presented herein.

4.6.2 Inservice Testing Program

The HBRSEP, Unit No. 2, Inservice Testing (IST) Program provides for the Fourth Ten (10) Year Interval Inservice Pump and Valve Testing Program, as specified in 10 CFR 50.55a, paragraph (f)(4).

With respect to pump testing, the proposed power uprate will not add, delete, or modify any pumps that are currently in the program. In addition, the power uprate will not impact any of the parameters that the IST pump testing program measures or observes. Therefore, the power uprate will not impact pump test conditions or performance requirements.

Regarding valve testing, the power uprate will not be adding, deleting, or modifying any valves, with the exception of the MFRVs. As a result of the power uprate, the MFRV trim will be modified, and the TS stroke time requirement for MFRV and Bypass Valve closure will be revised from ≤ 30 seconds to ≤ 20 seconds to ensure that the calculated containment pressure during postulated accidents remains below the allowable limit. Reference values and acceptance criteria for these valves are revised in the IST Program as necessary to reflect changes identified during post-modification testing in accordance with the plant design control process. Therefore, with the exception of the MFRVs and Bypass Valves, the proposed power uprate will not change any valve test conditions or performance requirements.

4.6.3 Maintenance Rule Program

The HBRSEP, Unit No. 2, Maintenance Rule Program is designed to fulfill the Maintenance Rule requirements of 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The objective of the Maintenance Rule is to provide reasonable assurance that structures, systems, and components (SSCs) are capable of fulfilling their intended safety functions. The SSCs were determined using scoping fields (i.e., safety related; relied upon to mitigate accidents or transients; used or use implied in the Emergency Operating Procedures (EOPs); failure that could prevent a safety function; and failure that could cause a reactor scram or Engineered Safety Features (ESF) actuation).

The proposed power uprate will not impact any scoping field. Plant SSCs are monitored by plant-level performance criteria (i.e., plant trips and shutdowns, unplanned production losses, and ESF actuations). HBRSEP, Unit No. 2, has established a criterion for unplanned production loss that is currently based on operation at 718 MWe. Revision of this criterion to reflect changes resulting from the power uprate is being tracked as an action item in the Maintenance Rule Program database.

4.6.4 Fire Protection Program

The Fire Protection Program at HBRSEP, Unit No. 2, is based on Nuclear Regulatory Commission (NRC) criteria, National Fire Protection Association (NFPA) standards, Institute of Electrical and Electronic Engineers (IEEE) standards, and other industry codes. The program complies with the intent of Appendix A of Branch Technical Position (BTP) APCSB 9.5-1, dated August 23, 1976.

The objective of the Fire Protection Program is to minimize both the probability and consequence of postulated fires. The probability and consequences are minimized by a combination of design features, procedural controls, and personnel training, including a well-trained fire brigade.

The procedural controls include provisions to ensure that plant modifications and design changes are controlled in order to ensure that plant structures, systems, and components continue to meet their performance and functional objectives. Plant procedures provide written instructions that describe the modification process and the means for documenting the required changes and activities. As a part of this process, consideration of the effects the modification may have on the Fire Protection Program is required.

The combustible equipment and changes to the plant systems, structures, and components that are being installed or modified to support the proposed power uprate have been evaluated with respect to their impact on the Fire Protection Program. Fire loading margins will not be challenged or exceeded by the power uprated related plant modifications. Consequently, there is no impact to the Fire Protection Program as a result of the proposed power uprate.

4.6.5 Flow Accelerated Corrosion Program

The purpose of the HBRSEP, Unit No. 2, Flow Accelerated Corrosion (FAC) Program is to maintain design margin in pipe wall thickness. The program is implemented by plant procedures and fulfills the requirements of Generic Letter 89-08, "Erosion-Corrosion Induced Pipe Wall Thinning." The FAC Program uses the CHECWORKS computer program to model FAC in piping systems.

An evaluation of plant piping systems identified a number of feedwater heater components that may exhibit susceptibility to FAC under power uprate operating conditions. These components are presently modeled in the FAC Program, and are discussed in greater detail in Section 3.8.3. In accordance with the provisions of the FAC Program, the projected power uprate operating conditions (i.e., flow and thermodynamic states) for these components are updated in the CHECWORKS model, as appropriate, and results of these models are factored into future pipe inspection and replacement plans.

4.7 Probabilistic Safety Assessment Summary

The HBRSEP, Unit No. 2, Probabilistic Safety Assessment (PSA) is based on the original Individual Plant Examination (IPE) analysis Probabilistic Risk Assessment (PRA) model, and has been periodically updated to reflect plant configuration changes, plant operating data, or industry standard practices. It is a model that employs probabilistic techniques, best-estimate methodologies, and engineering judgment to provide risk insights.

An evaluation of the proposed power uprate determined that there is no impact to initiating event identification, accident sequences evaluation, event trees, success criteria, or containment response as a result of the proposed power uprate. The evaluation also considered risk-important operator actions credited in the PSA and found the impact due to power uprate to be negligible. The procedural basis for operator actions credited in the PSA is not affected by the power uprate, and the timing for events and human actions will not be significantly impacted. Consequently, the risk significance of the proposed power uprate is negligible.

4.8 Radioactive Waste Disposal Systems

The Radioactive Waste (Radwaste) Management, as discussed in Chapter 11 of the HBRSEP, Unit No. 2, UFSAR, addresses system/component activity inventories and activity releases associated with the liquid, gaseous, and solid waste management systems, as well as the process and effluent radiological monitoring and sampling systems.

Gaseous waste is stored in the Waste Decay Tanks prior to release. Recent history indicates that annual releases are a small fraction of regulatory limits.

The solid and liquid waste management systems are designed to control, collect, store, process, and dispose of radioactive waste due to normal operation of the plant, including anticipated transients. Operation of these systems is primarily influenced by the volume of wastes processed, which is not expected to change as a result of the proposed power uprate.

Radwaste management is based on activity profiles, which assume 1 percent failed fuel at the present licensed power level. Compliance with the plant Technical Specifications and Technical Requirements Manual requirements ensure that changes that result from the power uprate will remain well within the current activity profiles.

The proposed power uprate has no significant impact on waste subsystems, or components of these systems. These systems are typically operated in a batch mode, such that the only potential effect of the power uprate is a slight increase in the frequency at which the batches are processed. The radioactive waste disposal systems continue to have sufficient capacity to meet station processing demands and control the release of radioactivity to well below 10 CFR 50, Appendix I, and 10 CFR 20 limits.

4.9 Radiation Protection

The proposed power uprate will not result in radiation exposures in excess of the criteria (for restricted and unrestricted areas) provided in 10 CFR 20, "Standards for Protection Against Radiation." From an operations perspective, radiation levels in most areas of the plant are expected to increase by no more than the percent increase in uprated power. Individual worker exposures will be maintained within acceptable limits by the site ALARA Program, which controls access to radiation areas.

Gaseous and liquid effluent releases are also expected to increase by no more than the percent increase in uprated power level. Offsite release concentrations and doses will be maintained within the limits of the current 10 CFR 20 and 10 CFR 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as Reasonably Achievable' for Radioactive Material in

Light-Water-Cooled Nuclear Power Effluents," by the site radioactive effluent control program.

4.9.1 Radiation Sources During Normal Operation

The maximum normal operation radiation sources will be unchanged from the values in effect prior to the power uprate, with the exception of doses resulting from neutron flux and N-16, which only affect dose rates inside containment. TS 3.4.16, "RCS Specific Activity," and TS 3.4.13, "RCS Operational Leakage," limit the primary activity and primary-to-secondary leakage, respectively. These TS limit the normal radiation sources to less than the original design basis values reflected in Chapter 11 of the HBRSEP, Unit No. 2, UFSAR, and are unchanged. Therefore, the maximum allowable normal operation radiation sources, with the exception of neutron flux and N-16, are unchanged and bounded by the existing design basis sources.

4.9.2 Normal Operation Offsite Doses

TS 3.4.16, "RCS Specific Activity," and TS 3.4.13, "RCS Operational Leakage," limit the primary activity and primary-to-secondary leakage, respectively. TS 5.6.2, "Annual Radiological Environmental Operating Report," also provides requirements for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. These requirements are implemented by the Offsite Dose Calculation Manual (ODCM) and plant procedures. Additionally, a review of recent Annual Radioactive Effluent Release Reports demonstrates that the actual releases from the plant are historically a very small percentage of the allowable limits. Therefore, the offsite doses from normal effluent releases will remain significantly below the bounding limits of 10 CFR 50, Appendix I following the power uprate.

4.9.3 Shielding for Normal Operation

The shielding designs and radiation zones outside containment are based on the source terms described in Chapters 11 and 12 of the HBRSEP, Unit No. 2, UFSAR. As applied to shielding for areas outside containment, the source terms for normal operations remain bounding for the power uprate, such that the power uprate has no impact on shielding designs or radiation zones outside containment. The plant TS control the normal radiation sources to less than the original design basis values assumed in UFSAR 11.1 and 12.2.

The dose rates inside containment may increase by the same percentage as the power uprate (i.e., approximately 1.7 percent) under uprated power conditions due to increases in neutron flux and N-16. The increase in neutron flux dose rates is not expected to be significant because of the ability of fuel management techniques to influence neutron flux leakage at the perimeter of the reactor core, thereby making it possible that source term and dose rates inside containment could remain bounding depending on the fuel management program and initially assumed N-16 generation rate. During shutdown, dose rates inside containment are affected by many factors (i.e., hot spots in piping, shutdown evolutions, fuel performance, etc.). The small increase in power is expected to have an insignificant effect on these factors, and it is expected that dose rates in containment during shutdown will be less than the percent power increase.

Current occupational doses are well below the dose criteria in 10 CFR 20, and will remain below the 10 CFR 20 limits following an approximate 1.7 percent increase in source term due to the power uprate. Individual worker exposures will be maintained within acceptable limits by the site ALARA Program, which controls access to radiation areas. The 2000 Occupational Exposure Report demonstrates that the doses incurred by plant personnel are historically a small percentage of the allowable limits.

4.10 Heating, Ventilation, and Air Conditioning Systems

4.10.1 Non-Radiological Impact

The normal environments for plant buildings (i.e., Control Room, Containment, Reactor Auxiliary Building, Fuel Handling Building, Turbine Building, Radwaste Building, Relay Room #2, Cable Room #2, etc.) were assessed to determine the impact of the power uprate on their associated HVAC systems. The proposed power uprate has a negligible effect on process fluid temperatures in the Reactor Auxiliary Building and Turbine Building. With the exception of small increases in motor brake horsepower for the feedwater and condensate pumps, the increase in heat loads for areas outside containment is caused by the increase in decay heat load, which is transferred to the CCW System and SW System.

The effect of increased heat loads for areas outside containment has been evaluated to have an insignificant impact on the associated plant HVAC Systems. The small expected changes in fluid temperature were also found to have an insignificant effect on area temperatures.

For the plant areas other than the containment, the performance characteristics (i.e., flow, pressure) of their associated HVAC systems are not changed by the power uprate. Therefore, there is no non-radiological impact on the adequacy of these plant ventilation systems (i.e., frequency of air change, pressure differential) for area access.

Similar conclusions were reached following evaluation of the normal environment in the containment. The containment environment is controlled during normal operation in accordance with the plant TS. The increase in heat load inside containment results primarily from the small increase in T_{hot}. This increase is bounded by the margin provided in the original design.

For post-LOCA environments located outside the containment, there is no increase in the post-LOCA ECCS motor brake horsepower ratings, since there is no increase in recirculation water temperature. Therefore, the design heat loads post-LOCA for environments outside containment are not impacted by the proposed power uprate.

4.10.2 Control Room Heating, Ventilation, and Air Conditioning System

The Control Room HVAC System is part of the Control Room Habitability System, and is comprised of two parts, an environmental control system and an air cleanup system.

The environmental control system operates continually during normal and emergency conditions. The environmental control system consists of redundant centrifugal fans and gravity dampers arranged in parallel, a medium efficiency filter, and redundant cooling coils. A non-safety related fan provides exhaust from the Control Room kitchen and toilet areas, and a non-safety related electric duct heater provides heating when required.

The air cleanup system includes a single outside air intake that connects to redundant, parallel control dampers. The Control Room kitchen and toilet area contain redundant control dampers that are installed in series.

The Control Room HVAC System is designed to provide three operational modes: normal ventilation, emergency pressurization, and emergency recirculation. During normal operation, a single train of both the environmental control system and environmental cleanup system are operated in conjunction with the kitchen and toilet area exhaust fan.

In the emergency pressurization operational mode, a single train of both the environmental control system and environmental cleanup system are operated, and the kitchen and toilet area fans are shut down with the associated exhaust dampers closed. An SI signal, or a signal from the Control Room radiation monitor, will automatically place the air cleanup system in the emergency pressurization operational mode. Positive pressure is maintained in the Control Room envelope with respect to adjacent areas and the outdoors in this operating mode (with a potential exception for limited plant areas following a postulated Auxiliary Building exhaust fan failure).

The emergency recirculation operational mode is activated by first placing the Control Room HVAC System in the emergency pressurization operational mode, and then isolating both outside air intake dampers. This operational mode is not a design basis requirement, but provides flexibility to allow isolation of Control Room outside air makeup.

In the event of a fire, portable fans provide Control Room smoke purge capability.

Non-radiological impacts of the power uprate on the Control Room HVAC System are discussed in Section 4.10.1.

The ability of the Control Room HVAC System to mitigate the radiological consequences of various events has been reanalyzed as part of the incorporation of an Alternative Source Term (AST). This reanalysis assumed a core thermal power level of 2346 MWt and bounds the proposed power uprate. The reanalysis was provided in CP&L letters dated

March 13, 2002 (RNP-RA/02-0027), and May 10, 2002 (RNP-RA/02-0067). Following approval of these AST submittals, the changes associated with the proposed power uprate will have no impact on Control Room habitability.

4.10.3 Reactor Containment Building Post-Accident Heating, Ventilation, and Air Conditioning System

The Reactor Containment Building Post-Accident Heating, Ventilation, and Air Conditioning (HVAC) System includes three subsystems: the Reactor Containment Air Recirculating Cooling System (CARCS), the Containment Purge System (CPS), and the Post Accident Containment Venting System (PACVS).

The CARCS consists of four air handling units, as well as the associated piping, valves, and instrumentation. During normal operation, air entering the CARCS is routed through normally open butterfly valves and normal dampers (during emergency operation the normal dampers close); a service water supplied air cooling coil; a centrifugal fan with direct motor drive; a gravity operated fan discharge damper; and exits the CARCS at the discharge of a ductwork air distribution system that is common to the four air handling units.

The CPS consists of two physically separated subsystems; a supply subsystem, and an exhaust subsystem. The exhaust subsystem has separate duct arrangements inside the containment for general purging and purging during refueling operations. Those portions of the CPS essential to the prevention or mitigation of the consequences of nuclear accidents are provided with protection against natural phenomena (i.e., high winds, earthquake, heavy icing, external flooding, missiles).

The PACVS consists of two full capacity supply lines through which instrument air or station air can be admitted to the containment, two full capacity exhaust lines through which hydrogen bearing gases may be vented from the containment, and associated valves and instrumentation. The supply lines include equipment and piping that provide instrument air and service air during normal operations. One of the exhaust lines employs equipment and piping that is used to provide normal pressure relief for the containment. The other exhaust line is PACVS dedicated.

Non-radiological impacts of the power uprate on the Reactor Containment Building Post-Accident HVAC System are discussed in Section 4.10.1.

With the exception of mitigation of the consequences of a fuel handling accident inside the containment, it is concluded that the proposed power uprate will not impact the safety or operational functions of the Reactor Containment Building Post-Accident HVAC Systems. The radiological consequences of the fuel handling accident inside containment have been reanalyzed utilizing an Alternative Source Term (AST) in accordance with 10 CFR 50.67. This analysis assumed a core thermal power level of 2346 MWt and bounds the proposed power uprate. The analysis was submitted for NRC review by letter (RNP-RA/02-0027) dated March 13, 2002.

Table 4.6-1

Plant Programs

Issues and Programs	Update Required
Plant Simulator	Yes
Fire Protection (Appendix R)	No
Air-Operated Valve Program	No[1]
Check Valve Program	No
Motor-Operated Valve Administrative Program (Generic Letter 89-10)	No
Inservice Inspection Program	No[1]
Equipment Qualification	No
Station Blackout	No
Anticipated Transient Without Scram	No
Flow Accelerated Corrosion Program	Yes
Steam Generator Program	No
Primary Containment Leakage Testing (Appendix J)	No
Maintenance Rule Program	Yes

Yes – Programs impacted and changes to be addressed during uprate implementation.
No – Programs not impacted by uprate change or bounded by existing analysis.

Notes:
1. Program impact to be determined for MFRVs following operation at uprated power conditions.

5.0 NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Carolina Power and Light (CP&L) Company is proposing changes to the Facility Operating License (OL) for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, including the Appendix A, Technical Specifications (TS). The following change is requested.

- Revision of the maximum reactor core power level stated in OL paragraph 3.A, and the TS 1.1 definition of "RATED THERMAL POWER (RTP)," from 2300 MWt to 2339 MWt
- Revision of the reactor core safety limit curve in TS 2.1.1, "Reactor Core SLs."
- Revision of the reference T_{avg} value in TS 3.3.1, "Reactor Protection System (RPS) Instrumentation."
- Revision of the Allowable Value for the "Steam Line High Differential Pressure Between Steam Header and Steam Lines" function in TS 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."
- Revision of the RCS pressure-temperature limit curves in TS 3.4.3, "RCS Pressure and Temperature (P/T) Limits."
- Revision of the Required Actions in TS 3.7.1, "Main Steam Safety Valves (MSSVs)."
- Revision of the Main Feedwater Regulation Valve (MFRV) and Bypass Valve stroke time Surveillance Requirements in TS 3.7.3, "Main Feedwater Isolation Valves, (MFIVs), Main Feedwater Regulation Valves (MFRVs), and Bypass Valves."

An evaluation of the proposed change has been performed in accordance with 10 CFR 50.91(a)(1) regarding no significant hazards considerations using the standards in 10 CFR 50.92(c). A discussion of these standards as they relate to this amendment request follows:

1. The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

   The proposed change described above does not involve a significant increase in the probability of an accident previously evaluated based on the results of comprehensive analytical efforts that were performed to demonstrate the acceptability of the proposed power uprate changes.

   An evaluation has been performed that identified the systems and components that could be affected by these proposed changes. The evaluation determined that these systems and components will function as designed and that performance requirements remain acceptable.

The primary loop components (reactor vessel, reactor internals, control rod drive mechanisms (CRDMs), loop piping and supports, reactor coolant pumps, steam generators and pressurizer) will continue to comply with their applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components leading to an accident.

The Leak-Before-Break analysis conclusions remain valid and the breaks previously exempted from structural considerations remain unchanged.

Systems included within the scope of the Nuclear Steam Supply System (NSSS) will continue to perform their intended design functions during normal and accident conditions. Additionally, NSSS components will continue to comply with applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components.

The NSSS/Balance of Plant interface systems will continue to perform their intended design functions. The MSSVs will provide adequate relief capacity to maintain the Main Steam System within design limits. The maximum feedwater flow rate and the isolation time for the MFRVs and Bypass Valves will continue to ensure that the analyzed containment pressure during postulated accidents remains below the allowable limit.

The current loss-of-coolant (LOCA) hydraulic analyses remain bounding.

The reduction in power measurement uncertainty achieved through the use of the Caldon Leading Edge Flow Meter (LEFM) Check Plus ™ system allows for certain safety analyses to continue to be used, without modification, at the 2346 MWt power level (102 percent of 2300 MWt). Other safety analyses performed at a nominal power level of 2300 MWt have been either re-performed or re-evaluated to support the 2339 MWt power level, and continue to meet their applicable acceptance criteria. Some existing safety analyses had been previously performed at a power level greater than or equal to 2346 MWt, and thus continue to bound the 2339 MWt power level.

The proposed changes to the RCS pressure-temperature limit curves impose a conservative projection of the increase in neutron fluence associated with the power uprate. This projection will ensure that the requirements of 10 CFR 50, Appendix G, "Fracture Toughness Requirements," will continue to be met following the proposed power uprate. The design basis events that were protected against by these limits have not changed, therefore, the probability of an accident previously evaluated is not increased.

Accident dose consequences have been evaluated assuming a bounding initial power level of 2346 MWt. The revised dose consequences were provided to the NRC in letters dated March 13, 2002, and May 10, 2002, requesting implementation of an Alternative Source Term. Upon NRC approval of the evaluations contained in these two submittals, the accident dose consequence will bound operation at the uprated power level of 2339 MWt.

The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed power uprate changes, as well as the proposed changes unrelated to the power uprate. Systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component, and do not challenge the performance or integrity of any safety-related system.

Based on the foregoing, it is concluded that this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident From Any Previously Evaluated.

   The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated because no new accident scenarios, failure mechanisms, or single failures are introduced as a result of the proposed power uprate changes. Systems, structures, and components previously required for the mitigation of an event remain capable of fulfilling their intended design functions. The proposed changes have no adverse effects on any safety-related system or component, and do not challenge the performance or integrity of any safety-related system.

   Thus, this change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

3. The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety.

   Extensive analyses of the primary fission product barriers conducted in support of the proposed power uprate have concluded that relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and compliance with regulatory acceptance criteria. As appropriate, evaluations have been performed using methods that have either been reviewed and approved by the Nuclear Regulatory Commission (NRC), or that are in compliance with applicable regulatory review guidance and standards.

Therefore, this change does not involve a significant reduction in a margin of safety.

Based on the above discussion, CP&L has determined that the requested change does not involve a significant hazards consideration.

6.0 ENVIRONMENTAL IMPACT CONSIDERATION

The environmental review conducted for the proposed power uprate assessed the existing operating license, the National Pollutant Discharge Elimination System (NPDES) permit limits for the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 1 and No. 2, and information contained in the HBRSEP, Unit No. 2, Final Environmental Report (FER). This assessment concluded that the power uprate would not cause the plant to exceed the discharge limitations and NPDES permit conditions associated with operation of the plant. Additionally, a review of recent HBRSEP, Unit No. 2, Annual Radioactive Effluent Discharge Reports demonstrates that actual releases from the plant are a small percentage of the Technical Specification limits. The discharge amounts will not be significantly increased by the thermal power uprate and will continue to be a small percentage of the allowable limits and FER estimates.

Onsite and offsite radiation exposures from normal operation and postulated accidents have been evaluated. The offsite doses postulated under accident conditions remain within the guidelines of 10 CFR 100.

The FER addresses the non-radiological impacts of plant operation as a function of plant design features, the relative loss of renewable resources, and the relative loss or degradation of available habitat. Environmental impacts associated with operating license issuance were evaluated in the FER. After weighing the environmental, economic, technical, and other benefits against environmental costs and considering available alternatives, and subject to certain conditions, from the standpoint of environmental effects, the FER concluded that the issuance of an operating license for HBRSEP, Unit No. 2, was an acceptable action. These assessments and the assumptions on which they are based remain valid and are not impacted as a result of the thermal power uprate.

6.1 National Pollutant Discharge Elimination System Permit Impact

The HBRSEP consists of two electric generating units. Unit No. 1 is a coal-fired facility, and Unit No. 2 is a nuclear powered facility. The NPDES permit and its associated limits are applied collectively to both the HBRSEP, Unit No. 1 and Unit No. 2, facilities. HBRSEP, Unit No. 2, employs an open-loop cooling systems that withdraws and returns cooling water from Lake Robinson. Cooling is accomplished by evaporation from the surface of Lake Robinson and by mixing with lake waters.

The HBRSEP NPDES permit (Permit No. SC0002925) places limits on the following plant discharges:

- Flow - 855 million gallons per day, maximum
- Discharge Temperature - 90.0°F to 111.2°F, maximum (depending on time of year)

- Dam Release Temperature - 91.4°F, maximum

The heat duty increase associated with the power uprate is mainly associated with the Circulating Water System and will be approximately 5.403×10^9 Btu/hr. This represents a 0.9 percent increase over the present heat duty of 5.355×10^9 Btu/hr, and is an insignificant change when compared to the current total heat load from the two HBRSEP units. The expected increase in temperature for circulating water discharged from HBRSEP, Unit No. 2, as a result of the uprate will be approximately 0.2°F, and will remain within current NPDES permit discharge temperature limits.

These NPDES discharge limits will not change as a result of the proposed power uprate, and HBRSEP, Unit No. 2, will continue to comply with its NPDES permit. Therefore, the proposed thermal power uprate for HBRSEP will have no adverse impacts on the environment and will not result in exceeding NPDES permit limits

The power uprate related changes to licensed power level and circulating water discharge temperature were submitted to the South Carolina Department of Health and Environmental Control as changes to the HBRSEP NPDES Permit Renewal Application in a letter dated March 27, 20002 (Serial: RNP-RA/02-0034).

6.2 Environmental Impact Consideration Summary

10 CFR 51.22(c)(9) provides criteria for identification of licensing and regulatory actions for categorical exclusion for performing an environmental assessment. A proposed change for an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed change would not (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite; (3) result in an increase in individual or cumulative occupational radiation exposure. Carolina Power and Light (CP&L) Company has reviewed this request and determined that the proposed change 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 needs to be prepared in connection with the issuance of the amendment. The basis for this determination follows.

Proposed Change

The proposed change would revise the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, Technical Specifications and Facility Operating License to support an increase in the authorized rated thermal power level from 2300 MWt to 2339 MWt (approximately 1.7 percent). The change is based on recovery of measurement uncertainty in the analytical margin originally required of Emergency Core Cooling System (ECCS) evaluation models in accordance with the requirements set forth in 10 CFR Part 50, Appendix K.

Basis

The proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) for the following reasons:

1. As demonstrated in the No Significant Hazards Consideration Determination, the proposed change does not involve a significant hazards consideration.

2. As demonstrated in the No Significant Hazards Consideration Determination, the proposed change does not result in a significant increase in the consequences of an accident previously evaluated and does not result in the possibility of a new or different kind of accident. Therefore, the proposed change does not result in a significant change in the types or significant increases in the amounts of any effluents that may be released offsite.

3. Radiation levels in most areas of the plant are expected to increase by no more than the approximate 1.7 percent increase in uprated power. Individual worker exposures will be maintained within acceptable limits by the site "As Low as Reasonably Achievable" (ALARA) Program, which controls access to radiation areas.

   TS 3.4.16, "RCS Specific Activity," and TS 3.4.13, "RCS Operational Leakage," limit the primary activity and primary-to-secondary leakage, respectively. TS 5.6.2, "Annual Radiological Environmental Operating Report," also provides requirements for maintaining doses to members of the public from radioactive effluents as low as reasonably achievable. These requirements are implemented by the Offsite Dose Calculation Manual (ODCM) and plant procedures. Additionally, a review of recent Annual Radioactive Effluent Release Reports demonstrates that the actual releases from the plant are historically a very small percentage of the allowable limits. Therefore, the proposed change does not result in a significant increase in individual or cumulative occupational radiation exposures.

United States Nuclear Regulatory Commission
Attachment III to Serial: RNP-RA/02-0066
12 Pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
REQUEST FOR TECHNICAL SPECIFICATIONS
CHANGES TO INCREASE AUTHORIZED REACTOR POWER LEVEL

MARKUP OF TECHNICAL SPECIFICATIONS PAGES

3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20. Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Section 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; 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 below:

   A. <u>Maximum Power Level</u>

   The licensee is authorized to operate the facility at a steady state reactor core power level not in excess of ~~2300~~ megawatts thermal.

   2339

   B. <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. ~~191~~ are hereby incorporated in the license.

   The licensee shall operate the facility in accordance with the Technical Specifications.

   (1) For Surveillance Requirements (SRs) that are new in Amendment 176 to Final Operating License DPR-23, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 176. For SRs that existed prior to Amendment 176, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 176.

   C. <u>Reports</u>

   Carolina Power & Light Company shall make certain reports in accordance with the requirements of the Technical Specifications.

   D. <u>Records</u>

   Carolina Power & Light Company shall keep facility operating records in accordance with the requirements of the Technical Specifications.

4. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No ~~191~~, are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the additional conditions.

5. This license is effective as of the date of issuance and shall expire at midnight July 31, 2010.

Attachment
Appendix A - Technical Specifications

Date of Issuance: JUL 31 1970

1.1 Definitions

MODE (continued)

specified in Table 1.1-1 with fuel in the reactor vessel.

OPERABLE - OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

a. Described in Chapter 14, Initial Test Program of the Updated Final Safety Analysis Report (UFSAR);

b. Authorized under the provisions of 10 CFR 50.59; or

c. Otherwise approved by the Nuclear Regulatory Commission.

QUADRANT POWER TILT RATIO (QPTR)

QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

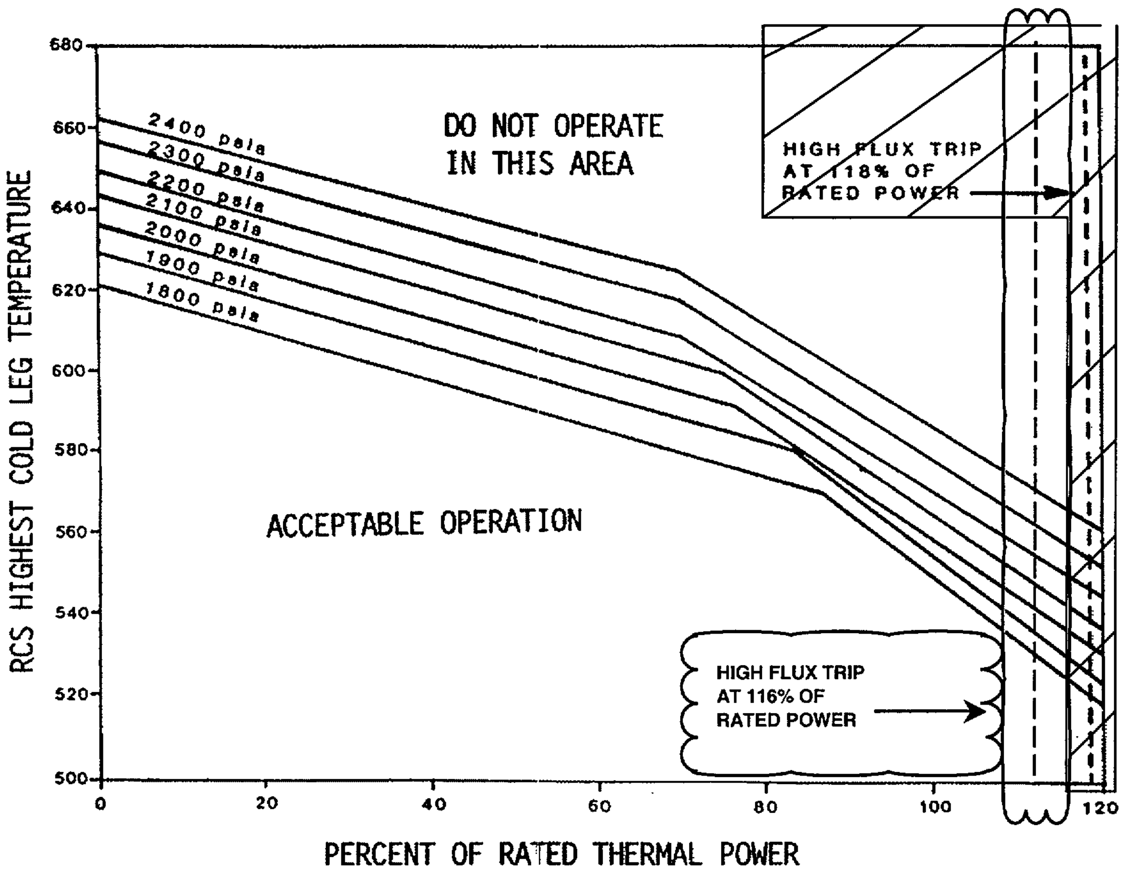
RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2300 MWt. 2339

SHUTDOWN MARGIN (SDM)

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

(continued)



NOTE: BASED ON A MINIMUM RCS FLOW OF 97.3 x 10^6 lbm/hr

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

Table 3.3.1-1 (page 6 of 7)
Reactor Protection System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Nominal Trip Setpoint by more than 2.96% of ΔT span.

$$\Delta T_{setpoint} \le \Delta T_O \left\{ K_1 - K_2 \frac{(1+\tau_1 S)}{(1+\tau_2 S)}(T - T') + K_3(P - P') - f(\Delta I) \right\}$$

Where: ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T' is the reference T_{avg} at RTP, ≤ ~~575.4°F.~~ 575.9°F

P is the measured pressurizer pressure, psig
P' is the nominal RCS operating pressure, ≤ 2235 psig

$K_1 \le 1.1265$ $K_2 = 0.01228/°F$ $K_3 = 0.00089/psig$
$\tau_1 \ge 20.08$ sec $\tau_2 \le 3.08$ sec

f(ΔI) =	$2.4\{(q_b - q_t) - 17\}$	when $q_t - q_b < -17\%$ RTP
	0% of RTP	when -17% RTP $\le q_t - q_b \le 12\%$ RTP
	$2.4\{(q_t - q_b) - 12\}$	when $q_t - q_b > 12\%$ RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

Table 3.3.1-1 (page 7 of 7)
Reactor Protection System Instrumentation

<u>Note 2: Overpower ΔT</u>

The Overpower ΔT Function Allowable Value shall not exceed the following Nominal Trip Setpoint by more than 3.17% of ΔT span.

$$\Delta T_{setpoint} \le \Delta T_0 \left\{ K_4 - K_5 \left[\frac{\tau_3 S}{1 + \tau_3 S} \right] T - K_6 (T - T') - f(\Delta I) \right\}$$

Where: ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T' is the reference T_{avg} at RTP, ≤ ~~575.4°F.~~ 575.9°F

$K_4 \le 1.06$ $K_5 \ge 0.02/°F$ for increasing T_{avg} $K_6 \ge 0.00277/°F$ when T > T'
0/°F for decreasing T_{avg} 0/°F when T ≤ T'
$\tau_3 \ge 9$ sec

f(ΔI) = as defined in Note 1 for Overtemperature ΔT

Table 3.3.2-1 (page 1 of 4)

Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT (1)
1. Safety Injection						
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.6	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
c. Containment Pressure - High	1,2,3,4	3	E	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ 4.45 psig	4 psig
d. Pressurizer Pressure - Low	1,2,3[(a)]	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 1709.89 psig	1715 psig
e. Steam Line High Differential Pressure Between Steam Header and Steam Lines	1,2,3[(a)]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	~~≤ 108.95 psig~~ ≥ 83.76 psig ≤ 116.24 psig	100 psig
f. High Steam Flow in Two Steam Lines	1,2[(b)],3[(b)]	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(c)	(d)
Coincident with T_{avg} - Low	1,2[(b)],3[(b)]	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 541.50 °F	543°F
g. High Steam Flow in Two Steam Lines	1,2[(b)],3[(b)]	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(c)	(d)
Coincident with Steam Line Pressure - Low	1,2[(b)],3[(b)]	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 605.05 psig	614 psig

(continued)

(1) A channel is OPERABLE with an actual Trip Setpoint value found outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint.

(a) Above the Pressurizer Pressure interlock.

(b) Above the T_{avg}-Low interlock.

(c) Less than or equal to a function defined as ΔP corresponding to 41.58% full steam flow below 20% load, and ΔP increasing linearly from 41.58% full steam flow at 20% load to 110.5% full steam flow at 100% load, and ΔP corresponding to 110.5% full steam flow above 100% load.

(d) A function defined as ΔP corresponding to 37.25% full steam flow between 0% and 20% load and then a ΔP increasing linearly from 37.25% steam flow at 20% load to 109% full steam flow at 100% load.

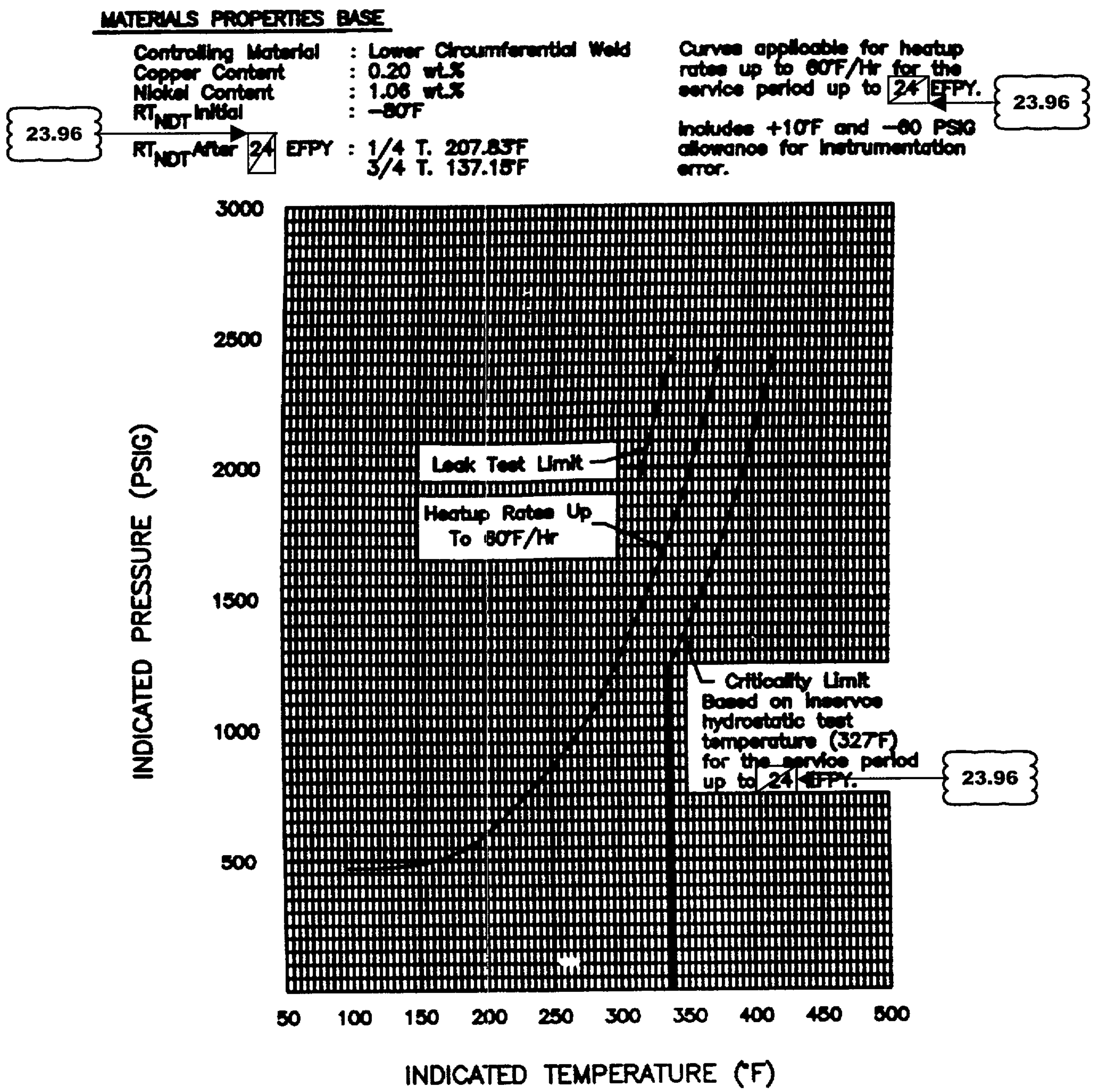


Figure 3.4.3-1
Reactor Coolant System ~~Heatup~~ Limits
Applicable Up to 24 EFPY

23.96

MATERIALS PROPERTIES BASE

Controlling Material	:	Lower Circumferential Weld
Copper Content	:	0.20 wt.%
Nickel Content	:	1.06 wt.%
RT_{NDT} Initial	:	−80°F
RT_{NDT} After ~~24~~ EFPY	:	1/4 T. 207.83°F 3/4 T. 137.15°F

23.96

Curves applicable for cooldown rates up to 100°F/Hr for the service period up to ~~24~~ EFPY.

23.96

Includes +10°F and −60 PSIG allowance for instrumentation error.

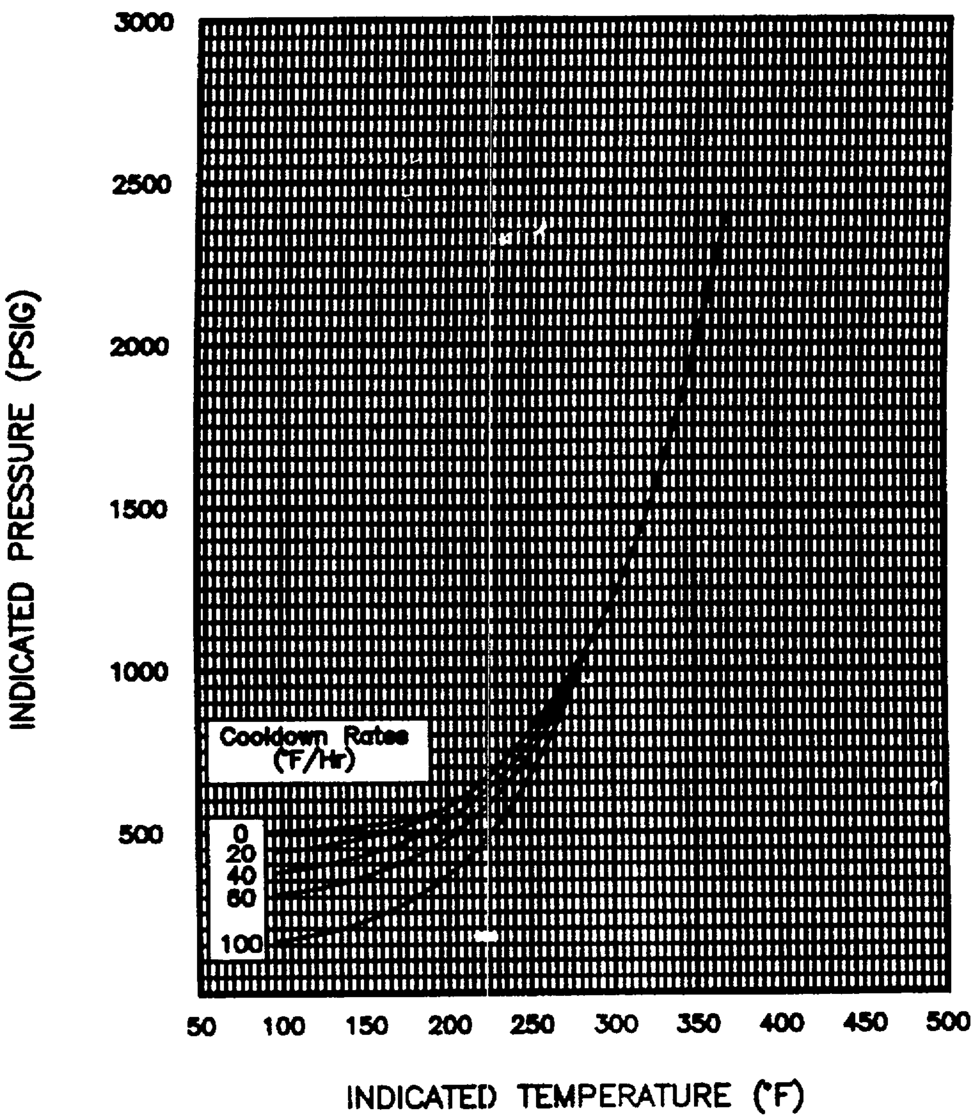


Figure 3.4.3-2
Reactor Coolant System Cooldown Limitations
Applicable Up to ~~24~~ EFPY

23.96

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 The MSSVs shall be OPERABLE as specified in Table 3.7.1-1 and Table 3.7.1-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONs

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more steam generators with one MSSV inoperable and the Moderator Temperature Coefficient (MTC) zero or negative at all power levels.	A.1 Reduce THERMAL POWER to < ~~51~~ % RTP. 50	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more bypass valves inoperable.	C.1 Close or isolate bypass valve. AND C.2 Verify bypass valve is closed or isolated.	8 hours Once per 7 days
D Two valves in the same flow path inoperable.	D.1 Isolate affected flow path.	8 hours
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the closure time of each MFRV and bypass valve is ≤ ~~30~~ 20 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program
SR 3.7.3.2	Verify the closure time of each MFIV is ≤ 50 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program

United States Nuclear Regulatory Commission
Attachment IV to Serial: RNP-RA/02-0066
12 Pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
REQUEST FOR TECHNICAL SPECIFICATIONS
<u>CHANGES TO INCREASE AUTHORIZED REACTOR POWER LEVEL</u>

RETYPED TECHNICAL SPECIFICATIONS PAGES

3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20. Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Section 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; 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 below:

   A. Maximum Power Level

      The licensee is authorized to operate the facility at a steady state reactor core power level not in excess of 2339 megawatts thermal.

   B. Technical Specifications

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. are hereby incorporated in the license.

      The licensee shall operate the facility in accordance with the Technical Specifications.

      (1) For Surveillance Requirements (SRs) that are new in Amendment 176 to Final Operating License DPR-23, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 176. For SRs that existed prior to Amendment 176, including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 176.

   C. Reports

      Carolina Power & Light Company shall make certain reports in accordance with the requirements of the Technical Specifications.

   D. Records

      Carolina Power & Light Company shall keep facility operating records in accordance with the requirements of the Technical Specifications.

4. <u>Additional Conditions</u>

   The Additional Conditions contained in Appendix B, as revised through Amendment No.   , are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the additional conditions.

5. This license is effective as of the date of issuance and shall expire at midnight July 31, 2010.

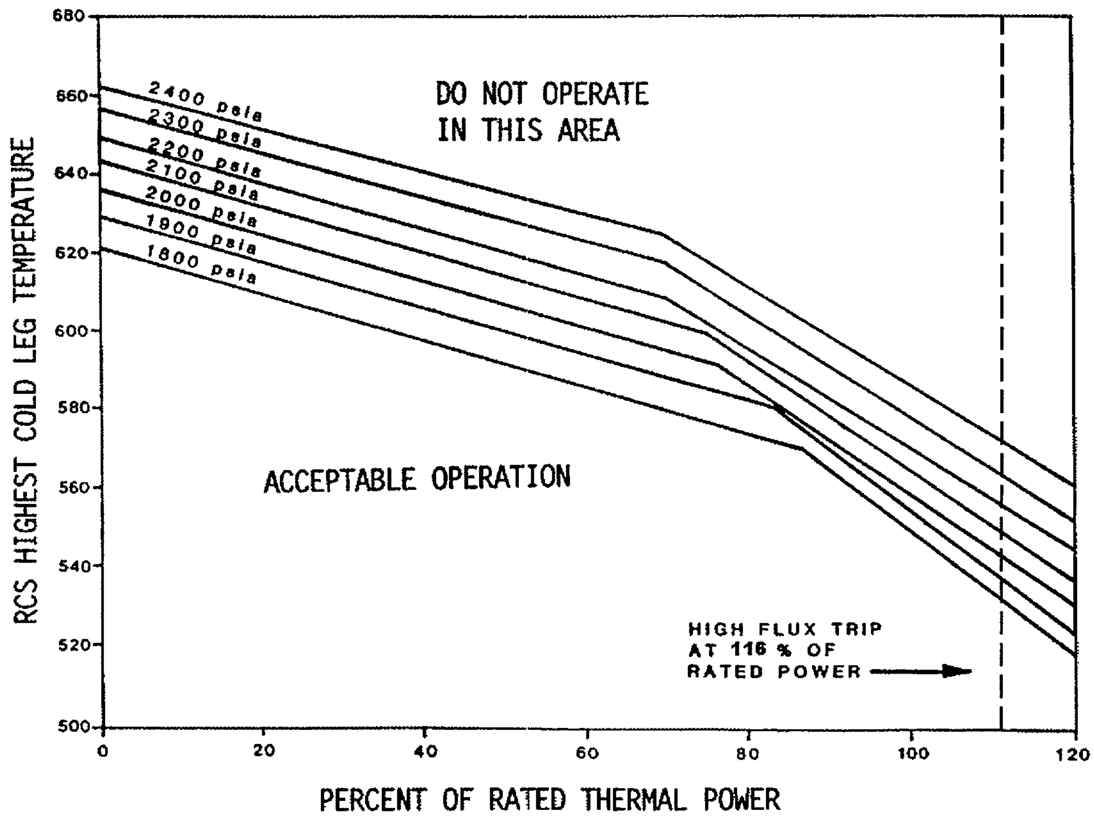
Attachment
Appendix A - Technical Specifications

Date of Issuance: JUL 31 1970

1.1 Definitions

MODE (continued)	specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE - OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program of the Updated Final Safety Analysis Report (UFSAR); b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2339 MWt.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

(continued)



NOTE: BASED ON A MINIMUM RCS FLOW OF 97.3 x 10^6 lbm/hr

Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

Table 3.3.1-1 (page 6 of 7)
Reactor Protection System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following Nominal Trip Setpoint by more than 2.96% of ΔT span.

$$\Delta T_{setpoint} \leq \Delta T_O \left\{ K_1 - K_2 \frac{(1+\tau_1 S)}{(1+\tau_2 S)}(T - T') + K_3(P - P') - f(\Delta I) \right\}$$

Where: ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T' is the reference T_{avg} at RTP, ≤ 575.9°F.

P is the measured pressurizer pressure, psig
P' is the nominal RCS operating pressure, ≤ 2235 psig

$K_1 \leq 1.1265$ $K_2 = 0.01228/°F$ $K_3 = 0.00089/psig$
$\tau_1 \geq 20.08$ sec $\tau_2 \leq 3.08$ sec

$f(\Delta I) =$	$2.4\{(q_b - q_t) - 17\}$	when $q_t - q_b < -17\%$ RTP
	0% of RTP	when -17% RTP $\leq q_t - q_b \leq 12\%$ RTP
	$2.4\{(q_t - q_b) - 12\}$	when $q_t - q_b > 12\%$ RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

Table 3.3.1-1 (page 7 of 7)
Reactor Protection System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Nominal Trip Setpoint by more than 3.17% of ΔT span.

$$\Delta T_{setpoint} \leq \Delta T_0 \left\{ K_4 - K_5 \left[\frac{\tau_3 S}{1 + \tau_3 S} \right] T - K_6 (T - T') - f(\Delta I) \right\}$$

Where: ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T' is the reference T_{avg} at RTP, $\leq$ 575.9°F.

$K_4 \leq 1.06$

$K_5 \geq 0.02/°F$ for increasing T_{avg}
0/°F for decreasing T_{avg}

$K_6 \geq 0.00277/°F$ when $T > T'$
0/°F when $T \leq T'$

$\tau_3 \geq 9$ sec

$f(\Delta I)$ = as defined in Note 1 for Overtemperature ΔT

Table 3.3.2-1 (page 1 of 4)

Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT (1)
1. Safety Injection						
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.6	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
c. Containment Pressure - High	1,2,3,4	3	E	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ 4.45 psig	4 psig
d. Pressurizer Pressure - Low	1,2,3[(a)]	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 1709.89 psig	1715 psig
e. Steam Line High Differential Pressure Between Steam Header and Steam Lines	1,2,3[(a)]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 83.76 psig ≤ 116.24 psig	100 psig
f. High Steam Flow in Two Steam Lines	1,2[(b)],3[(b)]	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(c)	(d)
Coincident with T_{avg} - Low	1,2[(b)],3[(b)]	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 541.50 °F	543°F
g. High Steam Flow in Two Steam Lines	1,2[(b)],3[(b)]	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(c)	(d)
Coincident with Steam Line Pressure - Low	1,2[(b)],3[(b)]	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 605.05 psig	614 psig

(continued)

(1) A channel is OPERABLE with an actual Trip Setpoint value found outside its calibration tolerance band provided the Trip Setpoint value is conservative with respect to its associated Allowable Value and the channel is re-adjusted to within the established calibration tolerance band of the Nominal Trip Setpoint.
(a) Above the Pressurizer Pressure interlock.
(b) Above the T_{avg}-Low interlock.
(c) Less than or equal to a function defined as ΔP corresponding to 41.58% full steam flow below 20% load, and ΔP increasing linearly from 41.58% full steam flow at 20% load to 110.5% full steam flow at 100% load, and ΔP corresponding to 110.5% full steam flow above 100% load.
(d) A function defined as ΔP corresponding to 37.25% full steam flow between 0% and 20% load and then a ΔP increasing linearly from 37.25% steam flow at 20% load to 109% full steam flow at 100% load.

MATERIALS PROPERTIES BASE

Controlling Material	: Lower Circumferential Weld	Curves applicable for heatup rates up to 80°F/Hr for the service period up to 23.96 EFPY.
Copper Content	: 0.20 wt.%	
Nickel Content	: 1.06 wt.%	
RT_{NDT} Initial	: −80°F	Includes +10°F and −60 PSIG allowance for instrumentation error.
RT_{NDT} After 23.96 EFPY	: 1/4 T. 207.83°F 3/4 T. 137.18°F	

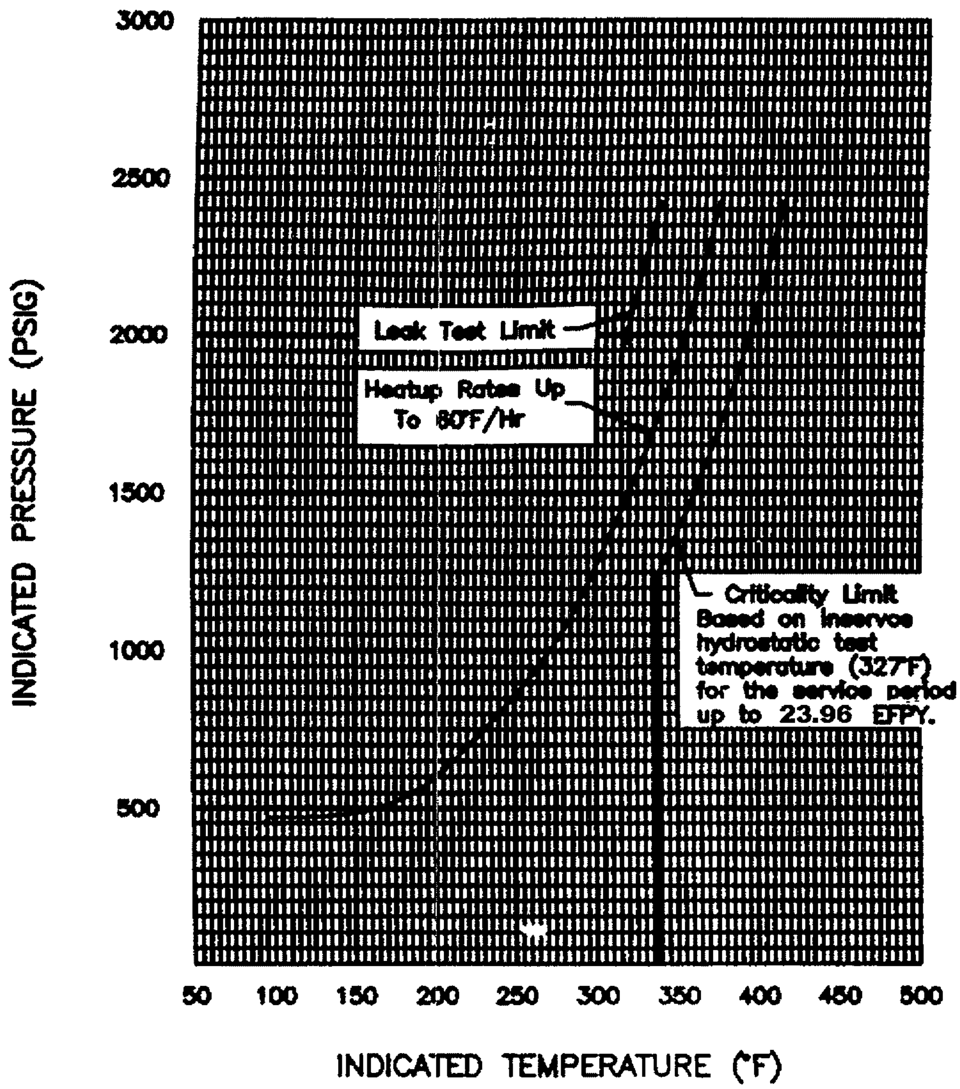


Figure 3.4.3-1
Reactor Coolant System Heatup Limits
Applicable Up to 23.96 EFPY

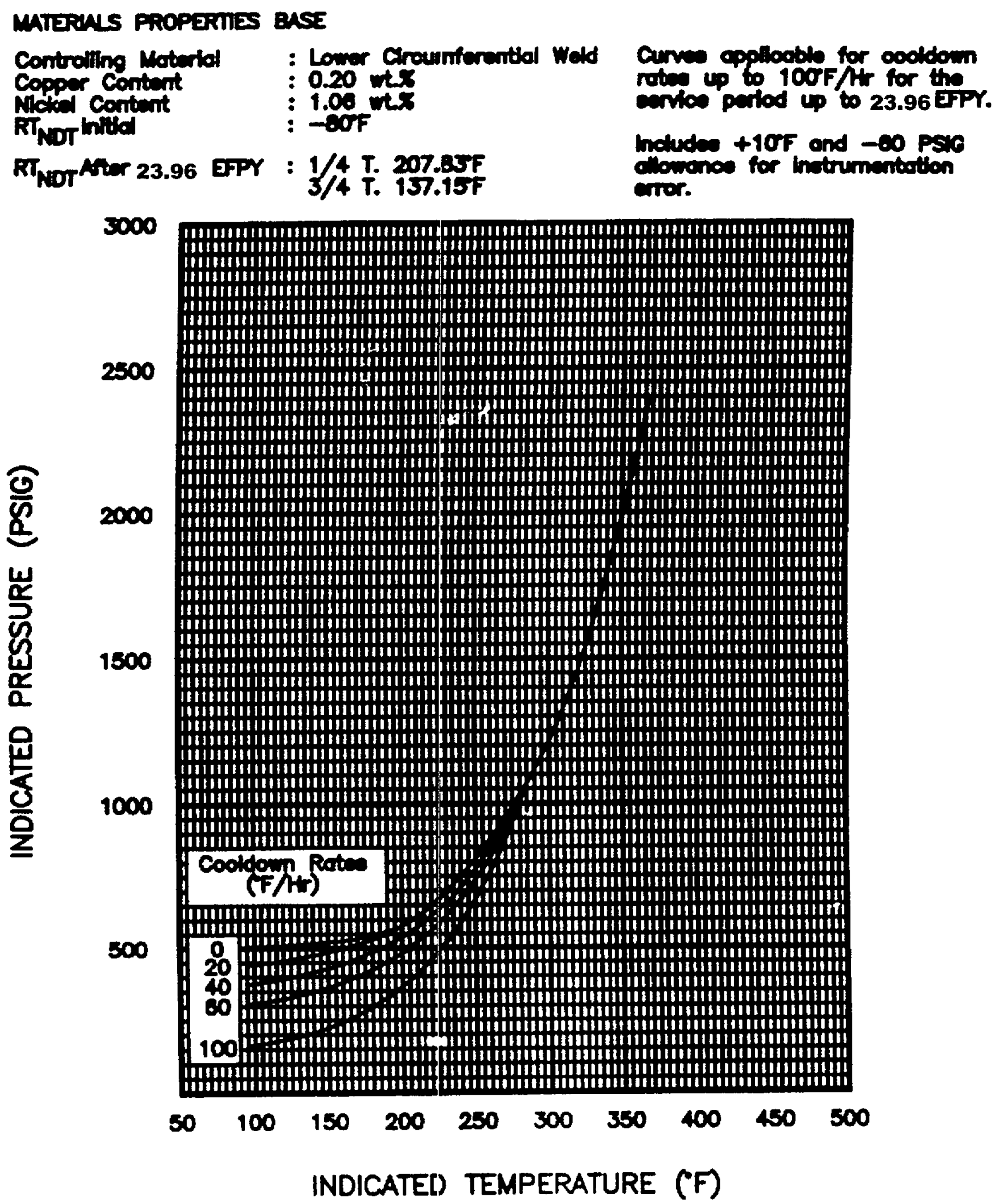


Figure 3.4.3-2
Reactor Coolant System Cooldown Limitations
Applicable Up to 23.96 EFPY

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 The MSSVs shall be OPERABLE as specified in Table 3.7.1-1 and Table 3.7.1-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more steam generators with one MSSV inoperable and the Moderator Temperature Coefficient (MTC) zero or negative at all power levels.	A.1 Reduce THERMAL POWER to < 50 % RTP.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more bypass valves inoperable.	C.1 Close or isolate bypass valve. <u>AND</u> C.2 Verify bypass valve is closed or isolated.	8 hours Once per 7 days
D Two valves in the same flow path inoperable.	D.1 Isolate affected flow path.	8 hours
E. Required Action and associated Completion Time not met.	E.1 Be in MODE 3. <u>AND</u> E.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.3.1	Verify the closure time of each MFRV and bypass valve is $\leq$ 20 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program
SR 3.7.3.2	Verify the closure time of each MFIV is $\leq$ 50 seconds on an actual or simulated actuation signal.	In accordance with the Inservice Testing Program

United States Nuclear Regulatory Commission
Attachment V to Serial: RNP-RA/02-0066
6 Pages

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
REQUEST FOR TECHNICAL SPECIFICATIONS
CHANGES TO INCREASE AUTHORIZED REACTOR POWER LEVEL

REGULATORY ISSUE SUMMARY 2002-03 CROSS-REFERENCE

REGULATORY ISSUE SUMMARY 2002-03 CROSS-REFERENCE

This attachment provides a cross-reference listing of the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, responses provided in Attachment II of this submittal to the topics listed in Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications."

I. Feedwater flow measurement technique and power measurement uncertainty

A detailed discussion of the plant-specific feedwater flow measurement technique and measurement uncertainty is provided in Section 3.2. The discussion in Section 3.2 also describes the expected power increase made possible as a result of implementing this improved flow measurement technique. Calculation of the total power measurement uncertainty is discussed in Section 3.2, and Table 3.2-1.

References to safety evaluations documenting NRC approval of the vendor topical reports associated with this feedwater flow measurement technique are provided in Section 3.2.1. Section 3.2.1 also discusses compliance with the guidelines provided in these topical reports. Disposition of the four NRC-identified criteria related to use of these vendor topical reports is addressed in Sections 3.2.1.1 through 3.2.1.4.

Maintenance and calibration of the feedwater flow instrumentation is discussed in Section 3.2.1.1. Control of software and hardware configurations, corrective actions, reporting of deficiencies to the manufacturer, and receiving and addressing manufacturer deficiency reports are also discussed in Section 3.2.1.1. Additionally, Section 3.2.1.1 provides a description of the proposed allowed outage time for the feedwater flow measurement instrumentation, and actions to reduce power if the allowed outage time is exceeded.

II. Accidents and transients for which the existing analyses of record bound plant operation at the proposed uprated power level

A discussion of the transients, accidents, and events described in the HBRSEP, Unit No.2, Updated Final Safety Analysis Report (UFSAR) is provided in Sections 3.10.3 through 3.10.8, and Sections 4.3 and 4.4. Section 3.10.3 describes the transients and accidents described in Chapter 15 of the UFSAR. Radiological consequences of transients and accidents are discussed in Section 3.10.4. Containment performance, as described in Chapter 6 of the UFSAR, is described in Section 3.10.5. Anticipated Transient Without Scram (ATWS), Environmental Qualification (EQ), and Flooding are discussed in Sections 3.10.6 through 3.10.8. Station Blackout and Appendix R are discussed in Sections 4.3 and 4.4.

For the UFSAR Chapter 6 and Chapter 15 events, the descriptions provided in Attachment II for these transients and accidents indicate whether the proposed power uprate will continue to be bounded by the existing analyses of record. Additionally, statements are provided for these events specifying whether the analyses of record have either been previously approved by the NRC or were conducted using methods or processes that were previously approved by the NRC, and confirming that bounding event determinations continue to be valid. References to the analysis methodologies utilized and prior NRC approvals are provided, where applicable.

For events and conditions that are not described in UFSAR Chapter 6 or Chapter 15 (i.e., ATWS, Appendix R, Station Blackout, and Flooding), evaluation of the continued ability to satisfy design requirements is discussed in Sections 3.10.6 through 3.10.8, and Sections 4.4 and 4.5.

III. Accidents and transients for which the existing analyses of record do not bound plant operation at the proposed uprated power level

The response to item II remains applicable for this item. For the UFSAR Chapter 6 and Chapter 15 transient and accident events, a discussion is provided in the summary descriptions of Sections 3.10.3 through 3.10.5 indicating whether re-analysis was required to support the power uprate, and any applicable changes to plant parameters or analysis methodologies. References to the analysis methodologies utilized and NRC approval of these methodologies are provided, where applicable.

For events and conditions that are not contained in UFSAR Chapter 6 or Chapter 15, evaluation of their continued ability to satisfy design requirements is described in Sections 3.10.6 through 3.10.8, and Sections 4.4 and 4.5.

IV. Mechanical/Structural/Material Component Integrity and Design

The effect of the power uprate on the structural integrity of major plant components is discussed in Section 3.6 (including subsections). Section 3.6 addresses the structural integrity of NSSS components, and includes a discussion of power uprate effects on loss-of-coolant (LOCA) forces; line break loadings; Reactor Coolant System (RCS) component stress and fatigue; reactor vessel internals and supports; RCS attached piping and supports; control rod drive mechanisms; the reactor vessel head; the pressurizer shell, nozzle, and spray and surge lines; reactor coolant pumps; steam generators; and fuel assemblies.

The steam generator discussion provided in Section 3.6.2.10 includes consideration of tube wear resulting from flow induced vibration, changes in fluid-elastic stability, and gap velocity.

The effect of the power uprate on Low Temperature Overpressure Protection analyses is described in Section 3.5.7. Changes to the 10 CFR 50, Appendix G, curves and reference temperature values, and the Technical Specification (TS) pressure-temperature limit curves for heatup and cooldown due to increased fluence, are discussed in Section 3.6.2.1. The effect of the power uprate on the surveillance capsule withdrawal schedule provided in the Reactor Vessel Radiation Surveillance Program is discussed in Section 3.6.2.1.

The effect of the power uprate on safety-related valves is discussed in Sections 3.7.1 (subsections 3.7.1.1 through 3.7.1.5), 3.7.2 (subsections 3.7.2.1 through 3.7.2.3), 4.5 (including subsections), and 4.6.2. Section 3.7.1 and 3.7.2 discuss the effect of the power uprate on major safety-related valves in Nuclear Steam Supply System/Balance-of-Plant (NSSS/BOP) interface systems. Section 4.5 describes the effect of the power uprate on plant programs for safety-related valves. Section 4.6.2 describes the effect of the power uprate on the plant Inservice Testing Program.

Power uprate related changes to the Flow Accelerated Corrosion Program are discussed in Section 4.6.5.

V. Electrical Equipment Design

The effect of the power uprate on electrical equipment design is described in Sections 3.9, 3.10.7, and 4.3. Section 3.9 discusses the ability of plant electrical equipment, including the main generator and auxiliaries, transformers, emergency diesel generators, electrical busses, and medium and low voltage systems, to satisfy design requirements under power uprate conditions. A discussion of grid stability is also provided in Section 3.9. The discussions in Section 3.9 include identification of relevant changes to plant parameters and equipment performance related to the power uprate.

The effect of the power uprate on the environmental qualification of electrical equipment is discussed in Section 3.10.7. The effect of the power uprate on Station Blackout equipment is addressed in Section 4.3

VI. System Design

Section 3.7 (including subsections) addresses the effect of the power uprate on BOP systems that interface with the NSSS, including the Main Steam Safety Valves, Main Steam Isolation Valves, Main Steam Isolation Bypass Valves, Main Steam Check Valves, and Main Steam Dump Valves. These discussions describe the effects of changes in temperature, pressure, and flow resulting from the power uprate.

Section 3.10.5 discusses the effect of the power uprate on containment performance. Sections 3.7.5 through 3.7.7 discuss the effect of the power uprate on safety-related cooling water systems: Component Cooling Water, Service Water, and Ultimate Heat Sink. Radioactive Waste Management Systems are discussed in Section 4.8. The effect of the power uprate on heating, ventilation, and air conditioning systems is discussed in Section 4.10 (including subsections).

VII. Other

A discussion of the effect of the power uprate on operator actions is provided in Section 4.2.1. Additionally, operator action times related to the plant Appendix R Program are discussed in Section 4.4. The effect of the power uprate on emergency and abnormal procedures is also discussed in Section 4.2.1. Section 4.1.1 discusses power uprate impacts to Control Room controls and displays, and Sections 4.1.2 and 4.2.2 discuss impacts to the control room plant reference simulator. The effect of the power uprate on the operator training program is discussed in Section 4.2.2. These sections also provide statements regarding scheduled completion of necessary changes to emergency and abnormal operating procedures, Control Room controls and displays, and the plant simulator resulting from the power uprate. Additionally, Section 4.2.1 describes intended actions related to temporary operation above steady-state licensed power levels to reduce the magnitude of the allowed deviation from the licensed power level.

Section 6.2 provides a discussion of the 10 CFR 51.22 criteria for categorical exclusion for environmental review. Sections 3.10.4 and 4.9 describe onsite and offsite radiation exposures from normal operation and accident events. The effect of the power uprate on effluents that may be released offsite, the Final Environmental Statement, and previous environmental assessments for the plant are discussed in Sections 6.0 and 6.1.

VIII. Changes to technical specifications, protection system settings, and emergency system settings

Changes to Reactor Protection System (RPS) and Engineered Safeguard Features Actuation System (ESFAS) protection system settings resulting from the power uprate are discussed in Sections 3.10.1.1 and 3.10.1.2. These discussions include a description of changes in plant parameters and analysis assumptions for the RPS and ESFAS protective features, and identification of necessary TS changes to Nominal Trip Setpoints and Allowable Values.

The basis for the TS changes requested in this submittal are described in the following locations:

- The basis for changes to the authorized power level stated in the Operating License, and reflected in the TS 1.1 definition for Rated Thermal Power, is provided in Sections 1.0 and 2.0, and is supported by the information provided in Attachment II.

- The basis for the TS 2.1.1 change to revise the High Flux Trip shown on TS Figure 2.1.1-1 is discussed in Section 3.10.1.1 for the Power Range Neutron Flux Trip.

- The basis for the TS 3.3.1 changes to revise the value of T_{avg} used to calculate the Overtemperature ΔT (OTDT) and Overpressure ΔT (OPDT) RPS Trips is described in the Section 3.10.1.1 discussions of these trip functions.

- The basis for the TS 3.3.2 change to revise the Allowable Value for the "Steam Line High Differential Pressure Between Steam Header and Steam Lines," function is described in the Section 3.10.1.2 discussions of this protective function.

- The basis for the TS 3.4.3 change to revise the RCS heatup and cooldown curves is described in the Section 3.6.2.1.

- The basis for the TS 3.7.1 change to revise the power level associated with the Required Action for an inoperable Main Steam Safety Valve (MSSV) is described in the Section 3.7.1.1.

- The basis for the TS 3.7.1 change to reduce the stroke time for the Main Feedwater Regulation Valves (MFRVs) and Bypass Valves is described in the Sections 3.7.2.1 and 3.7.2.2.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
REQUEST FOR TECHNICAL SPECIFICATIONS
CHANGES TO INCREASE AUTHORIZED REACTOR POWER LEVEL

REGULATORY COMMITMENT SUMMARY

Listing of Regulatory Commitments

The actions committed to by CP&L in this letter are identified below. Any other statements in this submittal are provided for information and are not considered regulatory commitments.

Commitment	Scheduled Completion Date
1. Plant maintenance and calibration procedures will be revised to incorporate Caldon maintenance and calibration requirements.	Prior to implementing the power uprate.
2. Operability requirements for the LEFM system will be incorporated into the HBRSEP, Unit No. 2, Technical Requirements Manual (TRM).	Prior to implementing the power uprate.
3. For the feedwater heater components that were evaluated to exceed HEI guidelines, but that have not been recently measured and inspected, measurement and inspection of these feedwater heater components will be performed.	Prior to implementing the power uprate.
4. The shell-side feedwater heater relief valves will be replaced with relief valves that meet HEI guidelines.	Prior to implementing the power uprate.
5. Normal and Abnormal Operating Procedures, Emergency Operating Procedures, and Off-Normal Procedures will be revised to reflect the power uprate.	Prior to use at the uprated power conditions.
6. Plant operating procedures will be revised to indicate that temporary operation above the licensed power level shall be limited to 0.3 percent of RTP, consistent with the reduced power measurement uncertainty.	Prior to implementing the power uprate.
7. Simulator training will be provided on power uprate-related changes to the plant that affect operator performance.	Prior to implementing the power uprate.
8. Training will be provided on power uprate-related changes to plant procedures that affect operator performance.	Prior to their use at the uprated power conditions
9. Complete hardware and software changes to the simulator, including changes involving plant process computer inputs, which affect operator performance.	Prior to implementing the power uprate.