December 26, 2007

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing & Regulatory Programs
15760 W. Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT REGARDING MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NO. MD5500)

Dear Mr. Young:

The Commission has issued the enclosed Amendment No. 228 to Facility Operating License No. DPR-72 for Crystal River Unit 3 in response to your letter dated April 25, 2007, as supplemented by letters dated June 28, August 30, September 13, October 18, and November 1, 2007.

The amendment increases the licensed core power level 1.6 percent to 2609 megawatts thermal. This increase will be achieved by the use of high-accuracy heat balanced instrumentation, including a Caldon Leading Edge Flowmeter CheckPlus™ ultrasonic flow measurement system, which allows more accurate measurement of feedwater flow.

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

Sincerely,

/RA/

Stewart N. Bailey, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures:
1. Amendment No. 228 to DPR-72
2. Safety Evaluation

cc w/enclosures: See next page

December 26, 2007

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ATTN: Supervisor, Licensing & Regulatory Programs
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NAME	JMcHale	AHiser	MMitchel	TChan	GWilson
DATE	9/06/2007*	11/09/2007*	12/14/2007*	12/21/07	11/28/2007*
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NAME	WKemper	KManoly	NSalgado	TKobetz	MHart
DATE	12/12/2007*	7/16/2007*	10/18/2007*	12/20/07	8/20/2007*
OFFICE	AFPB/BC	SNPB/BC	SRXB/BC	SRXB/BC	SBPB/BC
NAME	AKlein	AMendiola	GCranston	GCranston	DHarrison
DATE	10/04/2007*	9/27/2007*	12/13/2007*	12/20/2007*	12/20/07
OFFICE	LPL2-2/PM	LPL2-2/LA	OGC	LPL2-2/BC	DORL/D
NAME	SBailey	CSola for BClayton	E.Williamson	LRaghavan for TBoyce	CHaney

DATE	12/26/07	12/20/07	12/21/07	12/26/07	12/26/07

*by memo

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Letter to Dale E. Young from Stewart N. Bailey dated December 26, 2007

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT REGARDING MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NO. MD5500)

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FLORIDA POWER CORPORATION

CITY OF ALACHUA

CITY OF BUSHNELL

CITY OF GAINESVILLE

CITY OF KISSIMMEE

CITY OF LEESBURG

CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION

CITY OF NEW SMYRNA BEACH

CITY OF OCALA

ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO

SEMINOLE ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 228
License No. DPR-72

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

   A. The application for amendment by Florida Power Corporation, et al. (the licensees), dated April 25, 2007, as supplemented by letters dated June 28, August 30, September 13, October 18, and November 1, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-72 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 228, are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: December 26, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 228

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following page 4 of Facility Operating License DPR-72 with the attached revised page 4

Remove	Insert
4	4

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

Remove	Insert
1.1-4	1.1-4
1.1-6	1.1-6
3.3-1	3.3-1
3.3-2	3.3-2
3.3-3	3.3-3
3.3-5	3.3-5

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 228 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

1.0 INTRODUCTION

By letter dated April 25, 2007, as supplemented by letters dated June 28, August 30, September 13, October 18, and November 1, 2007, the Florida Power Corporation (FPC, or the licensee) submitted License Amendment Request (LAR) No. 296, requesting an increase in the licensed thermal power level for Crystal River Unit 3 (CR-3).

The amendment would increase the licensed core power level by 1.6 percent from 2568 megawatts thermal (MWt) to 2609 MWt. This increase will be achieved by the use of high-accuracy heat balanced instrumentation including a Caldon Leading Edge Flowmeter (LEFM) CheckPlus™ ultrasonic flow measurement (UFM) system, which allows more accurate measurement of feedwater (FW) flow rate. This type of application is commonly referred to as a measurement uncertainty recapture (MUR) power uprate. The licensee developed the LAR following the guidance of Nuclear Regulatory Commission (NRC) Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications."

The June 28, 2007, letter provided Revision 1 to LAR No. 296, which replaced Revision 0 in its entirety. The supplements dated September 13, October 18, and November 1, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed (e.g., Revision 1), and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on September 11, 2007 (72 FR 51862).

2.0 BACKGROUND

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

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

In LAR 296, the licensee proposed to use a power measurement uncertainty of 0.4 percent of RTP. To achieve this level of accuracy, the licensee will install a Caldon LEFM CheckPlus™ UFM system for measuring the main FW flow rate and temperature. The Caldon system provides a more accurate measurement of FW flow than what was assumed during the development of the original 10 CFR Part 50, Appendix K requirements. The Caldon system will measure FW mass flow to within 0.34 percent for CR-3. This bounding FW mass flow uncertainty supports a total power measurement uncertainty of 0.4 percent of RTP. On the basis of this, the licensee proposed to increase the reactor power level by 1.6 percent.

In large part, the basis for acceptability of a proposed MUR power uprate that the uprated conditions are bounded by the current analyses of record. Historically, the majority of analyses were performed assuming 102 percent core power. Therefore, the analyzed power level, including uncertainty, does not change for the MUR power uprate. The exceptions to this are reviewed in detail by the NRC staff. RIS 2002-03 recommends that, to improve efficiency of the staff's review, licensees requesting an MUR power uprate should identify existing design basis accident (DBA) analyses of record which bound plant operation at the proposed uprated power level. For any existing DBA analyses that do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the reanalysis.

3.0 EVALUATION

The licensee stated that for the Babcock and Wilcox (B&W)-designed plants, the power measurement uncertainty is accounted for in the DBA analyses (e.g., the analyses were performed at 102 percent of RTP). The power uncertainty is also included in the determination of setpoints that are based on power level. The licensee also stated that the majority of the mechanical system design is based on generic, bounding B&W evaluations that were performed using a power level of 2772 MWt (108 percent of RTP). Therefore, the majority of the plant design remains bounded by the current analyses of record.

The licensee stated that it reviewed the CR-3 Final Safety Analysis Report (FSAR) and other design basis analyses to verify that there was no impact from the MUR power uprate. If an analysis was not bounding for the MUR conditions, the licensee either revised the analysis using NRC-approved methods, or determined that the analysis was still acceptable.

In its review, the NRC staff focused on the power measurement uncertainty that is the basis for the MUR power uprate, the reasonableness of the licensee's determination that the MUR remains bounding, and the licensee's justification for continued reliance on any analyses that are not performed at a bounding power level. The NRC staff also reviewed the acceptability of the proposed changes to the Technical Specifications (TSs) and Facility Operating License (FOL).

3.1 FW Flow Measurement Technique and Power Measurement Uncertainty

Core power level is determined by an automatic or manual calculation of the energy balance around the plant's nuclear steam supply system (NSSS). The licensee performs a calculation called a "secondary calorimetric" to determine the energy removed by the steam generators (SGs), and then corrects for reactor coolant pump (RCP) heat addition, reactor coolant system (RCS) heat loss, and other factors to determine core power level. The accuracy of the secondary calorimetric depends primarily upon the accuracy of FW flow rate. Section I of Attachment 1 to RIS 2003-02 provides guidance on the information that should be submitted to support a licensee's determination of power measurement uncertainty. This information is reviewed by the NRC staff's Instrumentation and Controls (I&C) Branch, Reactor Systems Branch, and Nuclear Performance and Code Review Branch.

CR-3 currently uses flow nozzles to measure FW flow rate and resistance temperature detectors (RTDs) to measure temperature. The CheckPlus™ UFM system uses the transit time methodology. Ultrasonic pulses transmitted into a fluid stream travel faster in the direction of the fluid flow than opposite the flow. The difference in the upstream and downstream traversing times of the ultrasonic pulses is proportional to the fluid velocity in the pipe. Temperature is determined from the mean propagation times using a pre established correlation with fluid pressure. The mean fluid density is obtained from the measured pressure and the derived mean fluid temperature.

The licensee determined the UFM accuracy in accordance with Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check™ System," and ER-157P, "Basis for a Power Uprate With the LEFM Check™ or LEFM CheckPlus™ System." In a safety evaluation (SE) dated March 8, 1999 (Agencywide Documents and Management System (ADAMS) Accession No. 9903190065, which is in the legacy library), the staff approved ER-80P use in MUR power uprates up to 1 percent. In an SE dated December 20, 2001 (ML013540256), the NRC staff approved ER-157P for use in MUR power uprates up to 1.7 percent. The licensee also referred to a letter from Mr. Thomas (NRC) to Mr. Hauser (Caldon), "Evaluation of the Hydraulic Aspects of the Caldon [LEFM] Check and CheckPlus™ [UFMs]," dated July 5, 2006 (ML061700222), which addressed the hydraulic aspects of the Caldon UFMs in response to industry operating experience. These documents collectively provide the generic acceptability of the Caldon CheckPlus™ system. The CR-3 specific secondary calorimetric uncertainty is provided in Cameron (formerly Caldon) Engineering Report ER 579, "Bounding Uncertainty Analysis for Thermal Power Determination at Crystal River Unit 3 Using LEFM [Checkplus™] System." The licensee provided this report by letter dated August 30, 2007.

The LEFM Check™ system, as described in Topical Report ER-80P, consists of a spool piece with eight transducer assemblies forming the four chordal acoustic paths in one plane of the spool piece. The system includes an electronics unit with hardware and software installed to provide flow and temperature measurements and an on-line verification of these measurements. An LEFM CheckPlus™ system, both hydraulically and electronically, is made up of two LEFM Check™ systems in a single spool piece. This layout has two sets of four chordal acoustic paths in two perpendicular planes of the spool piece. The electronics for the two subsystems, while electrically separated, are housed in a single cabinet. To ensure independence, the two measurement planes of an LEFM CheckPlus™ system have independent clocks for measuring transit times of the ultrasound pulses.

The UFM system at CR-3 will consist of one CheckPlus™ measurement section/spool piece in each of the two 18 inch main FW lines. These spool pieces and the electronic cabinet will be located in the CR-3 intermediate building. Each measurement section consists of sixteen (16) ultrasonic transducer housings, as described above. Each transducer can be removed at full power conditions without disturbing the pressure boundary.

The licensee stated that the CheckPlus™ system software was developed, and will be maintained, under a verification and validation (V&V) program. The V&V program has been applied to all system software and hardware, and includes a detailed code review. The FW mass flow rate and FW temperature are displayed on the electronic cabinet and transmitted via Ethernet to the automated unit load demand (AULD) and plant process computer (CP) for use in the secondary calorimetric. The AULD and the plant process computer independently perform secondary heat balance calculations. The electronic cabinet has outputs for internally generated system trouble alarms, which will be wired into the plant process computer. The AULD-calculated heat balance is used in conjunction with the integrated control system to automatically control plant power at the operator-selected core thermal power. The CP-calculated heat balance is normally used by the plant operators to calibrate the nuclear instrumentation (NI) and can be used by the plant operators to manually control reactor power upon loss of AULD. These two software routines are independent but receive identical inputs.

The LEFM indications of FW mass flow and temperature will be directly substituted for the FW flow nozzle and RTD inputs that are currently used in the plant calorimetric calculations. The existing FW flow and temperature measurements will continue to be used for FW control and other functions that they currently fulfill.

The NRC staff reviewed the proposed plant-specific implementation of the FW flow measurement technique and the power increase gained as a result of implementing this technique. The review was performed in accordance with the guidelines A through H in Section I of Attachment 1 to RIS 2002-03. The staff review confirmed that the licensee's implementation of the proposed FW flow measurement device was consistent with the staff-approved Caldon Topical Reports ER-80P and ER-157P and adequately addressed the four additional requirements listed in the staff's approval of those reports. The NRC staff also reviewed the power uncertainty calculations to ensure that (1) the proposed uncertainty value of 0.4 percent correctly accounted for all uncertainties due to power level instrumentation errors, and (2) the calculations met the relevant requirements of 10 CFR Part 50 Appendix K, as described in Section 2.0 of this SE.

The staff SE on Caldon Topical Report ER-80P included four additional criteria to be addressed by a licensee referencing this topical report for power uprate. The licensee addressed each of the four criteria as follows:

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

In response, the licensee indicated that it will develop the necessary procedures and documents required for operation, maintenance, calibration, testing, and training as part of implementing the modification. The licensee stated that a preventative maintenance program will be developed using the vendor's maintenance and troubleshooting manual, and that vendor personnel will be present to oversee the commissioning of the system and will be present to help resolve problems or failures during startup from Refueling Outage 15. The vendor will also provide training to licensee personnel on the theory, components, software, and troubleshooting.

The licensee's letter dated November 1, 2007, describes a preventative maintenance program that includes a number of periodic inspections and activities. The program includes periodic checks of the analog inputs, clock speed checks, cleaning activities, and component tests (including relay checks). In addition, the LEFM features continuous monitoring and self-assessment. The staff reviewed the elements of the licensee's maintenance program and concludes that the program, along with the continuous monitoring of the LEFM, ensures that the LEFM will remain bounded by the analysis and the assumptions set forth in Topical Report ER 80P. The staff finds that, because the calibration of the LEFM system is verifiable online, and because the preventive maintenance activities are developed in accordance with guidance provided by Caldon, the proposed maintenance and calibration procedure are acceptable.

With respect to the processes and contingencies for an inoperable LEFM and the effect on thermal power measurement and plant operation, the licensee stated that CR-3 would rely on the currently-installed flow nozzles and other instrumentation to perform the secondary calorimetric when the high-accuracy instrumentation is not available. The licensee stated that AULD can operate from either the high-accuracy calorimetric or the existing instrumentation. The licensee will reduce the core power level to 2568 MWt (the current RTP) within 12 hours if the high-accuracy heat balance is not restored. The licensee stated that this is a reasonable timeframe to conduct an orderly power reduction. The licensee will also change the Nuclear Overpower - High Setpoint (Function 1.a. of TS Table 3.3.1-1, "Reactor Protection System Instrumentation," also called the high-power trip) trip setpoint, as described below.

The high-power trip receives a signal from the NI, which is calibrated to the core power level as determined through the secondary calorimetric. TS Surveillance Requirement (SR) 3.3.1.2 requires the NI to be compared to the heat balance every 12 hours, and SR 3.3.1.5 requires the high-power trip to be calibrated every 92 days. The licensee determined that there is insufficient margin in the DBA analyses to support the higher high-power trip setpoint if the high-accuracy heat balance is not available. The licensee will reduce the setpoint to 103.3 percent of RTP, which is approximately equal to the existing setpoint (103.3 percent of 2609 MWt is approximately 104.9 percent of 2568 MWt). The licensee proposed a completion time of 48 hours to reduce the trip setpoint, stating that resetting the setpoint due to failed equipment is expected to take a maximum of 16 hours, with additional time required for callouts, planning, and preparations. The licensee also stated that the NI was compared to the last known good high-accuracy heat balance and the NI do not routinely require adjustments; therefore, the NIs can continue to be relied upon for power measurement. The licensee's submittal shows that the current value used for the drift component in the setpoint calculation is 0.399 percent over 30 months. As such, the expected setpoint drift over 48 hours is insignificant and the NIs will remain calibrated for an extended period of time. Therefore, the staff finds the licensee's actions and associated timeframes acceptable for addressing loss of the high-accuracy calorimetric.

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

Criterion 2 does not apply to CR-3.

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

The licensee confirmed that feed flow and temperature uncertainties were combined with other plant measurement uncertainties to calculate the overall heat balance uncertainty using accepted plant setpoint methodology. The LEFM uncertainty calculation itself, however, is based on the American Society for Mechanical Engineers (ASME) Standard PTC 19.1, "Measurement Uncertainty," methodology, and on calibration tests performed by the Alden Research Laboratory (ARL). Both the LEFM uncertainty calculation and the setpoint methodology use a square-root-sum-squares calculation and are, therefore, consistent. The staff concludes that CR-3 meets Criterion 3.

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

The licensee stated that it applied a bounding uncertainty for the LEFM for the total power measurement uncertainty, and that the acceptability of the bounding calibration factor for the CR-3 spool prices was established by tests of the spools at ARL. The tests used a full-scale model of pertinent portions of the CR-3 FW piping to accurately account for flow profile effects. These tests are documented in Caldon Report ER-608, Revision 2, which was submitted by letter dated November 1, 2007. The licensee stated that this test report and the site-specific bounding uncertainty analysis using the results of this test report will be maintained as the CR-3 design basis calculation.

The NRC staff's July 5, 2006, letter regarding Caldon UFMs concluded, in part, that (1) the CheckPlus™ system is relatively unaffected by flow profile and swirl, and can provide an approximation of the flow profile (this conclusion does not necessarily apply if the flow profile consists of multiple individual flow paths, such as the profile identified in the licensee's ARL tests), (2) the flatness ratio (defined as the ratio of the measured average axial velocity at the outside chords to the inside chords) can be correlated to the calibration coefficient so that reliance on a Reynolds Number extrapolation is not necessary to apply ARL test results to plant applications, and (3) there is a firm theoretical and operational understanding of behavior such that, with one exception, there is no further need to re-examine the hydraulic bases for use of the Check and CheckPlus™ systems in nuclear power plant FW applications. The exception was that the effect of transducer replacement on the uncertainty should be evaluated.

With respect to transducer replacement, Caldon addressed the issue on a generic basis and submitted ER-551P Rev. 1, "LEFM [CheckPlusTM] Transducer Installation Sensitivity," dated March 2007 (ML072740228), and PR-612P Rev. 0, "Flow Measurement Uncertainty due to Transducer Replacement in Caldon LEFM Check and CheckPlus Systems," dated March 15, 2007 (ML070870435). Caldon conducted a number of tests in which the 16 transducers were removed and replaced. Each of the tests consisted of a statistically meaningful number of individual determinations of the calibration factor. The calibration factors and associated uncertainties were provided. The calibration factor variation was shown to be limited to changes in the fourth significant figure. For the CR-3 installation, the licensee added an uncertainty term in the overall CheckPlus™ uncertainty calculation to address uncertainty due to transducer installation variability. The NRC staff finds that transducer installation variability has been acceptably addressed for CR-3.

The licensee's submittals described the ARL test configuration and the UFM calibration that was performed for CR-3. In its letter dated November 1, 2007, the licensee provided a copy of the test report and isometric drawings of pertinent portions of the FW piping. The hydraulic loop configuration at ARL was designed to duplicate the principle hydraulic features of the CR-3 installation.

The NRC staff reviewed the piping configurations and the testing performed at ARL. The NRC staff observed that there is a substantial distance from the flow control valves, through elbows and the FW heaters, before reaching the part of the FW system that was modeled in the tests. The staff determined that no components that would cause perturbations significant to flow measurement were omitted from the testing. However, as noted by the licensee, the test section of Train B was a vertical image of the plant installation. The licensee indicated that the fluid inertial forces are produced by changes in direction due to the upstream bends, so the vertical image does not impact the tests. The licensee indicated the fluid velocity profile is dominated by vortices and other transverse velocity components created by inertial forces, and these forces far exceed any forces due to gravity or viscosity. The NRC staff agrees, but notes that any distortion of the flow profile due to elbows in the mirror configuration will also be a mirror image and, unless the UFM is also rotated 180 degrees, the calibration will be based on a rotated flow profile. The NRC staff does not consider this effect to be significant because of the CheckPlus™ symmetry and its demonstrated lack of sensitivity to changes in flow profile caused by such hardware as elbows and valves. In addition, the licensee ran tests in which the UFM was rotated in the pipe, and the effect was quantitatively shown to be negligible.

The licensee initially intended to install the UFMs between the FW flow straighteners and the flow nozzles. The licensee's submittals described an initial ARL test configuration with the CheckPlus™ located immediately downstream of tubular flow straighteners similar to those currently installed in the CR-3 FW piping. The channelization of flow exiting the flow straighteners was found to adversely impact CheckPlus™ performance. As a result, the licensee changed the CR-3 FW system as follows:

- Train A initial test results did not show significant swirl, as was expected due to the lack of elevation changes, the long pipe runs, and the presence of a single elbow. Further, plant interferences made it difficult to relocate the flow straightener downstream of the UFM. Consequently, the licensee removed the flow straightener from Train A.

- Train B has a more complex upstream geometry that includes horizontal and vertical bends. The resulting swirl was reported to not adversely impact the CheckPlus™ but there was concern it might affect the existing flow nozzle if the flow straightener was removed. Therefore, the licensee relocated the Train B flow straightener to downstream of the UFM, but upstream of the flow nozzle.

These results and conclusions are reasonable. The tubular flow straightener will result in numerous small flow profiles (one for each tube) immediately downstream of the flow straightener, and it will take a few pipe diameters for a smooth, continuous flow profile to become re-established. The CheckPlus™ sonic path selections have been demonstrated to provide reasonable representation of symmetric and skewed flow profiles, but may be affected by flow that consists of multiple individual flow profiles. There is no practical theoretical method to predict the distance downstream of a flow straightener that is necessary for a full-pipe flow profile to be re-established. The staff considers this an excellent example of the need to perform UFM tests that accurately simulate the installation and operation conditions. Since the installation configuration will not include flow straighteners that potentially influence the CheckPlus™, the NRC staff did not examine the data from the cases where the flow straightener effects were found.

The licensee concluded that the final tested configuration accurately reflects the post-MUR plant configuration over a sufficient length of pipe to capture all necessary parameters. Therefore, the staff concludes that additional justifications are not needed to satisfy Condition 4 of the staff's SE on Caldon Topical Report ER-80P. The staff finds that the test configuration was acceptably modeled.

The calibration tests consisted of a statistically meaningful number of runs that were parametric in flow rate, with multiple CheckPlus™ flow rate indications per test. Flatness ratios, swirl, and flow rate behavior were consistent with observations at other facilities in that there was little variation of calibration factor (changes were small changes and in the third significant figure). In addition, one Train B test was conducted with the CheckPlus™ rotated in the pipe to provide an independent check on the assumption that the mirror image would have no significant effect on the test results. This case showed essentially no change in calibration factor and a small change in flatness ratio compared to the corresponding nonrotated test. These changes are judged to be well within the repeatability of the tests.

The data were averaged over all flow rates for each model test when calculating the calibration factor. Examination of the flow rate data showed a slight, insignificant upward trend in correction factor with increasing flow rate. Thus, the calibration factor based on the average flow rate will be slightly low which will introduce a nonconservatism into the results. The NRC staff concluded that the effect was so small that it could be neglected, in part because other conservatisms introduced into the overall calibration, such as the treatment of the sensitivity of the calibration to transducer installation, and the use of an allowable variation in flatness ratio during plant operation to assure remaining within the claimed uncertainty, are more than sufficient to compensate for the small nonconservatism.

The test data were combined to yield calibration factors of 0.9983 and 0.9991 for the Train A and Train B UFMs, respectively. The corresponding uncertainties were both 0.26 percent, which yields a combined uncertainty of 0.20 percent for both loops. The small correction between the uncalibrated CheckPlus^{TM} and ARL flow rates means that any reasonable calibration error will have little effect on indicated flow rate. This is consistent with the staff's previous observations, as documented in the July 5, 2006, letter to Caldon.

In summary, the licensee has adequately addressed plant-specific implementation of the LEFM CheckPlus^{TM} system, including the four criteria specified in the staff SEs on the Caldon topical reports, and therefore has adequately addressed the guidance in Items A, B, C, D, G, and H of Section I of Attachment 1 to RIS 2002-03.

The NRC staff finds that the hydraulic aspects of the Caldon LEFM CheckPlus^{TM} UFM system have been accurately described and that there is a firm theoretical and operational understanding of the hydraulic aspects of its behavior. The NRC staff further finds that the calibration testing at ARL is appropriate for installation at CR-3. Therefore, the NRC staff finds that the proposed installation of the UFM at CR-3 is acceptable.

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

To address Item E of RIS 2002-03, the licensee submitted the heat balance uncertainty calculation as Attachment E to the June 28, 2007, letter. This calculation details all uncertainties contributing to the total power measurement uncertainty. The calculation assumed that the uncertainty contribution of the UFM would be 0.34 percent. By letter dated August 30, 2007, the licensee submitted Engineering Report ER-579, Revision 2. This calculation determined that the total power measurement uncertainty due to the UFM (flow and temperature measurement) and FW pressure contributors is 0.3 percent. This verifies the conservatism in the heat balance and supports the proposed 1.6 percent power uprate.

The staff reviewed the heat balance calculation. The calculation determined individual measurement uncertainties of all parameters contributing to the core thermal power measurement uncertainty, and applied them as a bias or combined them using the square root of sum of squares (SRSS) methodology, as described in Regulatory Guide 1.105 and Instrumentation, Systems, and Automation Society Standard S67.04. The computed residual uncertainty is 0.394 percent of 2609 MWt and breakes down into the following contributions: FW flow and temperature measurement accounts for approximately 84 percent; new steam temperature/pressure instrumentation approximately 4 percent; steam pressure measurement approximately 1 percent; and RCP energy, ambient loss, and atmospheric pressure correction approximately 11 percent of the uncertainty. Errors and biases were computed and combined according to the procedures defined in ASME PTC 19.1. This document defines the contributions of individual uncertainty elements through the use of sensitivity coefficients. That is, the governing equation is differentiated with respect to the contributing measured variables, and the products of the partial derivatives and individual measurement uncertainties are squared, summed, and their square root is computed. To obtain the total core power uncertainty, these elements are then combined as SRSS or treated as biases. The staff concludes that the uncertainty of 0.394 percent of 2609 MWt was determined using NRC-approved methodologies. Based on its review of the licensee's submittals, the staff finds the computed value of uncertainty to be acceptable in support of the TS change. The licensee has adequately addressed the guidance in Item E of Section I of Attachment 1 to RIS 2002-03.

To address Item F of RIS 2003-02, the licensee addressed each of the five aspects of the calibration and maintenance procedures for the instruments that affect the power calorimetric. The calibration of the UFM was addressed in response to Criterion 1. For the other instruments, the licensee stated that they would be maintained according to their required calibration and maintenance procedures, the hardware and software would be controlled in accordance with the licensee's engineering change procedures, and any problems would be addressed through the licensee's corrective action program. Reporting deficiencies to the manufacturer and addressing manufacturer deficiency reports would also be addressed through the licensee's standard programs. The staff review of the licensee's above statements found that the licensee addressed the calibration and maintenance aspects of the CheckPlus™ system and all other instruments affecting power calorimetric and, thus, complied with the guidance of item F of Section I of Attachment 1 to RIS 2002-03.

In summary, the staff reviewed the licensee's plant-specific implementation of the FW flow measurement device and found it consistent with the staff-approved topical reports. The staff reviewed the licensee's power uncertainty calculations and found that the licensee adequately accounted for instrumentation uncertainties in the reactor thermal power measurement, and the calculations meet the relevant requirements of 10 CFR Part 50, Appendix K. Finally, the staff found that the licensee's proposed power measurement uncertainty of 0.4 percent of RTP acceptable in support of the proposed 1.6 percent power uprate.

3.2 Instrumentation and Controls (I&C)

In addition to the plant-specific application of the CheckPlus™ UFM, the NRC staff in the area of I&C also reviewed the licensee's setpoint determination that supports the change to TS 3.3-1, "Reactor Protection System (RPS) Instrumentation." Paragraph 10 CFR 50.36(d)(1)(ii)(A) states, "Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded." Furthermore, 10 CFR 50.36(d)(3) states, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met." RIS 2006-17, "NRC Staff Position on the Requirements of 10 CFR 50.36, "Technical Specifications," Regarding Limiting Safety System Settings During Periodic Testing and Calibration of Instrument Channels," dated August 24, 2006 (ML051810077), addresses the NRC's requirements on limiting safety system settings (LSSSs) assessed during periodic testing and calibration of instrumentation. This RIS discusses issues that could occur during testing of LSSSs and which, therefore, may have adverse effect on equipment operability. If the high-accuracy heat balance is not available, the licensee must reduce the high-power trip, which is an LSSS parameter in the CR-3 TSs.

In its October 18, 2007, response to staff request for additional information (RAI) regarding the high-power trip SR, the licensee added the two notes to TS Table 3.3.1-1, and provided statements in the bases, as described in the NRC letter to Nuclear Energy Institute dated September 7, 2005 (ML052500004) and further explained in RIS 2006-17.

In its November 1, 2007, letter, the licensee provided the NI calibration procedure and high-power trip setpoint calculation. The document calculated high-power trip instrumentation limiting setpoint and termed it "In-Plant Setpoint." FPC stated that the in-plant setpoint is based on the total loop uncertainty that is calculated using the CR-3 plant specific methodology. The setpoint calculation also determined as-found and as-left setpoint tolerances of the NI to establish setpoint allowable values (AVs). The licensee stated that the RPS instrumentation setpoint AVs are based on protecting the analytical limits used in CR-3 safety analysis with the consideration of appropriate uncertainties. The licensee's calibration procedure demonstrates how the surveillances are performed. The licensee stated that the methodology used to develop the in-plant setpoint, the as-left and as-found setpoints, and setpoint AVs will be documented in the CR-3 FSAR.

The staff reviewed the licensee's setpoint calculation and surveillance procedure. The staff found that the methodology is acceptable and that the calculated setpoint tolerances have sufficient margin to the AV. The staff also found that the proposed changes to the TSs adequately address the issue in RIS 2006-17 and are acceptable. The changes to the TSs are discussed in detail in Section 3.11 of this SE.

3.3 Reactor Systems

Nuclear Steam Supply System Parameters

The NSSS operating parameters provide the RCS and secondary system conditions (pressures, temperatures, and flow) that are used in the analyses and component evaluations. The licensee established the operating parameters using conservative assumptions to provide bounding conditions to be used in the analyses. The major operating conditions are as follows:

Parameter	Case A	Case B	Case C
Core Thermal Power (MWt)	2568	2609	2609
Other RCS Power MWt (RCPs, heat loss, etc.)	16	16	16
Total RCS Power (MWt)	2584	2625	2625
Pressurizer Control Pressure (psig)	2155	2155	2155
SG A/B Tube Plugging (percent)	2.4/5.7	2.4/5.7	20/20
T_{hot} (ºF)	601.9	602.2	602.9
T_{cold} (ºF)	556.2	555.8	555.1
T_{avg} (ºF)	579	579	579
RCS Mass Flow (million lb_m/hr)	144.05	144.08	139.81
RCS Volumetric Flow (gpm)	386,873	386,729	374,896
Steam Temperature (ºF)	591.0	590.5	580.7
Steam Superheat (ºF)	54.9	54.4	44.6
FW/Steam Flow Rate (million lb_m/hr)	10.86	11.07	11.19
SG Steam Pressure (psia) (input)	931.7	931.7	931.7
FW Temperature (ºF) (input)	456.7	458.4	458.5

Case A – Existing tube plugging at 2568 MWt
Case B – Existing tube plugging at 2609 MWt
Case C – 20 percent tube plugging at 2609 MWt

In all analyses, the licensee referenced the current analysis of record, which used previously NRC-approved computer codes and methodologies for each analysis. The power uprate does not impact on the events that initiate at zero power because there was no change in the zero power conditions or in any of the trip setpoints. For events that are analyzed at power, the analyzed core power level was generally 2619 MWt, 2 percent greater than the current licensed core power level of 2568 MWt and 0.4 percent greater than the MUR core power level of 2609 MWt. Exceptions to this are noted below. In some cases, generic, bounding B&W analyses are referenced, which were performed using an analyzed power level of 2772 MWt (108 percent of current RTP). The staff previously reviewed and approved the licensee's transient accident analyses at 2619 MWt conditions during the review of the previous power uprate (see the staff's SE dated November 1, 2002, ML023050463), confirming that the acceptance criteria were still met under these conditions.

Results of the NRC staff's review of the MUR conditions are summarized in the following table. The staff dispositioned those analyses that were not performed at or greater than 102 percent of current RTP as follows:

Uncompensated Operating Reactivity Changes

The licensee stated that this analysis was originally performed to show that variations in reactivity during the cycle change slowly and are well within the capability of the control systems or by manual operator action to mitigate. The licensee stated this analysis is not significantly effected by core power because higher core power results in only minor changes to the fuel depletion, burnable poison depletion, and fission product concentrations that cause the reactivity change. The staff finds this conclusion to be reasonable. Also, no safety system actuation is required to mitigate this occurrence, and from a transient perspective this analysis is bound by the control rod withdrawal at power event.

Rod Withdrawal at Rated Power Operation Accident

These transients are terminated by the RCS pressure trip or the Nuclear Overpower - High Setpoint trip, depending on the rate of reactivity insertion. For the slow reactivity insertion rates, the event is terminated by the RCS pressure trip. Although the DBA analysis assumes an initial operating power of 2568 MWt, the licensee stated that the same amount of energy would be required to increase the RCS pressure and temperature to the assumed analytical limit because the initial RCS average temperature and pressurizer level are the same for MUR conditions. Therefore, while a different reactivity insertion rate may become limiting, the current analyses bound the proposed MUR uprate condition. For the high reactivity insertion rates, which are terminated by the high-power trip, the licensee indicated that these events are more severe when there is a larger difference between the initial power level and the trip setpoint (for a given trip setpoint). The analysis assumes a high-power trip analytical limit of 112 percent of current RTP. As a part of the MUR, this analytical limit will be reduced to 110.2 percent of the uprated RTP, which is the same power level in MWt (2876 MWt) as the existing analyses. Thus, since the event uses the same setpoint and a lower initial power level, the current analyses bound the MUR conditions.

Disposition of Accidents and Transients in CR-3 FSAR

Accident/Transient	Analyzed Core Power Level (based on 2568 MWt)	Analysis of Record Bounds MUR Uprate	NRC Staff Conclusion/Discussion
Uncompensated Operating Reactivity Changes	100.2 %	Yes	Acceptable, see below
Startup Accident	10^{-7}%	Yes	Acceptable
Rod Withdrawal at Rated Power Operation Accident	100% 2% uncertainty in setpoint	Yes	Acceptable, see below
Moderator Dilution Accident	102%	Yes	Acceptable
Cold Water Accident	50%	Yes	Acceptable
Loss-of-Coolant Flow Accident	102%	Yes	Acceptable
Stuck-Out, Stuck-In, or Dropped Control Rod Accident	108%	Yes	Acceptable
Load Rejection Accident (Turbine Trip)	112%	Yes	Acceptable
Station Blackout Accident	108%	Yes	Acceptable
Steam Line Failure Accident	100%/102%	Yes	Acceptable, see below
Steam Line Failure Mass & Energy Releases	102%	Yes	Acceptable
Steam Generator Tube Rupture Accident	102%	Yes	Acceptable
Fuel Handling Accident	102%	Yes	Acceptable
Rod Ejection Accident	0.1% 100%	Yes	Acceptable, see below
Loss-of Coolant Accident	102%	Yes	Acceptable
Loss-of Coolant Accident Mass & Energy Release	102%	Yes	Acceptable
Makeup System Letdown Line Failure Accident	102%	Yes	Acceptable
Waste Gas Decay Tank Rupture Accident		Yes	Acceptable, based on tank inventory, not power level
Loss of FW and Main FW Line Break Accident	102%	Yes	Acceptable
Anticipated Transient without Scram	108% 52%	Yes	Acceptable, see below
Anticipatory Reactor Trip System	108%	Yes	Acceptable

Steam Line Failure Accident

The licensee stated that the core power in the analysis is 100 percent of current RTP in order to minimize the heat input to the RCS. A power level of 102 percent of RTP was assumed for determining the SG mass inventory. This analysis method (assuming lower core power in the primary than the secondary) has previously been approved by the staff. In that sense, the licensee indicated that the current analysis bounds the uprated conditions. The licensee also stated that dose release calculations are evaluated on a reload basis assuming 102 percent current RTP, and the containment response is based on 102 percent of current RTP. Based on the above, the staff concludes the steam line break accident analyses are acceptable for MUR conditions.

Rod Ejection Accident

The licensee stated that this event resulted in a small (2 cal/gm) increase in the peak enthalpy of the fuel, but that the results remain within the acceptance criteria. In its letter dated November 1, 2007, the licensee stated that, based on the analysis in the CR-3 FSAR, ejection of a rod of maximum worth at CR-3 causes localized fuel enthalpy to increase by approximately 105 cal/gm to a peak of approximately 200 cal/gm. The acceptance criterion is 280 cal/gm. The licensee indicated that the change is due to the initial fuel enthalpy, which is approximately proportional to the initial local power level, and that the reactivity transient is based on rod worth and is unaffected. The slight increase in peak fuel enthalpy remains bounded by the acceptance criterion for this accident. Therefore, the staff finds the results of this accident would be acceptable at CR-3 under MUR conditions.

Anticipated Transient without Scram (ATWS)

The ATWS event was analyzed to demonstrate compliance with 10 CFR 50.62, “Requirements for the reduction of risk from [ATWS] events for light-water-cooled nuclear power plants.” An ATWS event is an anticipated occupational occurrence followed by the failure of the reactor trip system to scram the reactor. The licensee installed a diverse scram system (DSS) and ATWS mitigating system actuating circuitry (AMSAC) to address this event. The DSS interrupts power to control systems at high RCS pressure, resulting in a scram. AMSAC actuates the emergency FW pumps and trips the turbine. The CR-3 TSs require AMSAC to be operable (armed) at power levels above 50 percent of current RTP.

The limiting ATWS transient for the B&W plant is a loss-of-FW initiating event. The CR-3 analysis of record is a generic B&W evaluation that is based on 2772 MWt, which bounds the MUR conditions for cases where AMSAC is armed. In order to verify that the ATWS arming setpoint of 50 percent, as specified in the CR-3 TSs, remains valid for the MUR power level, the licensee performed an ATWS analysis assuming 52 percent of 2609 MWt. The purpose of the analysis was to demonstrate that, for events initiated at lower core power and without AMSAC, the peak RCS pressure will not exceed 3250 pounds per square inch absolute (psia). The licensee used the staff-approved RELAP/5/MOD2-B&W computer code and the base model from the CR-3 FSAR Chapter 14 analyses.

The revised analysis was summarized in Attachment F to the June 28, 2007, submittal. The licensee determined that DSS and emergency FW actuation (from low SG level) terminated the event without reliance on AMSAC. Peak RCS pressure was 2616 psia, which is significantly below the acceptance criterion of 3250 psia. Therefore, the licensee has adequately demonstrated that AMSAC arming setpoint remains acceptable for MUR conditions.

3.4 Electrical Systems

The NRC staff reviewed the power uprate for impact on the CR-3 electrical systems. The staff applied the following regulatory requirements: 10 CFR Part 50, Appendix A, General Design Criterion (GDC) 17, "Electric power systems," which requires that an onsite power system and an offsite electrical power system be provided with sufficient capacity and capability to permit functioning of structures, systems, and components (SSCs) important to safety; 10 CFR 50.63, "Loss of all alternating current [AC] power," which requires that all nuclear plants have the capability to withstand a station blackout (SBO), defined as a loss of all AC power, for an established period of time and to recover therefrom; and 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," which requires licensees to establish programs to qualify electric equipment important to safety.

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

- AC Distribution System
- Power Block Equipment (Generator, Exciter, Transformers, Isolated-phase bus duct, Generator circuit breaker)
- Direct Current (DC) system
- Emergency Diesel Generators (EDGs)
- Switchyard
- Grid Stability
- SBO
- Equipment Qualification Program

AC Distribution System

The AC Distribution System is the source of power for the non-safety-related buses and the safety-related emergency buses supplying the redundant engineered safety features loads. It consists of the 6.9 kV, 4.16 kV, 480 V, and 120 V systems (not including the EDGs).

By letters dated September 13 and November 1, 2007, the licensee identified the components and power sources affected by the power uprate and provided the pre-MUR and post-MUR condition load changes. Based on the review of its calculations, the licensee concluded that the equipment affected by the MUR-related changes (condensate pumps, FW booster pumps, unit auxiliary transformer, and start up transformer) are acceptable for operation at 2609 MWt.

The staff reviewed the licensee's submittals and agrees with the licensee that, while the AC power system will experience minor load changes, the AC power system has adequate capacity to operate the plant equipment within the design to support the MUR power uprate.

Power Block Equipment

As a result of the power uprate, the rated thermal power will increase to 2609 MWt from the previously analyzed core power level of 2568 MWt. This increase in thermal power will result in an increase in the electrical power output, which will effect the power block equipment.

The staff asked the licensee to provide additional information on the effects of the MUR power uprate on the main generator rating and power factor, isolated-phase bus ducts, and main generator breaker. In response to the staff's questions, the licensee stated that the CR-3 electrical generator is rated at 989.4 megavolt-amps (MVA), 0.90 power factor (which equates to approximately 430 megavolt-amps reactive (MVAR)). The generator is operated with an administrative limit of 300 MVAR. At this reactive load, the maximum output from the generator reactive capability curve is approximately 950 megawatts electric (MWe). The MUR will increase output from approximately 900 MWe to 914 MWe. Therefore, the increase from the MUR power uprate is still well below the main generator maximum capability.

The isolated-phase bus is rated for 27,500 amps (A). The typical 100 percent power operating current ranges between 23,500 A and 24,500 A. The 1.6 power increase from the MUR will increase the current on the isolated-phase bus to approximately 24,000 A to 25,000 A. Therefore, the increase from the MUR power uprate is still well below the isolated-phase bus maximum capability.

The 500 kV line from the step-up transformer joins the 500 kV ring bus between the two generator breakers. The main generator output breakers are rated for 3000 A with a short circuit current rating of 37,000 A. The current through a breaker before uprate is approximately 1300 to 1800 A, which is well below the 3000 A rating of these breakers. With a 1.6 percent power uprate, the expected current through the breaker will be approximately 1325 to 1850 A, and therefore, there will still be significant margin in the breaker rating.

In response to the staff RAIs, the licensee stated that CR-3 maintains a maximum MVAR limit based on manufacturer's recommendations to prevent damage to the generator. This limit is approximately 430 MVAR. The transmission system operator will not require additional MVAR from CR-3 post implementation of the MUR uprate. Therefore, there is no depletion (shortfall) of MVAR capability resulting from the MUR uprate. As such, there are no compensatory measures to be taken to compensate for the shortfall.

The step-up transformer (main power transformer) will be replaced prior to the MUR uprate implementation. The current transformer rating is approximately 950 MVA, and the current load is 900 MWe. The new step-up transformer will have nominal rating of 1200 MVA, which is enough capacity to accept an additional 14 MWe due to the proposed MUR power uprate. The unit auxiliary transformer is capable of handing full in-house loads pre- and post-MUR uprate.

In summary, the staff reviewed the generator step-up transformers, unit auxiliary transformers, reserve auxiliary transformers, isolated-phase bus ducts, and generator circuit breaker analyses provided in the LAR, and the supplemental information provided by the licensee in its letters dated September 13, 2007 and November 1, 2007. Based on the above review, the staff agrees that the small increase in generator output (914 MWe) does not cause overloading of the generator circuit breaker, the isolated-phase bus duct or the newly installed generator step-up transformer. There are no significant changes in AC distribution system loads. Therefore, the ratings of unit auxiliary transformers and reserve auxiliary transformers are not impacted by MUR power uprate conditions.

DC System

The station 250/125 V DC system is comprised of batteries, battery chargers, and distribution equipment that supply power to station loads. The nuclear safety-related (Class 1E) portion of the DC System consists of 250/125 V DC motor control centers, 125 V batteries, battery chargers, two essential distribution panels, and 480 VAC/125 VDC rectifiers. It provides the source of power for direct current load groups, vital control and instrumentation, power and control of Class 1E, and selected nonClass 1E electrical equipment.

The licensee stated that the DC system, post-MUR, is bounded by the existing analyses of record. The LAR states that the analysis demonstrates that the system has adequate capacity and capability to operate the plant equipment.

The staff reviewed the LAR and the CR-3 FSAR. There are no significant changes in DC system loads. Therefore, the staff agrees with the licensee that the analyses for DC system reasonably bound the MUR power uprate conditions.

EDGs

The EDG system provides a safety-related source of AC power to sequentially energize and restart loads necessary to shutdown the reactor safely, and to maintain the reactor in a safe shutdown condition, in the event that the normal AC power is interrupted. There are two EDGs, each dedicated to one of the safety-related, redundant electrical trains. In addition, there is an alternate AC diesel that can be aligned to either safety-related AC distribution bus.

The licensee stated that margin currently exists on the EDGs and alternate AC diesel generator. The licensee also stated that the MUR power uprate will not significantly change the loading of the EDGs or the alternate AC diesel generator. As a result, the EDGs will continue to have adequate capacity to operate the plant equipment.

Based on the above, the staff agrees with the licensee that the EDG system will have adequate capacity to support the MUR power uprate conditions.

Switchyard

The switchyard equipment and associated components are classified as non-safety related. The primary function of the 500 KV switchyard and distribution system is to connect the station electrical system to the transmission grid. A separate 230kV switchyard functions to provide power input into the plant. The interconnection allows for (1) the normal flow of power out of the station to the grid when the main generator is operating, and (2) the flow of power from the grid to the station auxiliaries when the main generator is shut down. The small increase in plant output does not significantly impact the switchyard equipment. Therefore, the staff agrees that the analyses for the switchyard system by FPC reasonably bound the MUR power uprate conditions.

Grid Stability

The grid stability study evaluated the steady-state and transient performance of the FPC system with both the existing CR-3 power level and the uprated power level. The power flow study was performed to support the additional capacity expected to be installed in anticipated future power uprates at CR-3 and the addition of power plants at the Levy County site (expected after all the CR-3 uprates). This study includes the CR-3 full uprated condition, which is expected to result in an increase of 180 MWe and be completed by summer of 2012. The licensee analyzed both the power flow and grid stability.

The steady state analysis was performed to examine post transient power flow for the bulk Florida transmission system to determine if the loss of line and units subsequent to breaker failure will cause unacceptable voltage and/or overload of system components. The licensee determined that post-transient power flow does not affect grid stability/reliability.

The licensee studied the following scenarios:

1. Three phase fault on the Crystal River 500 kV bus, cleared normally (National Electric Reliability Council (NERC) Category B1).
2. Three phase fault on the Crystal River 500 kV bus, with breaker failure condition (NERC Category D1).
3. Three phase fault on Crystal River 500 kV bus, critical clearing time.

A comparison of the results indicated that an increase in capacity will not have an adverse effect on the stability of power grid. A NERC Category C fault was not studied since a more severe Category D event was simulated and the results are satisfactory. The loss of CR-3 does not violate any NERC criteria, and the dynamic performance of the bulk power grid remains stable.

Based on the analysis performed, the licensee concluded that the scenarios studied do not indicate that increasing the power at CR-3 has an adverse effect on grid stability. All the simulations and studies are based upon the full 180 MW increase, which bounds the MUR power uprate conditions.

The staff reviewed the grid stability study, and agrees with the licensee that the CR-3 MUR power uprate allows for stable and reliable grid operation.

SBO

The regulations in 10 CFR 50.63 require that each light water cooled nuclear power plant be able to withstand and recover from a loss of all AC power, which is also referred to as SBO. The licensee indicated that the MUR does not increase any loads in response to an SBO.

The CR-3 SBO coping duration is four hours. This is based on an evaluation of the offsite power design characteristics, emergency AC power system configuration, and emergency diesel generator reliability. The evaluation was done in accordance with the evaluation procedure outlined in NUMARC 87 00 and Regulatory Guide (RG) 1.155. The offsite power design characteristics include the expected frequency of grid-related loss of offsite power, the estimated frequency of loss of offsite power from severe and extremely severe weather, and the independence of offsite power. The analyses documented in the CR-3 FSAR were evaluated at 2772 MWt, which bounds the MUR power uprate conditions.

The DC power system supplies the equipment needed to cope with SBO. The MUR will not impact the loads needed to cope with a SBO and, thus, the MUR conditions are bounded by existing load profiles. Therefore, CR-3 will continue to meet the requirements of 10 CFR 50.63 for the MUR conditions.

Equipment Qualification Program

The licensee stated that the MUR power uprate does not change the accident/post accident temperature profiles inside containment; therefore, there is no impact on environmentally qualified equipment. The environmental qualification of electrical equipment was performed for a core power level of 2619 MWt, which bounds the MUR operating conditions. Based on the above, staff agrees that the MUR power uprate will have no significant impact on the environmental qualification of electrical equipment.

In summary, the NRC staff agrees that implementation of the MUR power uprate will continue to meet the requirements of applicable sections of 10 CFR Part 50 with respect to electrical systems.

3.4 Mechanical and Civil Engineering

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

These piping systems, components, and their supports, including core support structures, are designed in accordance with the rules of the ASME Boiler and Pressure Vessel Code (Code), Section III, and American Standards Association B31.7 and B31.1.

The NRC staff review focused on verifying that the licensee has provided reasonable assurance of the structural and functional integrity of piping systems, components, component internals, and their supports under normal and vibratory loadings, including those due to fluid flow, postulated accidents, and natural phenomena such as earthquakes. The acceptance criteria are based on continued conformance with the requirements of the following regulations: (1) 10 CFR 50.55a, "Codes and standards," and GDC 1 as they relate to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed; (2) GDC 2 as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC 4 as it relates to structures and components important to safety being designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal and accident conditions; (4) GDC 10 as it relates to reactor internals being designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences; (5) GDC 14 as it relates to the reactor coolant pressure boundary being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; and (6) GDC 15 as it relates to the RCS being designed with sufficient margin to ensure that the design conditions are not exceeded. The specific review areas are contained in Section 3.9 of the Standard Review Plan (NUREG-0800, or SRP). The review also includes the plant-specific provisions of Generic Letter (GL) 95-07, “Pressure Locking and Thermal binding of Safety-Related Power-Operated Gate Valves,” GL 96-06, “Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions,” GL 89-10, “Safety-Related Motor-Operated Valve Testing and Surveillance,” and GL 96-05, “Periodic Verification of Design--Basis Capability of Safety-Related Motor Operated Valves.”

The power uprate will be achieved by an increase in steam flow and FW flow, and an increase in the temperature difference across the core. The RCS pressure, RCS average temperature, and SG secondary pressure will remain the same.

The first table in Section 3.3 of this SE shows the pertinent temperatures, pressures, and flow rates for the current conditions and the uprated conditions. At full power, the hot-leg temperature increases from 601.9 to 602.9 degrees Fahrenheit, the cold-leg temperature decreases from 556.2 to 555.1 degrees Fahrenheit, the SG pressure remains unchanged at 931.7 psia, the steam flow increases from 10.86 to 11.19 million pounds per hour (Mlbm/hr), the FW temperature increases from 456.7 to 458.5 degrees Fahrenheit, and the FW flow increases from 10.86 to 11.19 Mlbm/hr. The proposed uprate does not change heatup or cooldown rates or the number of cycles assumed in the design analyses. In addition, there are no changes in the design transients since the safety analyses were performed at 102 percent of RTP. Thus, the limiting analyses are still bounding.

The design parameters for the RCS and SGs are found in Tables 4-6 and 4-4, respectively, of the CR-3 FSAR. The RCS components, including the RV, core support structures, and SGs, were designed to operate at a core power level of 2619 MWt. The RCS components are designed to 650 degrees Fahrenheit (except the pressurizer, which is designed to 670 degrees Fahrenheit) and 2500 psia. Section 4.1.1 of the CR-3 FSAR states that the SGs are capable of producing a total steam flow of 11.2 Mlbm/hr. The FW system design temperature is 459 degrees Fahrenheit (FSAR Table 4-4).

Reactor Pressure Vessel and Internals

The code of record for the RV, nozzles, and supports is the ASME Code, Section III, 1965 Edition, including all addenda through the Summer 1967 Addenda. The RV closure head was designed to ASME Code, Section III, 1989 Edition.

The licensee compared the expected temperatures and pressures for the proposed power uprate condition against the analysis of record. The licensee confirmed that there is no change in RCS design or operating pressure, and the effects of operating temperature changes for cold and hot legs are within design limits. The MUR power uprate conditions are bounded by the design conditions. In addition, the operating transients will not change as a result of the power uprate and no additional transients have been proposed. The existing loads, stresses, and

CUFs for RV and internals remain valid for the proposed power uprate. Since the post-MUR temperatures and pressures remain within the design conditions, the transients remain bounded, and no new transients are added to the design basis, the NRC staff finds that the RPV and internals are acceptable for operation at the uprated power level.

Control Rod Drive Mechanisms

The code of record for the pressure retaining components of the CRDMs is the ASME Code, Section III,1965 Edition through Summer 1967 Addenda. The design conditions in the existing analyses are based on the RCS functional specification. The MUR did not result in changes in the RCS design or operating pressures, and the effects of operating temperature changes for cold and hot legs are within design limits. The licensee further stated that the operating transients will not change as a result of the MUR power uprate and no additional transients have been developed; therefore, the existing loads, stresses, and CUFs remain valid. Since all MUR conditions are bounded by the design conditions, the staff concludes that the existing stresses and CUFs for the CRDMs remain applicable for the MUR conditions, and that the existing CRDM design-basis analyses support the MUR power uprate.

Reactor Coolant Piping and Components

The RCS piping was designed to USAS B31.7, 1968 Edition. The licensee reviewed the revised design conditions for impact on the existing design basis analyses for the RCS piping and supports. The licensee stated that there is no change in RCS design or operating pressure, and the effects of operating temperature changes for cold and hot legs are within design limits. The MUR power uprate conditions are bounded by the design conditions. In addition, the operating transients will not change as a result of the power uprate and no additional transients have been proposed. The existing loads, stresses, and CUF values for RCS piping and supports remain valid for the proposed power uprate. The NRC staff finds that the reactor coolant piping and supports are acceptable for operation at MUR power uprate level.

The SGs were designed to the ASME Code, Section III, 1965 Edition through Summer 1967 Addenda. The licensee reviewed the revised design conditions for impact on the existing design basis analyses for the reactor coolant piping and supports. The licensee stated that there is no change in RCS design or operating pressure, and the effects of operating temperature changes for cold and hot legs are within design limits. The MUR power uprate conditions are bounded by the design conditions. In addition, the operating transients will not change as a result of the power uprate and no additional transients have been proposed. The existing loads, stresses, and CUF values for SGs remain valid for the proposed power uprate. The change in RCS mass flow is negligible. Also, since the design of the SGs included modeling of flow-induced vibration and the steam and FW flow rates remain bounded by the design flow rates, the licensee concluded that the power uprate will have no effect on flow-induced vibration. The licensee stated that the existing tube loads due to LOCA, main steam line break (MSLB), and FW line break will not change as a result of the power uprate.

The pressure retaining parts of the RCPs were designed in accordance with the ASME Code, Section III, 1968 Edition. The licensee reviewed the revised design conditions for impact on the existing design basis analyses for the reactor coolant pumps. The licensee stated that there is no change in RCS design or operating pressure, and the effects of operating temperature changes for cold and hot legs are within design limits. The MUR power uprate conditions are bounded by the design conditions. In addition, the operating transients will not change as a result of the power uprate and no additional transients have been proposed. The existing loads, stresses, and fatigue CUF values for RCPs remain valid for the proposed power uprate.

The code of record for the pressurizer, including the nozzles, is the ASME Code, Section III, 1965 Edition, with addenda through Summer 1967 Addenda. The licensee reviewed the MUR operating conditions for impact on the existing design basis analyses for the pressurizer. The licensee stated that there is no change in RCS design or operating pressure, and the effects of operating temperature changes for cold and hot legs are within design limits. The MUR power uprate conditions are bounded by the design conditions. In addition, the operating transients will not change as a result of the power uprate and no additional transients have been proposed. The existing loads, stresses, and CUF values for pressurizer remain valid for the proposed power uprate.

The licensee reviewed the potential for thermal stratification (NRC Bulletins 88-08 and 88-11). The changes in the RCS temperature and mass flow rate are negligible and the licensee concluded that the effects on thermal stratification will not change as a result of the power uprate.

Based on the above, the NRC staff agrees with the licensee’s conclusion that the design of piping, components, including the SGs, RCPs, and pressurizer, and their supports, is adequate to maintain the structural and pressure boundary integrity of the reactor coolant loop because the analyses of record parameters are bounding for the proposed power uprate condition.

HELB Locations

The licensee stated that an engineering evaluation was performed to determine the impact of power uprate on HELB systems inside and outside containment. The HELB evaluations were performed at 2619 MWt (102 percent of current RTP) to bound the expected range of operation resulting from the MUR uprate. There are no new line breaks postulated for current HELB systems inside or outside containment as pressures and temperatures did not increase. Also, there are no new systems that qualify as HELB systems as a result of the uprate. The NRC staff reviewed the licensee’s evaluation and concludes that the results are reasonable and acceptable regarding HELB.

BOP Piping

The BOP piping includes NSSS-interface systems, safety related cooling water systems, and containment systems. The licensee stated that the MUR uprate operating conditions were reviewed for impact on the existing design basis analyses for the RCS attached piping and supports. No changes in RCS design or operating pressure were made as a part of the power uprate. The effects of operating temperature changes for cold and hot legs are within design limits and the MUR power uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the licensee concluded that the existing loads, stresses, and CUF values remain valid.

The revised conditions were reviewed for impact on the existing design basis analyses for the main steam and main FW piping and supports. No significant changes in the SG design or operating pressure were made as a part of the power uprate. The changes in the operating temperatures and flow rates due to the MUR power uprate have been evaluated and were determined by the licensee to have a negligible effect on the existing design basis analyses. The existing loads and stresses remain valid.

The licensee concluded that the CR-3 BOP piping systems remain acceptable for operation at the uprated conditions. System pressures and temperatures remain within the design limits. Based on the above, the NRC staff agrees with the licensee's conclusion that the proposed 1.6 percent power uprate will not have adverse effects on BOP systems or safety-related valves.

Safety Related Valves

The design criteria for safety-related valves are promulgated in 10 CFR 50.55a. Additional information is also provided by the plant-specific evaluations of GL 89-10 and GL 96-05, as related to plant specific program for motor-operated valves, GL 95-07, as related to the pressure locking and thermal binding for safety-related gate valves, and the plant-specific evaluation of the GL 96-06 program regarding the over-pressurization of isolated piping segments.

The licensee stated that the MUR operating conditions were reviewed for impact on the existing design basis analyses for the safety related valves and showed that the temperatures are bounded by those used in the existing analyses. Safety analysis confirmed that the installed capacities and lift set points of the RCS and main steam relief valves to be valid for the MUR conditions. None of the safety related valves required a change to their design or operation as a result of the MUR. The existing loads, stresses, and fatigue CUF values remain valid. The licensee did not identify any changes to the plant-specific provisions of GL 89-10 and GL 96-05, GL 95-07, or GL 96-06. The NRC staff does not anticipate any changes to the analysis of over pressurization of isolated piping segments because the analysis of record for containment temperature and pressure was performed at 102 percent of current RTP and remains bounding for the uprate conditions. Therefore, the staff finds the performance of existing safety-related valves acceptable with respect to the MUR power uprate.

Summary

The NRC staff has reviewed FPC's assessment of the impact of the proposed MUR power uprate on NSSS and BOP systems and components with regard to stresses, CUFs, and safety related valve programs. On the basis of this review described above, the NRC staff finds that the proposed MUR power uprate will not have an adverse impact on the structural integrity of the piping systems, components, their supports, reactor internals, core support structures, CRDMs, BOP piping, or safety-related valves.

3.5 Reactor Vessel Integrity

The staff's review in the area of RV and RV internals integrity focuses on the impact of the proposed MUR power uprate on pressurized thermal shock (PTS) calculations, fluence evaluations, heatup and cooldown pressure-temperature (P-T) limit curves, low-temperature overpressure protection, upper-shelf energy (USE), surveillance capsule withdrawal schedules, and RV internals. This review was conducted, consistent with the guidance contained in RIS 2002-03 to verify that the results of licensee analyses related to these areas continue to meet the requirements of 10 CFR 50.60, 10 CFR 50.61, 10 CFR 50.55a, and 10 CFR Part 50, Appendices G and H, following implementation of the proposed MUR power uprate.

RV Material Surveillance Program

The RV material surveillance program provides a means for determining and monitoring the fracture toughness of the RV beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the RV. Appendix H to 10 CFR Part 50 provides the staff's requirements for the design and implementation of the RV material surveillance program.

By letter dated June 5, 2002 (ML021640547), the licensee requested a 0.9 percent power uprate to 2568 MWt. This was reviewed and approved by NRC staff in an SE dated November 1, 2002 (ML023050463). The licensee calculated fluence values based on continued operation at 2544 MWt (the then-current RTP) and increased these values by 7 percent to bound operation at 2568 MWt (the proposed 0.9 percent power uprate). The staff concluded that the 7 percent increase was a conservative assumption and was acceptable.

The licensee retained these fluence values for the currently proposed power uprate, noting that an assessment in 2006 concluded that the actual 32 effective full power year (EFPY) fluence values would only increase 2 percent due to the MUR power uprate. The staff confirmed that the licensee used methods acceptable to the staff and used conservative assumptions for determining the 7 percent increase in fluence. The staff has concluded that the fluence values remain bounding with the 1.6 percent power uprate.

The licensee's RV material surveillance program is an integrated program designed by the B&W Owners Group, which is now a part of the PWR Owners Group (PWROG), for all seven operating B&W-designed 177-fuel assembly plants and six participating Westinghouse plants having B&W-fabricated reactor vessels. The program, which contains capsule withdrawal schedules, is revised periodically. The most recent version, documented in BAW-1543 (NP), Revision 4, Supplement 6, "Supplement to the Master Integrated Reactor Vessel Surveillance Program [MIRVSP]," was approved by the NRC in an SE dated June 28, 2007. This SE stated that the proposed withdrawal schedules satisfy the American Society for Testing and Materials Standard E185-82 for all plants participating in this PWROG MIRVSP, except for Turkey Point, Units 3 and 4. Table IX of Supplement 6 indicated that the peak end-of-license (EOL), i.e., 32 EFPY, inside diameter (ID) fluence for CR-3 is 8.03×10^{18} n/cm^2 (E>1.0 MeV). The fluence values considering MUR power uprate were reported in FRA-ANP 32-5013936-00, "Adjusted Reference Temperature [ART] for 32 EFPY for CR-3 Power Uprate."

The staff determined that the small changes of the EOL ID fluences will not have sufficient impact on the ARTs and, therefore, on the capsule withdrawal schedule of MIRVSP approved by the NRC on June 28, 2007. Therefore, the staff determined that the CR-3 RV surveillance program would continue to meet the requirements of 10 CFR Part 50, Appendix H, under the MUR power uprate condition.

P-T Limits and USE

Appendix G to 10 CFR Part 50 provides fracture toughness requirements for ferritic (low alloy steel or carbon steel) materials in the reactor coolant pressure boundary (RCPB), including requirements on the USE values used for assessing the safety margins of the RV materials against ductile tearing and for calculating P-T limits for the plant. These P-T limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests. The staff's review of the USE assessments covered the impact of the MUR power uprate on the neutron fluence values for the RV beltline materials and the USE values for the RV materials through the end of the current licensed operating period. The NRC staff's P-T limits review covered the P-T limits methodology and the calculations for the number of the EFPY specified for the proposed MUR power uprate, considering neutron embrittlement effects.

Regarding the P-T limits, the licensee concluded in Attachment D Section 4.2.3.3 of the June 28, 2007, submittal that:

> The current P-T Limit curves are licensed through 32 EFPY and are based on adjusted reference temperatures (ART) at the ¼-thickness (1/4T) and ¾-thickness (3/4T) wall locations for the limiting reactor vessel beltline material. The impact of the MUR power uprate on the P-T curves was assessed by performing a revised 32 EFPY ART calculation in accordance with Regulatory Guide 1.99, Revision 2, which considered recent reactor vessel surveillance data and an assumed 7 percent increase in 32 EFPY fluence due to a power uprate.

The previous power uprate resulted in ART values at 32 EFPY of 195.7 °F and 144.1 °F for limiting welds SA-1769 and WF-8/WF-18, at 1/4T and 3/4T, respectively. The staff confirmed that the ID fluence is consistent with the fluence cited in the SE for the previous power uprate, which also approved the current CR-3 P-T limits. Since the MUR power uprate fluence is bounded by the current P-T limit fluence at 32 EFPY, the MUR power uprate has no impact on the current P-T limit curves. Hence, the staff confirmed that the CR-3 RV materials would continue to meet the requirements of 10 CFR Part 50, Appendix G, under the MUR power uprate condition.

Regarding the topic of the RPV USE, the licensee concluded in Section 4.2.3.5 of Attachment D to the June 28, 2007 submittal that:

> An equivalent margin assessment was performed (on welds WF-70 and WF-8/WF-18) in a 1994 B&W Owners Group generic analysis. The welds were evaluated for ASME [Code] levels A, B, C and D Service Loadings based on the evaluation acceptance criteria of ASME Section XI, Code Case N-512, which later became ASME Section XI, Appendix K.
>
> The analysis demonstrated that the limiting RV beltline welds at CR-3 satisfy the ASME Code requirements of ASME Code Case N-512 (ASME Section XI, Appendix K) for ductile flaw extensions and tensile stability using projected low upper-shelf Charpy impact energy levels for the weld material at 32 EFPY considering fluence which bounds the MUR power uprate.

The NRC staff has evaluated the information provided by the licensee in the submittal as well as information regarding the equivalent margins analysis contained in BAW-2192-PA, "Low Upper-Shelf Fracture Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level A & B Service Loads," and BAW-2178-PA, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners

Reactor Vessel Working Group for Level C&D Service Loads." The equivalent margins analysis was based on EOL fluences and chemistry values from BAW-2192-PA and BAW-2178-PA, and are documented in Appendix F of the September 13, 2007 RAI response. For weld WF-70, the equivalent margins analysis was based on an ID fluence of 8.22 X 10^{18} n/cm^2 (E>1.0 MeV), copper chemical composition value of 0.35 weight percent (w/o), and a nickel value of 0.59 w/o. Data from the CR-3 power uprate in 2002 are an ID fluence of 8.27 X 10^{18} n/cm^2 (E>1.0 MeV), 0.32 w/o copper, and 0.58 w/o nickel. For welds WF-8/WF-18, the equivalent margins analysis was based on an ID fluence of 7.96 X 10^{18} n/cm^2 (E>1.0 MeV), 0.20 w/o copper, and 0.55 w/o nickel. Data from the CR-3 power uprate in 2002 are an ID fluence of 7.92 X 10^{18} n/cm^2 (E>1.0 MeV), 0.19 w/o copper, and 0.57 w/o nickel. Due to the slight differences in fluence values, and decreases in copper chemical composition values, the staff concludes that the USE would be bounded by the current equivalent margins analysis, and that the CR-3 RV materials would continue to meet the USE criteria requirements of 10 CFR Part 50, Appendix G, following the MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the P-T limits and USE.

PTS

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

The licensee provided the RT_{PTS} value in Attachment F to the September 13, 2007 submittal. The RT_{PTS} value for the limiting beltline material, weld WF-8/WF-18, is 206.0 °F at 32 EFPY, using an ID fluence of 7.92 x 10^{18} n/cm^2 (E>1.0 MeV) based on a fluence projection from the previous 2002 power uprate. The screening criterion for this weld metal is 300°F. Therefore, the RV will remain within its limits for PTS after the MUR power uprate.

The staff confirmed that the CR-3 RV materials would continue to meet the PTS screening criteria requirements of 10 CFR 50.61.

RV Internals and Core Support Materials

The RV internals and core supports perform safety functions or whose failure could affect safety functions performed by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the RCPB). The NRC's acceptance criteria for RV internals and core support materials are based on GDC-1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of RV internals and core supports. Matrix 1 of NRC RS-001, Revision 0, "Review Standard for Extended Power Uprates," refers to the NRC's approval of the recommended guidelines for RV internals in Topical Reports WCAP-14577, Revision 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals" (March 2001), and BAW-2248-A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" (March 2000).

The licensee discussed the impact of the MUR power uprate on the structural integrity of the CR-3 RV internal components in Attachment F of the September 13, 2007 RAI response. The licensee concluded that the temperature changes due to the MUR power uprate are minimal and therefore, the existing loads remain valid and the stresses and fatigue values also remain valid.

The RV internals of pressurized water reactors (PWRs) reactors may be susceptible to the following aging effects:

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

Table Matrix 1 of RS-001 provides the staff's basis for evaluating the potential for extended power uprates to induce these aging effects. Depending on the magnitude of the RV internals fluence, Table Matrix 1 may be applicable to the current MUR power uprate application. In Note 1 to Table Matrix 1, the staff stated that guidance on the neutron irradiation-related threshold for IASCC for PWR RV internals are given in BAW-2248-A and WCAP-14577, Revision 1-A. This Table Matrix 1 note further stated that, for thermal and neutron embrittlement of CASS, SCC, and void swelling, licensees will need to provide plant-specific degradation management programs or participate in industry programs that investigate degradation effects and determine appropriate management programs.

In the RAI response dated September 13, 2007, the licensee stated that it is an active participant of the Electric Power Research Institute (EPRI) Materials Reliability Project (MRP) Internals Focus Group, which is working to establish generic inspection and evaluation guidelines for PWR internals based, in part, on fluence levels (the focus is a plant's license renewal period). The licensee also stated that it is developing a CR-3-specific RV internals inspection program considering CR-3-specific parameters, including MUR conditions, based on the EPRI MRP recommendations. Since fluence values are unchanged from the previous 2002 power uprate and changes in operating temperatures are minimal, staff concludes that the MUR power uprate is not expected to have any significant impact on irradiation-related aging degradation of the RV internals.

In summary, the staff has reviewed the licensee's LAR to increase the RTP by 1.6 percent and has evaluated the impact that the uprated conditions will have on the structural integrity assessments for the RV and its internals. The staff has determined that the changes proposed in the LAR will not impact the remaining safety margins required for the following structural integrity assessments: (1) RV surveillance program; (2) RV USE assessment; (3) P-T limits; (4) PTS assessment; and (5) RV internals. Therefore, the staff finds the proposed power uprate acceptable with respect to the structural integrity of the RV and its internals..

3.6 Chemical Engineering and Steam Generators

The staff has reviewed the proposed request with respect to: (1) chemical and volume control system (CVCS), (2) SG blowdown system, (3) flow accelerated corrosion (FAC), (4) coatings, and (5) SG program.

CVCS

The CVCS provides a means for (1) maintaining water inventory and quality in the RCS, (2) supplying seal-water flow to the reactor coolant pumps and pressurizer auxiliary spray, (3) controlling the boron neutron absorber concentration in the reactor coolant, (4) controlling the primary-water chemistry and reducing coolant radioactivity level, and (5) supplying recycled coolant for demineralized water makeup for normal operation and high-pressure injection flow to the emergency core cooling system in the event of postulated accidents. The staff has reviewed the safety-related functional performance characteristics of CVCS components based on (1) GDC-4, "Reactor Coolant Pressure Boundary (RCPB)," as it requires that the RCPB be designed to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture; and (2) GDC-29, "Protection Against Anticipated Operational Occurrences," as it requires that the reactivity control systems be designed to assure an extremely high probability of accomplishing their functions in the event of condenser in-leakage or primary-to-secondary leakage. Specific review criteria are contained in SRP Section 9.3.4, "Chemical and Volume Control System (PWR)."

Under power uprate conditions, the licensee indicated that the hot-leg and cold-leg temperatures will change by 0.4 degrees Fahrenheit. This will result in a slightly lower temperature for the letdown line since the CVCS system takes suction from the cold-leg. The licensee concluded that the slightly lower temperature of the letdown line does not affect the performance of the letdown coolers because they remain bounded by current operation. In addition, the licensee reported that the MUR conditions do not result in changes to the makeup requirements.

The staff reviewed the information provided by the licensee and concluded that the CVCS is adequate for the uprated conditions. The proposed MUR power uprate will introduce negligible changes in the CVCS operating parameters and the system will continue to operate within its design limits.

SG Blowdown System (SGBS)

Control of secondary-side water chemistry is important for preventing degradation of SG tubes. The SGBS system provides a means for removing SG secondary-side impurities and, thus, assists in maintaining acceptable secondary-side water chemistry in the SGs. The design basis of the SGBS includes consideration of expected design flows for all modes of operation. The staff reviewed the ability of the SGBS to remove particulate and dissolved impurities from the SG secondary-side during normal operation, including condenser in-leakage and primary-to-secondary leakage. The NRC's acceptance criteria for the SGBS are based on GDC-14 as it requires that the RCPB be designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture. Specific review criteria are contained in SRP Section 10.4.8, "Steam Generator Blowdown System (PWR)."

The licensee indicated that CR-3 does not require continuous SG blowdown. Steam generator blowdown is used to achieve secondary chemistry limits on restart from outages and is typically terminated at approximately 20-percent reactor thermal power. The licensee concluded that the SGBS would not be impacted at 100-percent reactor thermal power.

On the basis of its review, the staff concludes that the SGBS remains adequate for power uprate conditions because the blowdown flow, the SG secondary-water chemistry, and the blowdown pressures and temperatures remain within the original system design. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the SGBS.

FAC Program

FAC is a corrosion mechanism occurring in carbon steel components exposed to single-phase or two-phase flow. Components made from stainless steel are immune to FAC, and FAC is significantly reduced in components containing small amounts of chromium or molybdenum. The rates of material loss due to FAC depend on flow velocity, fluid temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and the optimum conditions for minimizing FAC effects, in most cases, cannot be achieved. Loss of material by FAC is, therefore, likely to occur. The staff reviewed the effects of the proposed MUR power uprate on FAC and the adequacy of the licensee's FAC program to predict the rate of loss so that repair or replacement of components can be made before the loss of material results in the components reaching a minimum thickness.

The licensee indicated that it performed a FAC evaluation to identify limiting components. The FW risers were determined to be the most impacted by the power uprate conditions. The FW risers are scheduled to be replaced in 2009. The licensee also indicated that the CR-3 FAC program is updated on a continuous basis. The results of Refueling Outage 15 inspections will be incorporated into the FAC program. The licensee concluded that the impacts on wear rates and service lives are not expected to be significant due to the small increase in flow rate and temperature.

On the basis of its review, the staff concludes that the FAC program is acceptable for operation conditions because the effect of the power uprate on the parameters that affect FAC rates is expected to be small and will be adequately managed by the existing FAC program. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the FAC program.

Coatings

Protective coatings (paints) inside containment are used to protect equipment and structures from corrosion and contamination from radionuclides, and also provide wear protection during plant operation and maintenance activities. The coatings are subject to 10 CFR Part 50 Appendix B quality assurance requirements because their degradation can adversely impact safety-related equipment. The staff reviewed whether the pressure and temperature conditions under the power uprate continue to be bounded by the conditions to which the coatings were qualified.

The licensee indicated that the design basis accident temperature and pressure profiles are not changing due to power uprate conditions, and therefore there will be no impact on the coatings program.

On the basis of its review, the staff concludes that the coatings will not be adversely impacted by the MUR power uprate temperature and pressure conditions as they will continue to be bounded by the conditions to which the coatings were qualified. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the coatings program.

SG Program

SG tubes constitute a large part of the RCPB. The staff reviewed the effects of changes in operating parameters (e.g., pressure, temperature, and flow velocities) resulting from the proposed power uprate on the design and operation of the SGs. Specifically, the staff evaluated whether changes to these parameters continue to be bounded by those considered in the plant design and licensing basis (i.e., the TS tube plugging limits).

CR-3 has two B&W once-through SGs. Each steam generator contains 15,531 stress-relieved, mill-annealed, Alloy 600 tubes. Each tube has a nominal outside diameter of 0.625 inches and a nominal wall thickness of 0.034 inches. The tubes were mechanically roll expanded in both the hot and cold-leg tubesheet for approximately 1 inch of the 24 inches thick tubesheets. The tubes are supported by a number of carbon steel support plates.

The licensee evaluated all post-uprate system parameters in the existing SG analyses and concluded that the SGs will continue to satisfy all original design criteria under power uprate conditions. The licensee performed an evaluation to address flow-induced vibration (FIV) and its impact on the SG tube bundle and installed tube repair hardware for the MUR conditions. The licensee concluded that the tube bundle will not fail due to high-cycle fatigue, tube-to-tube impacts will not occur over the life of the plant, and all installed tube repair hardware will maintain functional integrity. The licensee also confirmed that the plugging limit continues to be appropriate for power uprate conditions, according to the guidance in RG 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes."

On the basis of its review, the staff concludes that the power uprate is acceptable from a SG design and inservice inspection perspective because the power uprate is expected to introduce only negligible changes in the SG operating parameters and the SGs will continue to operate within their design limits. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the SGs.

3.7 Human Factors

The staff reviewed the licensee's human factors evaluation to determine if it conforms to the NRC staff's guidance in Section VII of RIS 2002-03. RIS 2002-03 provides guidance to the licensee in evaluating the need for changes to the areas of operator manual actions, procedures, human-system interfaces, and operator training related to the MUR power uprate. The staff's human factors evaluation was conducted to confirm that operator performance would not be adversely affected as a result of system and procedure changes made to implement the proposed MUR power uprate.

The NRC staff has developed a standard set of questions for human factors reviews in RIS 2002 03, Attachment 1, Section VII, Items 1 through 4). The following sections evaluate the licensee's response to these questions in the LAR and additional clarifications in its RAI response.

Operator Manual Actions

The licensee stated in the submittal that no new operator manual actions or changes to existing operator manual actions will be required for to the emergency operating procedures (EOPs) or abnormal operating procedures (AOPs) as a result of the proposed MUR power uprate. The licensee provided a table in the RAI response letter that included a list of existing operator manual actions credited in CR-3 FSAR that were reviewed for potential changes due to the proposed MUR. The existing operator manual actions and the corresponding response times in the table were found to be unaffected by the increase in power level for the FSAR Chapter 14 events. More specifically, events such as ATWS and SBO did not have specific operator manual actions or any changes to the response times in the EOPs at the increased power level.

The NRC staff has reviewed the licensee's statements in the original submittal and responses to the RAI relating to any impacts of the MUR power uprate to existing or new operator manual actions. The NRC staff concludes that the proposed MUR power uprate will not have any impact on the overall existing operator manual actions and their response times.

Emergency and Abnormal Operating Procedures

The licensee reviewed the EOPs and AOPs for potential changes related to the proposed MUR power uprate. The licensee proposed to adjust existing power level values in the EOPs and AOPs for stabilizing reactor power in the event of main FW pump failures. An example of this adjustment is when a trip of the main FW booster pump occurs with four reactor coolant pumps running, the reactor power is required by the EOPs to be reduced to 52 percent power. With the proposed increase in power level to 2609 MWt, the limit will now be reduced to 50 percent. The licensee also restated that no new operator manual actions will be included in the EOPs and AOPs due to the proposed MUR power uprate and that the power level changes in the EOPs and AOPs will be incorporated in the normal operator training cycles prior to the implementation of the MUR power uprate.

The NRC staff has reviewed the licensee's evaluation of the effects of the MUR power uprate on CR-3 EOPs and AOPs. The NRC staff concludes that the proposed MUR power uprate does not present any adverse impacts on the EOPs and AOPs. This conclusion is based upon the licensee making revisions to the EOPs and AOPs that will reflect the new power level. The minor changes being made to the EOPs and AOPs will be reflected in the operator training program prior to MUR implementation.

Control Room Controls, Displays, and Alarms

In its submittal, the licensee described changes to control room controls, displays (including the Safety Parameter Display System), and alarms related to the proposed MUR power uprate. The licensee also provided supplemental information related to these changes in the RAI response. Notable proposed changes to controls, displays, and alarms include:

- The AULD subsystem will be modified to provide both existing and revised calorimetric values for the control room displays. Operators will use the modified AULD display to select which calorimetric will be used in the AULD as the controlled parameter used to establish thermal demand. The AULD display, in conjunction with the plant annunciator, will also alert the operators when the AULD has automatically transferred out of automatic upon the detection of a sufficient differential in the available secondary heat balance calculations. Operators will not be allowed by the AULD logic to return the AULD to automatic using the improved calorimetric as long as this differential exists. The AULD will be capable of automatically controlling the plant between 2568 MWt and 2609 MWt for the appropriate plant parameters.

- The Caldon LEFM system will be installed with alarm functions displayed throughout the AULD and the plant computer system in the control room. The alarm will alert operators when the LEFM system has self-diagnosed a condition that has resulted in an internal alert or failure. Operators will monitor control room displays and indicators along with making routine checks of local indications in the plant in the event of LEFM failures.

- The functions for the Fixed Incore Monitoring Detector System (FIDMS) will require changes to the plant computer software. These changes will be transparent to the operators and their responses to abnormal indications by the software will remain unchanged.

- Changes to the NI calibrations due to MUR power uprate will also be accommodated by corresponding changes to the Integrated Control System. The change in NI calibration will have no effects on any control room controls or the operator's ability to monitor core power production, and thus have no adverse effect on the CR-3 existing defense-in-depth or safety margins.

The NRC staff inquired in the RAI on how the proposed software changes to the AULD and FIDMS will be validated for no adverse effects to the operators after implementation of the proposed MUR power uprate. The licensee responded that the software for both systems would be thoroughly tested in information technology laboratory settings and the plant simulator. The testing would be conducted using the licensee's existing engineering change processes, which encompasses modifications being made to plant software and digital controls. The stated that any changes to the AULD and FIDMS will be validated, tested, and incorporated into the operator training program prior to implementation of the proposed MUR power uprate.

The NRC staff has reviewed the licensee's evaluation and proposed changes to the control room. The NRC staff concludes that the proposed changes do not present any adverse effects to the operators' functions in the control room.

Control Room Plant Reference Simulator and Operator Training

The licensee stated that the plant simulator changes will be made prior to the plant modifications, but kept in a separate software package to avoid impacting the initial operator training program. The software package will utilize the plant simulator for the existing operating crews on the modifications related to the MUR power uprate during a just-in-time training. The simulator performance will be verified for consistency of the intended plant design after MUR power uprate implementation by conducting a formal integrated simulator acceptance test. The acceptance test will serve as the verification portion of simulator testing as described in CR-3 operator training procedures. After plant modifications are made throughout the plant, the plant simulator will be modified to reflect any parameter changes that are slated by the engineering change process. The plant simulator will be physically modified from the original design to reflect the MUR power uprate based changes. The licensee will also calibrate the plant simulator data to the results of the actual plant data upon implementation or the MUR power uprate, which will validate the plant simulator testing. The results of the complete simulator change due to the MUR power uprate will be maintained as a simulator work package as a reference for future changes that are made to the plant simulator.

The licensee will also modify the operator training program to include the changes made to the EOPs and AOPs, control room components, and plant simulator modifications prior to the implementation of the MUR power uprate. The licensee also plans to develop and complete operator training on the new Caldon LEFM system prior to MUR power uprate implementation.

The NRC staff has reviewed the licensee's proposed changes to the operator training and plant simulator related to the MUR power uprate. The NRC staff concludes that the changes do not present any adverse effects on the plant simulator or the operator training program. The licensee committed to making the modifications to the plant simulator and incorporating these changes into the operator training program prior to MUR power uprate implementation.

Summary

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

3.8 Dose Consequences Analysis

The NRC staff evaluated the potential impact of the MUR power uprate on the results of the CR-3 dose consequence analyses, guided by Sections II and III of Attachment 1 to RIS 2002-03. The review is conducted to verify that dose consequences continue to meet the acceptance criteria in 10 CFR 50.67 and GDC 19 following implementation of the MUR power uprate.

Previously, in Amendment 199 of September 17, 2001 (ML012430210), CR-3 was granted implementation of a full-scope alternative source term in accordance with 10 CFR 50.67, and following the guidance of Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors." Therefore, this evaluation has been conducted to verify that the results of the licensee's DBA radiological dose consequence analyses continue to meet the dose acceptance criteria given in 10 CFR 50.67 for offsite doses and GDC-19 (or equivalent for plants licensed before the GDC were in existence) with respect to control room habitability. The applicable acceptance criteria are 5 rem Total Effective Dose Equivalent (TEDE) in the control room (CR), 25 rem TEDE at the exclusion area boundary, and 25 rem TEDE at the outer boundary of the low population zone. Except where the licensee proposed a suitable alternative, the staff utilized the regulatory guidance provided in applicable sections of RG 1.183, Chapter 15 of the SRP for DBAs, and Chapter 6.4 of the SRP for CR habitability, in performing this review.

The staff reviewed the regulatory and technical analyses performed by the licensee in support of its proposed MUR power uprate license amendment, as they relate to the radiological consequences of DBA analyses. Information regarding these analyses was provided by the licensee in Attachments A and D of the submittal dated April 25, 2007. The licensee stated that each of the current DBA dose analyses of record for CR-3 that depends on core power level were performed at a 2619 MWt, or 102 percent of the current RTP. Therefore, the current analyses bound any analyses that would be performed at the proposed MUR uprated power level.

Using the current licensing basis documentation, as contained in the current CR-3 FSAR, in addition to information provided by the licensee to support the MUR license amendment, the staff verified that the existing CR-3 FSAR Chapter 14 radiological analyses source term and release assumptions bound the conditions for the proposed 1.6 percent power uprate to 2609 MWt, considering the higher accuracy of the proposed FW flow measurement instrumentation. The specific DBA analyses that were reviewed were as follows:

- MSLB Accident
- Steam Generator Tube Rupture Accident
- Fuel Handling Accident
- Control Rod Ejection Accident
- LOCA
- Letdown Line Break Accident
- Waste Gas Decay Tank Rupture Accident

In summary, the staff reviewed the assumptions, inputs, and methods used by the licensee to reassess the radiological consequences of the postulated DBA with the proposed uprated power level. The staff finds that the licensee will continue to meet the applicable dose acceptance criteria following implementation of the proposed 1.6 percent MUR power uprate. The staff further finds reasonable assurance that CR-3 will continue to provide sufficient safety margins, with adequate defense-in-depth, to address unanticipated events and to compensate for uncertainties in accident progression, analysis assumptions, and input parameters. Therefore, the proposed license amendment is acceptable with respect to the radiological dose consequences of the design basis accidents.

3.9 Fire Protection

The purpose of the fire protection program is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary safe plant shutdown functions and will not significantly increase the risk of radioactive releases to the environment. The NRC staff's review focused on the effects of the increased decay heat on the plant's safe-shutdown analysis to ensure that credited SSCs will continue to be able to achieve and maintain safe shutdown following a fire.

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

The licensee re-evaluated the applicable SSCs and safety analyses at the proposed MUR core power level of 2609 MWt against the previously analyzed core power level of 2568 MWt. The staff reviewed the impact of proposed MUR power uprate on the results of safe-shutdown fire analysis as noted in RIS 2002-03, Attachment 1, Sections II and III. The review focused on the effects of MUR power uprate on the post-fire safe-shutdown capability and increase in decay heat generation following plant trips. In the LAR, the licensee stated that the current accidents and transients safe-shutdown fire analysis of record for CR-3 were unaffected by the requested power uprate because they were performed assuming 102 percent of 2568 MWt.

The results of the Appendix R evaluation are provided in Table D.2-1 of the June 28, 2007, submittal. The licensee indicated that the natural circulation cooldown time will increase slightly based upon the power uprate from 2568 MWt to 2609 MWt. The time to cool the plant to 200 degrees Fahrenheit will increase from 68.54 hours to 70.38 hours, which is still less than the 72 hour requirement in 10 CFR 50 Appendix R. Further, the licensee indicated that additional heat in the intermediate building from the MUR power uprate will not prevent required manual actions from occurring at their designated time.

The increases in decay heat from MUR power uprates usually do not affect the elements of a fire protection program related to (1) administrative controls, (2) fire suppression and detection systems, (3) fire barriers, (4) fire protection responsibilities of plant personnel, or (5) procedures and resources necessary for the repair of systems required to achieve and maintain cold shutdown. In addition, an increase in decay heat will usually not result in an increase in the potential for a radiological release resulting from a fire.

The staff has reviewed the licensee's fire-related safe-shutdown assessment and concludes that the licensee has adequately accounted for the effects of the increased decay heat on the ability of the required systems to achieve and maintain safe-shutdown conditions. The licensee's evaluation did not identify changes to design or operating conditions that will adversely impact the post-fire safe-shutdown capability. MUR uprate does not change the credited equipment necessary for post-fire safe-shutdown nor does it reroute essential cables or relocate essential components/equipment credited for post-fire safe-shutdown. The licensee has made no changes to the plant configuration or combustible loading as a result of modifications necessary to implement the MUR power uprate that affect the CR-3 fire protection program. The staff further concludes that the implementation of the proposed MUR power uprate does not affect compliance with the fire protection and safe-shutdown program. Therefore, the staff finds the proposed MUR power uprate acceptable with respect to fire protection.

3.10 Plant Systems

The NRC staff's review in the area of plant systems covers the impact of the proposed MUR power uprate on NSSS interface systems, containment systems, safety-related cooling water systems, spent fuel pool (SFP) storage and cooling, radioactive waste systems, and engineered safety feature (ESF) heating, ventilation, and air conditioning (HVAC) systems. The staff's review is based on the guidance in SRP Chapters 3, 6, 9, 10, and 11, and RIS 2002-03, Attachment 1, Sections II, III, and VI. The licensee evaluated the effect of the MUR on the plant systems. This evaluation is reflected in Section 6 of Attachment D of the application dated June 28, 2007.

NSSS Interface Systems

The NSSS interface systems include the main steam (MS) system, the atmospheric dump valves (ADVs) and turbine bypass valves (TBVs), the condensate system (CD), and main FW system.

The MS system provides isolation of the SGs after a steam line failure, provides overpressure relief and/or decay heat removal during accidents, and provides steam to the emergency feedwater (EF) system. For the MS system, the licensee stated that, following the MUR power uprate, there will be a slight increase in steam flow but MS system will continue to operate within its design parameters.

The ADVs provide a controlled path for venting steam to the atmosphere. For the ADVs, the licensee evaluated the valves for their functions to (1) close to isolate containment, (2) open and modulate to feliefe steam to the atmosphere, and (3) maintain pressure boundary. Since the power uprate conditions are bounded by the existing design conditions, the licensee determined that the functional performance of the ADVs will be unaffected by the power uprate.

The TBVs' primary function is to maintain stable turbine header pressure during load swings. The licensee determined that the steam flow rate is not changing significantly for the power uprate, and the system parameters are bounded by the existing design conditions of 102 percent RTP. Therefore, the TBVs will be unaffected by the MUR power uprate.

The CD system supplies preheated condensate to the FW system. For the CD system, the licensee evaluated the performance of the system and determined it was acceptable for operation at 2609 MWt. The condenser limiting back pressure is 9 inches of mercury absolute and the current maximum operating pressure is 3 to 4 inches of mercury absolute. Therefore, the licensee determined that the CD system is not impacted by the MUR power uprate.

The main FW system provides FW during normal operation and isolates during accidents. The licensee determined that the safety functions are not impacted by the uprate. The FW pumps, booster pumps, and heaters have adequate margin for the MUR uprate, and all parameters remain within the design limits. Therefore, the licensee determined that the FW system is not impacted by the uprate.

The staff reviewed the licensee's evaluation and concurs with the results. The licensee determined that there is no adverse impact on the NSSS interface systems from the MUR power uprate because there is sufficient operating margin to produce an additional 1.6 percent power, and all equipment will be operated within its design limits. The staff does not anticipate that an MUR power uprate will challenge the NSSS interface systems, and all systems have been shown to be operating within design. Therefore, the staff finds that the NSSS systems are acceptable for the MUR uprate.

Containment Systems

The containment systems include the containment building spray system, penetrations, and hatches. The spray system removes fission products form the post-accident containment atmosphere and assists in post-accident temperature and pressure control. The penetrations and hatches maintain structural integrity. As discussed in Section 3.3 of this SE, the containment response analyses to both LOCA and MSLB were evaluated using mass and energy release based on 102 percent of current RTP. These analyses are bounding for the MUR power uprate; therefore, the staff finds the containment systems acceptable for the MUR power uprate.

Safety-Related Cooling Water Systems

The safety-related cooling water systems include the decay heat closed cycle cooling (DC) system, the nuclear services closed cycle cooling (SW) system, the nuclear services and decay heat seawater (RW) system, and the EF system.

The DC system removes heat from the core via the low pressure injection/decay heat system. The DC system also cools various pumps that run post-accident. Heat is transferred to the RW system. Similarly, the SW system removes heat from various equipment that runs post-accident, and transfers the heat to the RW system. The RW system cools the SW and DC systems. For these three systems, the licensee confirmed that the applicable analyses were performed at 102 percent of current RTP and remain bounded by the MUR power uprate.

The EF system provides FW flow in the event of a loss of the main FW system. Since the CR-3 accident analyses were performed at 102 percent of current RTP, the licensee concluded that the EF system is not impacted by the MUR power uprate.

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

SFP Storage and Cooling

The SFP storage and cooling systems are described in Section 9.3 of the CR-3 FSAR. The principal function is to provide storage and cooling of the spent fuel. The primary impact of a power uprate would be to the decay heat of the fuel recently discharged from the core. The CR-3 FSAR describes the equipment available and the times after shutdown to reliably remove the decay heat from a full core offload and the fuel already in the SFP. The CR-3 FSAR also states that administrative controls assure that a fully capable backup is available within an appropriate timeframe, and that the capability and timely availability of the backup are determined based on specific conditions of the offload. The licensee concluded that the SFP cooling system is not impacted by the MUR power uprate. Based on the licensee's offload-specific evaluation, the staff finds that the SFP storage and cooling will not be impacted by the power uprate.

Radioactive Waste Systems

The CR-3 waste decay systems provide the means to sample, collect, process, store/hold, re-use, and/or release gaseous and liquid low-level effluents. The gaseous waste disposal system provides post-accident containment isolation and venting of excess gas from the reactor building. The licensee determined that this system is not impacted by the MUR, and that the WGDTR accident assumes the complete release of the maximum-allowed radionuclide inventory of all three tanks. The liquid waste disposal system processes liquid waste prior to release. The licensee determined that the volume of liquid waste is dependent on reactor coolant bleed-off, SG draindown, and leakage from various components, and will not change with the MUR power uprate. The licensee stated that the radionuclide concentration in the liquid may increase, but the result would be limited to a slight increase in the frequency of replacement of resins in the deionizers and this would not be a constraint to implementing the power uprate. The staff reviewed the licensee's assessment. The staff does not expect a 1.6 percent increase in power to result in a significant change to the operation of the radioactive waste systems; therefore, based on the licensee's assessment, the staff finds that the radioactive waste systems will function adequately for the MUR power uprate.

ESF HVAC Systems

The licensee evaluated the ESF HVAC systems and determined that the current design is based on102 percent of current RTP. Therefore, the licensee concluded that the safety functions of these systems are not impacted by the power uprate. Further, the post-accident containment response was evaluated for mass and energy releases that were based on 102 percent current RTP, so the MUR will not require any changes with respect to the containment air coolers or air flow rates. The staff reviewed the licensee's evaluation and, based on the systems being design to 102 percent current RTP, the staff concludes that the ESF HVAC systems are acceptable for the MUR power uprate.

Summary

In summary, the licensee reviewed the design and operation of the plant systems and determined that the proposed MUR power uprate does not adversely impact any of the system. For the reasons noted above, the staff concurs with the licensee's conclusion and finds that the plant systems will be acceptable for the MUR power uprate.

3.11 Changes to FOL and TSs

The maximum core power level is specified in the FOL and the definitions section of the TSs; therefore, the licensee proposed changes to these documents to reflect the increase in power level. The licensee also proposed changes to TS 3.3.1, "Reactor Protection System (RPS) Instrumentation."

The NRC's requirements related to the content of the TSs are set forth in 10 CFR 50.36, "Technical specifications." This regulation requires that the TSs include items in five specific categories. These categories include: (1) safety limits, limiting safety system settings and limiting control settings, (2) limiting conditions for operation (LCOs), (3) SRs, (4) design features, and (5) administrative controls.

Section 50.36(d)(2)(ii) of 10 CFR states that a TS LCO must be established for each item meeting one or more of the following criteria:

Criterion 1: Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of, or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of, or presents a challenge to the integrity of a fission product barrier.

Criterion 4: A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.

Section 50.36(d)(1)(ii)(A) of 10 CFR states, "Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is exceeded." Furthermore, Section 50.36(d)(3) states, "Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

The licensee proposed to change TS definition of RTP, and the FOL limit on "100 percent core power level," to specify that the maximum power level is 2609 MWt. The licensee proposed a corresponding change to the TS definition of Effective Full Power Days, which is the amount of energy produced in 24 hours of operation at RTP. The staff finds that these changes reflect the change in power level and are acceptable.

The licensee also proposed changes that result from the licensee's determination that the high-power trip, which is Function 1.a. of TS Table 3.3.1-1, needs to be reduced if the high-accuracy secondary heat balance (which consists of the UFM and other instrumentation) is not functional. The high-power trip is a safety limit related LSSS parameter that falls under 10 CFR 50.36(d)(1)(ii)(A). The safety analysis analytical limit does not support an AV of 104.9 percent RTP unless the high-accuracy heat balance is functional. Therefore, when the high-accuracy heat balance is not functional, the setpoint is reduced to 103.3 percent of RTP (103.3 percent of 2609 MWt is the same as the current setpoint, 104.9 percent of 2568 MWt). The licensee will also reduce core power to 2568 MWt when the high-accuracy calorimetric is not functional, to ensure that the core power level is within the analyzed limit.

The licensee proposed to add new Actions J and K to TS 3.3.1. Action J requires core power to be reduced to 2568 MWt within 12 hours and the high-power trip to be reduced to 103.3 percent RTP in 48 hours, if the secondary heat balance is not based on the high accuracy instrumentation. The licensee stated that these time frames allow for an orderly reduction of power and implementation of the setpoint change. The licensee indicated that the NI were compared to the last known good high-accuracy heat balance, and do not routinely require adjustments; therefore, the NI can continue to be relied upon for protection for the duration of the proposed completion time. In the setpoint calculations, the drift component in the setpoint is 0.399 percent over 30 months. As such, the expected setpoint drift over 48 hours is insignificant and the NI will remain calibrated for that time frame. Action K requires a reactor shutdown within 6 hours if Action J is not met. This timeframe is consistent with other TS required shutdowns. Therefore, the staff finds the proposed Actions J and K to be acceptable.

Every 24 hours, SR 3.3.1.2 requires the licensee to verify that the secondary heat balance is less than or equal to 2 percent RTP greater than the NI output. The licensee added a note to SR 3.3.1.2 to initiate entry into new Action J if the high accuracy instrumentation is not used for the secondary heat balance. The 2 percent criterion in SR 3.3.1.2 is not related to the accuracy of secondary heat balance; it is an allowed difference that is factored into the setpoint analyses for both cases. Therefore, the staff finds the proposed changes to SR 3.3.1.2 acceptable.

The proposed TSs allow for continued operation without the high-accuracy secondary heat balance. The licensee can use the other heat balance instrumentation as long as the core power and high-power trip setpoint are reduced in accordance with TS 3.3.1 Action J. The licensee proposed changes to TS Table 3.3.1-1 to include the AVs based on both sets of secondary calorimetric instrumentation, 104.9 percent and 103.3 percent. Notes clarify that the higher setpoint relies on the high-accuracy heat balance, and the lower setpoint is for conditions when the high-accuracy heat balance is not functional. The applicable in-plant setpoint and AV would be used for the periodic channel calibrations performed in accordance with SR 3.3.1.5. Further, LCO 3.0.4 would allow power ascension without the high-accuracy calorimetric as long as the licensee maintains the core power and high-power trip setpoint in accordance with TS 3.3.1 Action J.

As an additional change, the licensee proposed notes to Table 3.3.1-1 to clarify requirements during channel calibrations. These notes are intended to address the staff's concerns regarding channel operability and channel calibrations, as described in RIS 2006-17, as discussed in Section 3.2 of this SER. The proposed notes are as follows:

> If the as-found channel setpoint is conservative with respect to the Allowable Value (AV), but outside its predefined as-found acceptance criteria band, then the channel should be evaluated to verify that it is functioning as required before returning the channel to service. If the as found instrument channel setpoint is not conservative with respect to the AV, the channel shall be declared inoperable.
>
> The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the pre-established In-Plant Setpoint, or a value that is more conservative than the pre-established In-Plant Setpoint: otherwise the channel shall not be returned to OPERABLE status. The pre-established In-Plant Setpoint and the methodology used to determine the pre-established In-Plant Setpoint, the predefined as-found acceptance criteria band, and the as-left acceptance criteria are specified in the FSAR.

The staff reviewed the licensee's setpoint calculation and channel calibration procedure related to the high-power trip. The staff finds that the proposed notes adequately address the issue identified in RIS 2006-02 using the terminology that is consistent with the licensee's setpoint methodology. The license also committed to document in the CR-3 FSAR the methodology used to develop the AV, in-plant setpoint, and as-left and as-found criteria. Therefore, the staff finds the proposed changes to TS Table 3.3.1-1 acceptable.

The licensee also provided markups to the TS Bases. The NRC staff reviewed the Bases markups and verified that they adequately reflect the bases for the revised TSs. However, the TS Bases are controlled by the licensee's Bases Control Program and, therefore, they are not included in this amendment.

4.0 STATE CONSULTATION

Based upon a letter dated May 2, 2003, from Michael N. Stephens of the Florida Department of Health, Bureau of Radiation Control, to Brenda L. Mozafari, Senior Project Manager, U.S. Nuclear Regulatory Commission, the State of Florida does not desire notification of issuance of license amendments.

5.0 ENVIRONMENTAL CONSIDERATIONS

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (72 FR 51862). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

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