May 22, 2003

Mr. Michael Kansler, President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: INDIAN POINT NUCLEAR GENERATING UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: 1.4 PERCENT POWER UPRATE (TAC NO. MB6950)

Dear Mr. Kansler:

The Commission has issued the enclosed Amendment No. 237 to Facility Operating License No. DPR-26 for the Indian Point Nuclear Generating Unit No. 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated December 12, 2002, as supplemented on April 3 and May 2, 2003.

The amendment revises the Facility Operating License and the Technical Specifications to increase the licensed core thermal power level to 3114.4 megawatts (MWt), which is a 1.4-percent increase above the currently authorized power level of 3071.4 MWt. The power uprate is based on the improvement in the core power uncertainty allowance originally required for the emergency core cooling system (ECCS) evaluations performed in accordance with Appendix K, "ECCS Evaluation Models," to Part 50 of Title 10 of the *Code of Federal Regulations*. In addition, changes were made to TS Sections 1.1, 2.1, 2.3, 3.1, 3.4, 6.9, and the applicable TS Bases to account for the change in power level. Specifically, the reduced uncertainty is obtained by using a more accurate measurement of feedwater flow.

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

Sincerely,

/RA/

Patrick D. Milano, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-247

Enclosures: 1. Amendment No. 237 to DPR-26
2. Safety Evaluation

cc w/encls: See next page

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cc w/encls: See next page

TS(s): ML , Package: ML
ADAMS Accession Number: ML031420375 *SE Provided. No major changes made.

OFFICE	PDI-1/PM	PDI-1/PM	PDI-1/LA	SPSB/SC	SRXB/SC	SPLB/C	EMCB/C	EMEB/SC
NAME	NSanfilippo	PMilano	SLittle	MReinhart*	JUhle	JHannon	WBateman	KManoly
DATE	05/8/03	05/8/03	05/8/03	03/11/03	05/9/03		05/8/03	05/09/03
OFFICE	IEHB/C	EEIB/SC	EEIB/SC	RLEB/SC	OGC	PDI-1/SC	PDI/D	DLPM/D
NAME	TQuay	CHolden*	EMarinos	JTappert	RWeisman	RLaufer	CHolden	JZwolinski
DATE		04/21/03			05/20/03	05/22/03		

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DATED: May 22, 2003

AMENDMENT NO. 237 TO FACILITY OPERATING LICENSE NO. DPR-26 INDIAN POINT UNIT 2

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ENTERGY NUCLEAR OPERATIONS, INC.

DOCKET NO. 50-247

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 237
License No. DPR-26

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

   A. The application for amendment by Entergy Nuclear Operations, Inc. (the licensee) dated December 12, 2002, as supplemented on April 3 and May 2, 2003, 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-26 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 237, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days. Implementation shall include revisions to plant procedures for the operation and maintenance of the leading edge flowmeter (LEFM) Check system, including specific operator actions to be taken when the LEFM Check system is inoperable, as described in the licensee's December 12, 2002, application, and the NRC safety evaluation dated May 22, 2003.

FOR THE NUCLEAR REGULATORY COMMISSION

John A. Zwolinski, Director
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 22, 2003

ATTACHMENT TO LICENSE AMENDMENT NO. 237

FACILITY OPERATING LICENSE NO. DPR-26

DOCKET NO. 50-247

Replace the following page of the Facility Operating License with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove Page	Insert Page
3	3

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 marginal lines indicating the areas of change.

Remove Pages	Insert Pages
1-1	1-1
Figure 2.1-1	Figure 2.1-1
2.3-1	2.3-1
2.3-3	2.3-3
3.1.G-1	3.1.G-1
Table 3.4-1	Table 3.4-1
3.4-3	3.4-3
6-10	6-10

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 237 TO FACILITY OPERATING LICENSE NO. DPR-26

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

DOCKET NO. 50-247

1.0 INTRODUCTION

By letter dated December 12, 2002, as supplemented by letters dated April 3 and May 2, 2003, Entergy Nuclear Operations, Inc. (the licensee) submitted a request for changes to the Indian Point Nuclear Generating Unit No. 2 (IP2) Technical Specifications (TSs). The requested changes would revise the facility operating license and the TSs to reflect a 1.4-percent (%) increase in the reactor core thermal power level from 3071.4 megawatts thermal (MWt) to 3114.4 MWt. The April 3 and May 2 letters provided clarifying information that did not enlarge the scope of the original *Federal Register* (FR) notice or change the initial proposed no significant hazards consideration determination.

1.1 Background

Reactor core thermal power is computed by a nuclear steam supply system (NSSS) energy balance (calorimetric) calculation. The reliability of this calculation depends primarily on the accuracy of feedwater flow, temperature, and pressure measurements. Because the measuring instruments have measurement uncertainties, margins are included to ensure the reactor core thermal power levels do not exceed safe operating levels.

When the initial IP2 operating license was issued in 1973, Appendix K, “ECCS [emergency core cooling system] Evaluation Models,” to Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR) required licensees to assume a 2.0% measurement uncertainty for the reactor thermal power and to base their transient and accident analyses on an assumed power level of at least 102% of the licensed thermal power level. The 2% power margin was intended to address uncertainties related to heat sources and measuring instruments’ accuracy. Appendix K to 10 CFR Part 50 did not allow any credit for demonstrating that the measuring instruments may be more accurate than originally assumed in the ECCS rulemaking. Thus, Appendix K did not originally require the power measurement uncertainty be determined, but instead required a fixed 2% margin.

On June 1, 2000, the U.S. Nuclear Regulatory Commission (NRC) published a final rule (65 FR 34913) that allows licensees to justify a smaller margin for power measurement uncertainty when more accurate instrumentation is used to calculate the reactor thermal power and calibrate the neutron flux instrumentation. This revision to Appendix K to 10 CFR Part 50 allows licensees to use a power uncertainty of less than 2% in the design-basis loss-of-coolant accident (LOCA) analyses, provided that feedwater flow measurement devices that provide for a more accurate calculation of power are utilized. License amendments to increase power based on improved feedwater flow measurements are commonly referred to as measurement uncertainty recapture power uprates.

1.2 Proposed IP2 Measurement Uncertainty Amendment

In support of its December 12, 2002, application, the licensee performed a revision and re-evaluation of various NSSS parameters, safety-related systems and components, nuclear fuel, and accident analyses related to operation at the increased reactor power level of 3114.4 MWt. For the NSSS evaluations, a value of 3115 MWt was used, which conservatively bounds the amendment value of 3114.4 MWt. The licensee's application is based on a reduced core-thermal-power uncertainty because of a more accurate measurement of feedwater flow. The improved accuracy is achieved by installation of the Caldon Leading Edge Flow Meter (LEFM) Check System. The improved flow measurement instrumentation would allow the licensee to operate IP2 with a margin below the 2.0% margin previously used in the licensing basis ECCS analyses. The licensee stated that, as a result of the improvement in the flow measurement accuracy, the IP2 power measurement uncertainty has been reduced from 2.0% to 0.6%.

The licensee's application referenced Caldon Engineering Reports ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Using the LEFM Check System," ER-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM Check System," and ER-157P, "Basis for a Power Uprate with the LEFM Check or LEFM CheckPlus System," to provide a generic basis for the proposed 1.4% power uprate. Engineering Reports ER-80P, ER-160P, and ER-157P were approved by the NRC staff, respectively in safety evaluation (SEs) dated March 8, 1999, January 19, 2001, and December 20, 2001.

2.0 REGULATORY EVALUATION

As set forth below, the NRC staff finds that the licensee in its December 12, 2002, application addressed the applicable regulatory requirements. 10 CFR Part 50 establishes the fundamental regulatory requirements with respect to safety-related systems. The regulatory standards and other guidance which the NRC staff considers in its review are as follows:

a. 10 CFR 50.46, "Acceptance Criteria for [ECCSs] for light-water nuclear power reactors," requires, in part, that the ECCS cooling performance be calculated in accordance with an acceptable evaluation model and for a number of postulated LOCAs. Comparisons to experimental data must be made and uncertainties in the analysis method and inputs must be identified and assessed.

b. Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50 sets forth the requirements for the models.

c. Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50 provides, in part, requirements related to the establishment of reactor pressure vessel (RPV) pressure versus temperature (P-T) limit curves for any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests. Appendix G also references the requirements prescribed in the latest addition and addenda of the American Society for Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Appendix G, incorporated by reference into 10 CFR 50.55a(b)(2).

d. Appendix H, “Reactor Vessel Material Surveillance Program Requirements,” to 10 CFR Part 50 sets forth requirements related to the establishment of a facility’s RPV surveillance capsule program and withdrawal schedule. Appendix H also incorporates by reference the requirements in American Society for Testing and Materials Standard E 185, “Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels,” as modified by that appendix.

e. ASME Code Sections III and XI, as incorporated in 10 CFR 50.55a, provide additional guidance regarding the evaluation of the structural integrity of RPV internals and ASME Code Class 1, 2, and 3 components, component supports, and core support structures.

f. 10 CFR 50.49, “Environmental qualification of electric equipment important to safety for nuclear power plants,” requires licensees to establish programs to qualify electric equipment important to safety. Under the rules, each licensee must prepare and maintain a record of qualification to document that each item of equipment subject to the rule is qualified for its application and meets its specified performance requirements when subjected to the environmental conditions predicted to be present when it must perform its safety function up to the end of qualified life.

g. 10 CFR 50.61, “Fracture toughness requirements for protection against pressurized thermal shock events,” provides RPV fracture toughness criteria relevant to events that could cause severe overcooling of the reactor coolant system with or followed by significant pressure in the RPV.

h. 10 CFR 50.63, “Loss of all alternating current power,” requires that all nuclear power plants must have the capability to withstand a loss of all alternating current power to the essential and nonessential switchgear buses for a specified duration, and to recover therefrom.

i. 10 CFR Part 100, “Reactor Site Criteria,” establishes, in part, criteria important to assuring that radiological dose from normal operation and postulated accidents will be acceptably low.

j. Several General Design Criteria (GDC) in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities."

    1. GDC-4, “Environmental and dynamics effects design bases,” requires that systems, structures, and components, be designed to accommodate the effects of and to be compatible with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents. These systems, structures, and components shall also be appropriately protected for dynamic effects such as missiles, pipe whipping and discharging fluid that result from system failures and from events and conditions outside the plant.

    2. GDC-14, “Reactor coolant pressure boundary [RCPB],” requires the RCPB to be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

3. GDC-17, “Electric power systems,” requires that an onsite electric power system and an offsite electric power system be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be, in part, to provide sufficient capacity and capability to assure that the containment integrity and other vital functions are maintained in the event of postulated accidents.

4. GDC-19, “Control room,” requires, in part, that adequate radiation protection be provided to permit access and occupancy of the control room under accident conditions without radiation exposure in excess of specified levels.

5. GDC-30, “Quality of reactor coolant pressure boundary,” requires, in part, that means be provided for detecting and, to the extent practical, identifying the location of reactor coolant leakage.

6. GDC-31, “Fracture prevention of reactor coolant pressure boundary,” requires, in part, that the RCPB be designed with sufficient margin to assure that the RCPB behaves in a non-brittle manner and the probability of rapidly propagating fracture is minimized.

k. NRC Regulatory Guides (RGs)

1. RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” March 2001.

2. RG 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” provides guidance for an applicant’s preparation of RPV P-T limit curves.

3. RG 1.161, “Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy [USE] Less Than 50 Ft-Lb,” along with Appendix K to Section XI of the ASME Code provide guidance when USE equivalent margins analyses are required by Appendix G to 10 CFR Part 50.

l. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants.”

m. NUREG-1061, Volume 3, “Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Evaluation of Potential for Pipe Breaks,” and draft Standard Review Plan Section 3.6.3, “Leak Before Break [LBB] Evaluation Procedures,” provide guidance for evaluating the technical basis for a licensee’s application of LBB.

3.0 <u>TECHNICAL EVALUATION</u>

3.1 Feedwater Flow Measurement Technique and Power Measurement Uncertainty

<u>Power Calorimetric Instrumentation</u>

Neutron flux instrumentation is calibrated to the core thermal power, which is determined by an automatic or manual calculation of the energy balance around the plant NSSS. This calculation is called the "secondary calorimetric" for a pressurized-water reactor (PWR). The accuracy of this calculation depends primarily upon the accuracy of feedwater flow and feedwater net enthalpy measurements. Thus, an accurate measurement of feedwater flow and temperature will result in an accurate calorimetric calculation and an accurate calibration of the nuclear instrumentation.

The instrumentation for measuring feedwater flow typically uses a venturi meter, an orifice plate, or a flow nozzle to generate a differential pressure proportional to the feedwater velocity in the pipe. Of these three differential pressure devices, a venturi meter is most widely used for feedwater flow measurement in nuclear power plants. The feedwater temperature is typically measured by resistance temperature detectors (RTDs). The IP2 design uses a venturi and temperature detectors for flow and temperature measurement in each of the four feedwater lines. The major advantage of using a venturi flow meter is the relatively low head loss created as the feedwater passes through the device. The major disadvantage of the venturi flow meter is the effect of venturi fouling upon its measurement accuracy. Fouling causes a venturi flow meter to indicate higher differential pressures for equivalent flow velocities, which results in an output signal representing a higher than actual flow rate. Since feedwater flow rate is directly proportional to calorimetric power, this error in feedwater flow rate measurement leads the plant operator to calibrate the nuclear instrumentation at a higher than actual core power.

Calibrating the nuclear instrumentation to indicate higher than actual core power is conservative with respect to reactor safety, but causes the licensee to generate electrical power proportionately lower when the plant is operated at its indicated thermal power rating. To eliminate this effect of venturi fouling on reactor power operating limits, the venturi flow meter device must be removed, cleaned, and calibrated. The high cost of venturi flow meter calibration and the need to improve flow instrumentation accuracy prompted the nuclear industry to assess other flow measurement techniques. The industry found the LEFM, which implements a transit time methodology, to be a viable alternative.

The Caldon Chordal LEFM is an ultrasonic flow meter, using acoustic energy pulses to determine the feedwater mass flow rate and temperature. The meter is based on time-of-flight (transit time or counter-propagation) technology. The transit time technology sends an ultrasonic signal diagonally through the fluid and then measures the time it takes to travel upstream and downstream. The sound travels faster when the pulse traverses the pipe with the flow and slower when the pulse traverses the pipe against the flow. The difference in these times is proportional to the velocity of the fluid in the pipe. The LEFM uses these transit times to determine the fluid velocity and temperature (the temperature of the fluid is determined from a predetermined correlation between the fluid pressure and sound velocity in the fluid). The LEFM Check System is a digital system controlled by software using the ultrasonic transit time method to measure four velocities at precise locations with respect to the pipe centerline. The system numerically integrates the four measured velocities to determine the mass flow rate and the fluid temperature. These measurements are used by the plant computer to determine the reactor thermal power.

Caldon Engineering Report ER-80P, and its supplement ER-160P, describe the LEFM Check System and provide calculated uncertainties in percent power for a typical PWR and boiling-water reactor (BWR) using measurements by a single meter LEFM Check System. ER-80P and ER-160P provide a generic basis for an uprate up to 1.4% of the licensed reactor power with the use of the LEFM Check System.

At IP2, the LEFM Check System consists of an electronics cabinet and one LEFM Check measurement section (spool-piece), permanently installed in each of the four main feedwater lines.

The licensee stated that IP2 LEFM Check System is designed and manufactured in accordance with Caldon's 10 CFR Part 50, Appendix B, Quality Assurance Program, and the system software had been developed and will be maintained under a verification and validation (V&V) program. The licensee listed the applicable Institute of Electrical and Electronics Engineers (IEEE) and ASME standards and stated that Caldon's V&V program conforms to the provisions of those standards. The LEFM Check System will be included in the plant preventive maintenance program and site procedures for calibration and maintenance are being developed in accordance with the guidelines provided in Caldon technical manuals and Caldon Topical Report ER-80P. The licensee's submittal listed all applicable procedures and stated that all instruments that provide fluid condition data for calculation of reactor thermal power will be controlled and calibrated in accordance with plant procedures. These instruments are performance monitored to the conditions represented in the overall calorimetric uncertainty evaluation performed for the proposed 1.4% power uprate of IP2. All conditions that are adverse to quality are documented under the corrective action program and procedures are maintained for notification of deficiencies and error reporting.

The LEFM Check System indications of feedwater flow and temperature are displayed on a local display panel and transmitted to the plant computer for real-time calculation of reactor thermal power. When the plant computer executes the plant calorimetric software, it determines if the LEFM inputs are reliable. Only reliable data are used to calculate reactor thermal power and LEFM/Venturi feedwater flow correction factors, which are saved to be used in the event that unreliable LEFM inputs are encountered for the next periodic power calorimetric calculation. If the LEFM inputs are found to be unreliable, then the power calorimetric calculation is conducted with the corrected Venturi feedwater flow values. Since IP2 has installed one LEFM flow element in each of the four feedwater lines, failure of any one of the LEFMs will result in calculation of thermal power based on the data from the operable LEFMs and the corrected Venturi feedwater flow measurements. The LEFM system has a self-diagnostic feature that detects possible failures and changes in hydraulic profiles that affect the accuracy of the LEFM system. Alarm thresholds are provided on the plant computer display in the control room to alert operators to conditions that impair LEFM system availability or accuracy. The licensee stated that the LEFM Check System does not perform any safety function, and is not used to directly control any plant systems. Therefore, its inoperability has no immediate effect on plant operation.

The NRC staff SE on Caldon Engineering Report ER-80P identified four additional matters to be addressed by a licensee referencing ER-80P in their request for a power uprate. The licensee's submittals addressed each of the four matters as follows:

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

    - The plant procedures for maintenance and calibration of the LEFM Check System were developed for IP2 based on the vendor's recommendations. The licensee stated that if the LEFM Check System becomes unavailable, plant operations at a core thermal power level of 3114.4 MWt (uprate power level) may continue for an allowed outage time (AOT) of 7 days, as long as steady-state conditions persist (i.e., no power changes in excess of 10% during the AOT). If the 7-day outage period is exceeded, or if the plant experiences a power decrease of greater than 10% during the 7-day period, the permitted maximum power level would be reduced, consistent with the accuracy of the venturi and feedwater temperature detector, upon return to full power. The licensee stated that it is considered likely that any degradation as a result of venturi fouling, drift, and the like, would be imperceptible for the 7-day period as long as steady-state conditions persist. The licensee reviewed data collected over a several month period of feedwater flow measured by an LEFM against feedwater flow measured by a the venturi flowmeter. Based on this review, the licensee stated that the average ratio, when examined over any particular 7-day interval, typically varies by approximately 0.2% or less, and displays no discernable pattern or drift, but appears to vary randomly. As such, a calorimetric based on the venturi measurement of feedwater flow would be sufficiently accurate.

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 Engineering Report ER-80P.

   - The licensee stated that the LEFM flow elements were installed at IP2 in 1980. A complete refurbishment was accomplished in 1995 which included the replacement of transducers, reconditioning of transducer housing, and upgrade of original electronics to the current generation Caldon LEFM 8300 electronics unit in 1995. In the year 2002, the installed LEFM electronics was again upgraded to the LEFM Check System electronics and the system was recommissioned to verify that all provisions of Caldon Topical Report ER-80P were met. Uncertainties of all dimensions were bounded in the uncertainty analysis per the practices established and approved in Topical Report ER-80P. The system has been used and maintained throughout its installed history and the transducers were replaced as a normal maintenance item. The licensee confirmed that the Caldon LEFM Check System installed at IP2 is representative of the Caldon LEFM Check System discussed in Topical Report ER-80P, and is bounded by the provisions set forth in this topical report.

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

   - The licensee stated that the methodology used to calculate the Caldon LEFM Check System uncertainties is consistent with ASME Pressure Test Code (PTC) 19.1 and Instrument Society of America (ISA) 67.04 as approved in Topical Report ER-80P. The licensee's submittal included Caldon Engineering Report ER-290 Revision 2, "Bounding Uncertainty Analysis For Thermal Power Determination at Indian Point Unit 2 Nuclear Power Station Using The LEFM Check System (proprietary)" and Westinghouse analysis in WCAP-15904 Revision 0," Power Calorimetric Uncertainty For The 1.4% Uprating For Entergy Indian Point Unit 2 (proprietary)." Both documents include calculation of measurement uncertainties by combining the random and bias terms of their respective measurement uncertainty components using square-root-of-the-sum-of-the squares (SRSS) methodology, provided all the terms are independent, zero-centered, and normally distributed. (Those uncertainties that are dependent are arithmetically combined into the independent groups.) The SRSS methodology is an acceptable methodology for combining instrumentation uncertainties per Instrument Society of America (ISA) standard ISA-S67.04, Part 1 -1994, "Setpoints for Nuclear Safety-Related Instrumentation." ISA-S-67.04, Part 1 - 1994 is endorsed by NRC RG 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation." The Caldon engineering report calculations determined that IP2 LEFM Check System mass flow measurement uncertainty, which is proprietary, is better than 0.5% of rated flow with a 95% confidence interval. This calculated value of the LEFM Check System mass flow measurement uncertainty is used in the Westinghouse analysis to determine the power calorimetric measurement uncertainty. Westinghouse analysis of the IP2 power measurement uncertainty using the LEFM Check System established an uncertainty value that bounds the licensee's assumed 0.6% power calorimetric measurement uncertainty for the proposed 1.4% power uprate.

4. Licensees of plants 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. The 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 LEFM, the licensee should confirm that the piping configuration remains bounding for the original LEFM installation and calibration assumptions.

   - IP2 flow elements were not calibrated in a site-specific hydraulic geometry. However, the testing performed in the 1970s and 1980s under a Westinghouse program demonstrated that the profile factor for the four-path LEFM system is not very sensitive to varying pipe geometry. The licensee stated that two of the four LEFM flow elements (loops 21 and 22 elements) were tested at Alden Research Laboratory (ARL) in a straight pipe configuration. Calibration correction factors (profile factor and uncertainty) have been applied to the straight pipe test results to reflect the as installed pipe geometry. The licensee submittal included ARL Report No.106-79/C97 which documents the straight pipe testing and test results, and MPR Associates Inc. Report MPR-1614 (proprietary) dated October 31, 1995, which describes the correction factor methodology. The licensee stated that the calibration correction factors for the remaining two LEFM flow elements (loops 23 and 24 elements) are based on straight pipe testing of a population of seven flow elements at ARL, which is the same test population used to establish the correction factors for the flow elements installed at Indian Point Unit 3 (IP3). Also, the correction factor methodology and results for the flow elements installed in loops 23 and 24 are included in MPR-1614. The NRC staff's SE on the 1.4% power uprate of IP3 dated November 26, 2002, (ADAMS Accession No. ML02390636) found the LEFM Check flow element correction factors acceptable, and for the reasons set forth in that SE, the NRC staff here finds that the above stated

ARL test report and MPR configuration and uncertainty analysis have established acceptable correction factors for the LEFM Check System flow elements at IP2.

Based on the foregoing, the NRC staff finds that the licensee's responses have resolved the plant-specific concerns regarding LEFM Check System maintenance and calibration, hydraulic configuration, processes and contingencies for an inoperable LEFM Check System, and the methodology for the plant-specific calculations of the IP2 power measurement uncertainty.

Based on the staff's review of the licensee's submittals on the LEFM Check System and plant power calorimetric uncertainty, and for the reasons set forth above, the NRC staff finds that the IP2 thermal power measurement uncertainty with the LEFM Check System is limited to 0.6% of rated thermal power and can support the proposed 1.4% power uprate of IP2. The NRC staff also finds that the licensee addressed the four additional matters outlined in the NRC staff SE of Caldon Engineering Report ER-80P. The NRC staff, therefore, finds the licensee's request for a 1.4% thermal power uprate to be acceptable.

3.2 Evaluation of Accident and Transient Analyses

3.2.1 NSSS Design Parameters

In its 1.4% measurement uncertainty recapture power uprate application report (Attachment III in licensee's December 12, 2002, letter), the licensee provided a list of revised thermal design parameters to reflect the 1.4% increase in the IP2 licensed core power from 3071.4 MWt to 3114.4 MWt. The parameters include: reactor power; thermal design flow rate; reactor coolant pressure, temperature, and flow rate; feedwater temperature; and steam pressure, temperature and flow rate. These key plant parameters have been reconciled in the licensee's evaluations and safety analyses to support the 1.4% power uprate. Since values of these parameters are demonstrated acceptable by various analyses described in Chapter 14 of the Updated Final Safety Analysis Report (UFSAR) that use these parameters as inputs, the NRC staff find that these power uprate parameters are acceptable to support the licensee proposed power uprate.

3.2.2 NSSS Design Transients

In its power uprate application report, the licensee has evaluated the NSSS design transients to account for any impacts of the power uprate. The NSSS design transients are traditionally developed for fatigue analyses of the various NSSS components using conservative assumptions. The licensee provided a tabulation comparing the plant operating conditions at the original power rating and the 1.4% power uprate. In its power uprate application report, the licensee indicated that the current NSSS design transients were developed using conservative values associated with either frequency of occurrence or the transient assumptions. Also, the design transients are analyzed assuming a 2% power uncertainty allowance, which bounds the 1.4% power uprate plus the 0.6% power measurement uncertainty. Since the limiting values of primary and secondary system temperatures are not changed, the existing design transients remain valid for the proposed power uprate. The NRC staff has reviewed the licensee's submittal and concurs, for the reasons set forth by the licensee, with the licensee's conclusion.

3.2.3 Accident and Transient Analyses

In support of this power uprate, the licensee reevaluated the transient and accident analyses for IP2 for operation at a rated core thermal power of 3114.4 MWt with the power measurement uncertainty of 0.6%. The changes of NSSS parameters discussed in Section 3.2.1 of this report are used in the analyses to support the power uprate. The licensee performed the uprate analyses and evaluations in accordance with the current IP2 licensing bases methodologies. Since the licensee used NRC-approved methodologies and appropriate plant-specific input assumptions, the NRC staff finds the transient and accident analyses that were reevaluated acceptable. This holds true for all analyses discussed below in this SE, except where this SE describes changes in the values of input parameters.

The licensee evaluated the UFSAR Chapter 14 transient and accident analyses to identify whether any are affected by the 1.4% power uprate. There are five categories of events tabulated in Table 8-1 of its application report. They are: (1) LOCA-related events; (2) Affected Non-LOCA events reanalyzed for 1.4% power uprate; (3) Affected Non-LOCA events evaluated for the 1.4% power uprate using existing departure from nucleate boiling (DNB) margin; (4) Non-LOCA events bounded by the current 102% power assumption; (5) Non-limiting/ Bounding Events; and (6) Events precluded by the IP2 TSs. The NRC staff's evaluation of these event categories is addressed below.

3.2.3.1 LOCA-Related Events

<u>Large-Break LOCA (LBLOCA)</u>

As part of the IP2 power uprate proposal, the licensee identified that a variant version of the Westinghouse best estimate LBLOCA analysis methodology that was approved by the NRC staff for application to IP2 in IP2 License Amendment No. 188 (March 31, 1997) would continue to be the licensing-basis LBLOCA analysis methodology for IP2 at the uprated power. The licensee also indicated that the initial LBLOCA analyses it had performed using the methodology in support of IP2 License Amendment No. 188 would continue to be the LBLOCA analysis of record for IP2 because it had performed the analyses at the uprated power.

The NRC staff's SE associated with IP2 License Amendment No. 188 placed three conditions on the IP2 version of the LOCA methodology:

a. This version of the EM may be referenced only for the initial IP2 analyses for as long as they remain applicable per 10 CFR 50.46 requirements or until they are superseded by updated analyses. Future analyses using the EM must be performed entirely using the <u>W</u> BE LBLOCA EM MOD 7A Rev. 1 version or other fully approved LBLOCA EM.

b. The imprecision of the correction [to the effect on peak clad temperature of the difference between the generically approved EM and the IP2-specific EM] must be tracked in IP2 10 CFR 50.46 reports as a permanent change or error.

c. Reference to the [Consolidated Edison of New York's] June 13, 1996, letter [on justification for one-time use of this EM] must be maintained in appropriate licensing documentation (TSs and/or core operating limits report (COLR).

In response to an NRC staff request to show that the proposed use of the present IP2 LBLOCA best estimate licensing basis LBLOCA methodology and present IP2 LBLOCA analyses continue to comply with Condition a., the licensee provided a statement that “Entergy Nuclear Operations, Inc. and Westinghouse have ongoing processes which assure that the ranges and values of LOCA analyses inputs for Peak Cladding Temperature (PCT) sensitive parameters bound the as-operated plant ranges and values for those parameters.”

The licensee performed the present IP2 LBLOCA analyses at a power that equals or bounds the proposed IP2 power. Accordingly, the NRC staff concludes that these confirm that the IP2 LBLOCA methodology and IP2 LBLOCA analyses for the uprated power continue to be applicable, and therefore meet Condition a.

The NRC staff consulted the licensee’s 10 CFR 50.46 reports since March 31,1997, and confirmed that the licensee has tracked the imprecision of the correction, a permanent change or error, thereby satisfying Condition b.

The NRC staff confirmed that the licensee referenced the June 13, 1996, letter in appropriate licensing documentation (TSs and/or COLR), thereby satisfying Condition c.

The NRC staff concludes that IP2 LBLOCA analysis methodology and the IP2 LBLOCA analyses using that methodology continue to be acceptable at the uprated power because the analysis methodology continues to satisfy the three conditions in the March 31, 1997, NRC staff SE, and because the licensee performed the analyses at a power that equals or bounds the proposed IP2 power.

Small-Break LOCA (SBLOCA)

The current licensing basis SBLOCA analyses using the 10 CFR Part 50, Appendix K methodology employs a nominal core power of 3071.4 MWt. The current licensing methodology applies a 2% calorimetric power measurement uncertainty allowance resulting in an assumed core power of 3132.8 MWt in accordance with the original requirements 10 CFR Part 50, Appendix K. This analytical power level of 3132.8 MWt is equivalent to the uprated power level of 3114.4 MWt with a 0.6% calorimetric power measurement uncertainty allowance. Therefore, the existing SBLOCA analyses are still applicable to the proposed power uprate conditions at IP2.

Post-LOCA Long-Term Core Cooling

10 CFR 50.46(b)(5) requires that: “After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.” Since credit for the control rods is not taken for a LBLOCA, the borated ECCS water provided by the refueling water storage tank and accumulators must have a boron concentration that, when mixed with other sources of water, will result in the reactor core remaining subcritical, assuming all control rods out. The water volumes and associated boron concentration of all the water sources involved following a LOCA are not affected by the 1.4% power uprate.   Therefore, the current long-term core cooling analysis of record is unaffected by the 1.4% power uprate.

3.2.3.2 Affected Non-LOCA Events Reanalyzed for 1.4% Power Uprate

Some non-LOCA events are affected by the 1.4% power uprate because the current analyses do not already explicitly account for a 2% power measurement uncertainty allowance. The following three events had to be analyzed to address the potential effects of the 1.4% power uprate.

Uncontrolled Control Rod Assembly Withdrawal at Power

To analyze this event, the licensee used the methodology documented in the analysis of record. The methodology utilizes the LOFTRAN computer code for the transient analysis simulation. The Westinghouse reload SE methodology described in WCAP-9272-P-A is also applied. As set forth in Standard Review Plan (SRP) Section 15.4.2, the staff considered minimum DNBR, peak primary pressure, peak secondary pressure, and peak core average heat flux. For the 1.4% power uprate with a 0.6% power measurement uncertainty allowance, the calculated minimum departure from nucleate boiling ratio (DNBR) is 1.7222, which is higher than the minimum DNBR safety analysis limit of 1.58 for the 1.4% power uprate program, corresponding to the WRB-1 DNBR correlation. The calculated peak primary pressure and peak secondary pressure are 2728.5 psia and 1168.1 psia which are less than 110% of their design pressures (2748.5 psia for the primary system and 1208.5 psia for the secondary system). The calculated peak core average heat flux is 116.9%, which is less than its safety limit of 118%, and this precludes fuel centerline melt. The results of the reanalysis meet the acceptance criteria of SRP 15.4.2, and therefore, are acceptable.

Loss of External Electric Load

To analyze this event, the licensee used the methodology documented in the analysis of record. The methodology utilizes the LOFTRAN computer code for the transient analysis simulation. The Westinghouse reload SE methodology described in WCAP-9272-P-A is also applied. As set forth in SRP 15.2.1, the staff considered minimum DNBR, peak primary pressure, and peak secondary pressure. For the 1.4% power uprate with 0.6% power measurement uncertainty allowance, the calculated minimum DNBR is 2.04 which is higher than the minimum DNBR safety analysis limit of 1.58. With respect to pressure effects of this event, the current licensing basis analysis in which pressurizer pressure control is assumed to be unavailable is not affected by an increase in the nominal full power. This is because the power level assumed in the current analysis for this case (with 2% uncertainty) is equivalent to that based upon the uprated power of 3114.4 MWt, combined with the lower uncertainty of 0.6 % and remains bounding for the 1.4% power uprate. The results of the licensee's evaluation and reanalysis meet the acceptance criteria of this event stated in SRP 15.2.1, and therefore, are acceptable.

Excessive Heat Removal Due to Feedwater System Malfunctions (Full-Power Analysis)

To analyze this event, the licensee used the methodology documented in the analysis of record. The methodology utilizes the LOFTRAN computer code for the transient analysis simulation. The Westinghouse reload SE methodology described in WCAP-9272-P-A is also applied. As set forth in SRP Section 15.1.1, the staff considered the minimum DNBR. For the 1.4% power uprate with 0.6% power measurement uncertainty allowance, the calculated minimum DNBR is 2.306 for both of the automatic rod control and manual rod control cases, which is higher than the minimum DNBR safety analysis limit of 1.58. Since this is a cooldown event, peak system pressure is not a concern for this event. The results of the reanalysis meet the acceptance criteria of this event stated in SRP 15.1.1, and therefore, are acceptable.

3.2.3.3 Affected Non-LOCA Events Evaluated Using Existing DNB Margin

The following non-LOCA events are affected by the 1.4% power uprate, but it was possible to sufficiently address the potential effects through technical evaluation and the use of available DNB margin rather than performing a full analysis.

Rod Assembly Misalignment and Rod Cluster Control Assembly (RCCA) Drop

The dropped RCCA transients were previously analyzed using the methodology described in WCAP-11394, and examined to determine that the DNB design basis is met. As set forth in SRP Section 15.4.3, the staff considered the minimum DNBR and fuel temperature limits. This methodology involves the use of generic statepoints for the dropped rod event. Sensitivity studies on the effect of a power increase on the generic statepoints were previously performed for a four-loop plant. The studies quantified the effect of an approximately 5% power increase on the four-loop generic statepoints. Since the IP2 1.4% power uprate is much smaller than the power increase used in the sensitivity studies, the generic statepoints continue to apply to IP2. The licensee has evaluated the DNB design basis using the generic statepoints and increased nominal heat core flux associated with 1.4% power uprate and confirmed that the DNB design basis continues to be met. Therefore, all applicable acceptance criteria for this event stated in SRP 15.4.3 continue to be met for the 1.4% power uprate.

Loss of Reactor Coolant Flow and Reactor Coolant Pump Shaft Seizure or Shaft Break

Since the 1.4% power uprate could potentially affect the minimum DNBR, an evaluation was completed for these events. As set forth in SRP Sections 15.3.1 through 15.3.4, the staff considered the minimum DNBR and fuel temperature limits. The licensee's evaluation concludes that considering the statepoints affected by the increased nominal heat flux due to the 1.4% power uprate, the DNB design limit remains satisfied for these events. With respect to peak system pressures, the current analysis for these event models assumes a 2% power measurement uncertainty which bounds the 0.6% uncertainty and 1.4% power uprate. Therefore, all applicable acceptance criteria for this event stated in SRP 15.3.1 through 15.3.4 continue to be met for the 1.4% power uprate.

3.2.3.4 Non-LOCA Events Bounded by Current 102% Power Assumption

The following non-LOCA events are currently analyzed with an explicit 2% power measurement uncertainty allowance that already bounds operation at the 1.4% uprate power level with the reduced power measurement uncertainty allowance of 0.6%: (1) Locked Rotor Accident (overpressure analysis); (2) Loss of External Electrical Load (overpressure analysis); (3) Loss of Normal Feedwater; (4) Loss of All AC Power to the Station Auxiliaries; (5) Rupture of a Control Rod Mechanism Housing (RCCA Ejection); and (6) Steam Generator Tube Rupture event. The licensee has evaluated the effects of the small changes in the plant initial operating conditions to these analyses and concluded that the current analyses of record for these events remain valid for the 1.4% power uprate conditions. The NRC staff finds the licensee's assessment acceptable since the 2% power measurement uncertainty bounds the 0.6% uncertainty and 1.4% power uprate.

3.2.3.5 Non-Limiting/Bounding Events

The following non-LOCA events are either bounded by the current respective analyses of record, or simply are not affected because they are performed starting at hot zero power or a power less than full power: 1) Uncontrolled control rod cluster assembly withdrawal from a subcritical or low startup power conditions, 2) Chemical and volume control system malfunction, 3) Excessive heat removal due to feedwater system malfunctions (zero power analysis), 4) Excessive load increase incident, 5) Rupture of a steam pipe (zero power analysis), and 6) Rupture of a control rod mechanism housing (zero power analysis). Since these events are not effected by the rated full power level, the NRC staff considers that the analyses of record remain valid for the proposed 1.4% power uprate.

Anticipated Transients Without Scram (ATWS)

IP2 has implemented the ATWS rule, 10 CFR 50.62, by installing a diverse turbine trip (DTT) and diverse emergency feedwater actuation system (DEFAS). These system designs were approved by the NRC based on their reliability, independence, and diversity from the plant protection system.

The licensee indicates that the Westinghouse generic ATWS analysis for four-loop plants at a rated power of 3025 MWt is applicable to IP2 with the current plant configuration except the total auxiliary feedwater pump capacity is 10% less than that assumed in the generic analysis. The peak primary pressure for the limiting loss of load ATWS event in the generic analysis is 2979 psia. The results of the sensitivity studies associated with the generic analysis show that an increase in power of 2% will increase peak pressure by 44 psi. Thus, for a plant with a power level 3.4% higher than the generic analysis, peak pressure will increase by 75 psi. Also, the sensitivity studies show that a reduction of emergency feedwater flow of 10% will increase peak pressure by 12 psi. Based on the data from the above sensitivity studies, the resulting peak pressure for a limiting loss of load ATWS will remain below the maximum allowable limit of 3200 psia with sufficient margin.

The NRC staff reviewed the licensee's submittal and, for the reasons set forth above, finds that the design at IP2 regarding ATWS remains effective for the proposed 1.4% power uprate conditions and therefore, is acceptable.

3.2.3.6 Event Prohibited by IP2 TSs

The current IP2 TSs do not allow less than four loop operation during Modes 1 and 2. Therefore, an event that involves startup of an inactive reactor coolant loop is prevented and analysis of this event is not necessary. The analysis of this event documented in UFSAR does not apply to the operation at IP2 with or without power uprate.

Station Blackout (SBO)

In response to the NRC staff's request for additional information regarding the impacts of the 1.4% power uprate on the ability of IP2 to cope with an SBO event, the licensee, in its April 3 letter, stated the following:

(1) At the current power level of 3071.4 MWt, 142,850 gallons of condensate inventory is required for 8 hours of decay heat removal and primary system cooldown during an SBO. For the 1.4% power uprate (3114.4 MWt), a small increase (less than 1%) in this water volume is needed. Since the current TS requires that a minimum of 360,000 gallons of condensate must be available in the condensate storage tank during plant operation above 350 ^{o}F, a large margin exists for the required water inventory during an SBO;

(2) There is no increase in load on the station batteries for coping with an SBO under the 1.4% power uprated conditions; and

(3) The air operated valves used for mitigating an SBO have sufficient backup source of air supply and they can be operated manually.

Since systems associated with SBO are not affected by the 1.4% power uprate, the turbine-driven auxiliary feedwater (AFW) pump will be available for decay heat removal following an SBO. Since a motor-driven AFW pump with a design capacity of 400 gpm is sufficient for decay heat removal during a loss of normal feedwater transient, the turbine-driven AFW pump with 800 gpm capacity is more than sufficient to remove decay heat following an SBO at 1.4% power uprated conditions at IP2. In view of the foregoing, the NRC staff has concluded that the SBO at IP2 is not affected by the 1.4% power uprate.

3.2.3.7 Evaluation of Radiological Consequences

The NRC staff reviewed the impact of the proposed changes on design basis accident radiological analyses, as documented in Chapter 14 of the IP2 UFSAR. In its submittal, the licensee identified the existing DBA radiological analyses of record by their location in the IP2 UFSAR. The licensee stated that the current radiological analyses of record for IP2 were unaffected by the requested power uprate, because they were performed assuming a nominal core power of 3216 MWt. Analyses performed at this power bound analyses performed assuming the requested uprated power of 3114.4 MWt with a 0.6% power measurement uncertainty. Using the current IP2 UFSAR documentation in addition to information in the December 12, 2002, submittal letter, the NRC staff verified that the existing IP2 UFSAR Chapter 14 radiological analyses source term and steam release assumptions, as appropriate, bound the conditions for the proposed 1.4% power uprate to 3114.4 MWt.

Based on the above discussion, the NRC staff finds that the existing IP2 UFSAR Chapter 14 radiological analyses, which were analyzed assuming a core thermal power of 3216 MWt, remain bounding for the proposed 1.4% power uprate to 3114.4 MWt, considering the higher accuracy of the Caldon LEFM system. These analyses of record show that the radiological consequences of postulated design-basis accidents meet the dose limits given in 10 CFR 50.67 and 10 CFR Part 50, Appendix A, GDC-19, as well as applicable dose acceptance criteria given in NUREG-0800, Standard Review Plan Chapter 15. Therefore, the NRC staff finds that the radiological consequences of design-basis accidents are acceptable for the proposed changes.

3.2.3.8 Summary

The NRC staff has reviewed the licensee’s analyses and proposed TS changes to support operation of IP2 at an uprated power level of 3114.4 MWt. Based on this review, the NRC staff finds that the supporting safety analyses are performed with the previously NRC-approved methods; the input parameters of the analysis adequately represent the plant conditions at the uprated power level; and the analytical results are within the applicable acceptance criteria. Therefore, the NRC staff concludes that the supporting analyses are acceptable.

3.3 Mechanical, Structural, and Material Component Integrity and Design

3.3.1 Nuclear Fuel

The licensee has evaluated the potential effects of the 1.4% power uprate on the nuclear fuel at IP2. The design margin for fuel internal pressure and cladding stress were re-evaluated based on the 1.4% power uprate conditions. The results indicate that these fuel rod design parameters continue to meet the acceptance criteria at the 1.4% power uprate conditions. Fuel evaluations are performed for each specific cycle according to the Westinghouse Reload Methodology in WCAP-9272-P-A to ensure that all fuel rod design criteria are satisfied for the specific operating conditions associated with that cycle. These analyses will be repeated prior to the implementation of the 1.4% power uprate of IP2 during Cycle 16, as well as prior to all subsequent cycles. Therefore, the NRC staff finds the fuel design acceptable to support the proposed power uprate.

3.3.2 Core Thermal-Hydraulic Design

The licensee has performed a thermal-hydraulic evaluation at the 1.4% power uprate conditions. The evaluation was based on the 15x15 VANTAGE+ fuel design with the intermediate flow mixer (IFM) grids that bound the 15X15 VANTAGE 5 fuel design for future reloads. The current design methodology for the IP2 reload SE remains unchanged for the 1.4% power uprate evaluation. The WRB-1 DNB correlation and the Revised Thermal Design Procedure (RTDP) DNB methodology are continuously used for DNB analysis. The W-3 DNB correlation is used for events where the conditions fall outside the applicable range of the WRB-1 correlation. The licensee has revised the DNBR safety limit to account for increase in the nominal power level at 1.4% power uprate of 3114.4 MWt. Based on the above stated evaluation results, the NRC staff finds the core thermal-hydraulic design acceptable to support the proposed power uprate.

3.3.3 Reactor Pressure Vessel

The licensee evaluated the reactor vessel for the effects of the revised design conditions in Table 2-1 of the application report on the most limiting vessel locations with regard to ranges of stress intensities and cumulative fatigue usage factors (CUFs) in each of the components, as identified in the original reactor vessel stress reports. The evaluations considered the operating parameters, which were identified for the uprated power condition. The existing NSSS design transients were not affected by the 1.4 percent uprating. The components of the reactor vessel affected by the power uprate include outlet nozzles, the RPV (main closure head flange, studs, and vessel flange), and CRDM housings. The licensee evaluated the maximum stresses and CUFs for the critical components at the core power uprated conditions. The evaluation was performed in accordance with the ASME Code, Section III, 1965 Edition with addenda through Winter 1965, which is the Code of record. There are no changes for the faulted condition loads as a result of the 1.4 percent power uprate. The results for the faulted condition previously evaluated are bounding and remain applicable.

The calculated maximum ranges of stress intensities and the maximum CUFs for the reactor vessel critical locations are provided in the application report. The results indicate that the maximum ranges of stress intensities are within the allowable Code limit of $3S_m$ [3 times the design stress intensity value] except the CRDM housing for which the simplified elastic-plastic analysis was performed to justify exceeding the $3S_m$ limit, as permitted by the ASME Code. The CUFs remain below the allowable ASME Code limit of 1.0. The licensee concluded that the current design of the reactor vessel continues to be in compliance with licensing basis codes and standards for the power uprate condition. Based on its review, the NRC staff agrees with the licensee's conclusion since the stress intensities and CUFs remain as allowed by the Code.

3.3.4 RPV Fluence

Fluence Calculational Methodology

IP2 has an approved pressure temperature limits report (PTLR) (Reference 4). In the process of that approval, the licensee submitted WCAP-15629, Revision 1 (Reference 5), regarding fluence methodology, which the NRC staff reviewed and approved.

In WCAP-15629, Revision 1 vessel fluence projections were calculated for 3216 MWt which bound the proposed 1.4% power uprate to 3114.4 MWt. The NRC evaluated WCAP-15629 in support of Amendment No. 224 dated February 15, 2002, (ADAMS Accession No. ML020420477) approving updated P-T limit curves for IP2. The NRC staff's evaluation of WCAP-15629, Revision 1, demonstrates that the cross section data, the cross section approximations, the geometrical approximations, and the analytical formulations conform to the guidance of RG 1.190. Therefore, the calculated fluences used in the 1.4% power uprate conform to the guidance in RG 1.190 and are acceptable.

Pressure-Temperature (P-T) Curves

Regarding P-T limit curves, the licensee concluded in Section 7.2.2 that the P-T limit curves were assessed to confirm they are based on vessel fluence projections that bound the 1.4% Measurement Uncertainty Recapture Power Uprate Programs (References 7-3 and 7-4).

The IP2 TSs contain 25 EFPY P-T limit curves (documented in Reference 7-3 and approved by NRC in Reference 7-4). The P-T limit curves developed in Reference 7-3 were based on fluences that bound the 1.4% power uprate. Therefore, the NRC staff accepts the licensee's argument that the existing heatup and cooldown curves for 25 EFPY are acceptable for the power uprate without any necessary change or reduction in EFPY.

Pressurized Thermal Shock (PTS) Reference Temperature (RT_{PTS})

As discussed in WCAP-15629, the licensee used acceptable methods for the calculation of the projected fluence value to the end-of-life (EOL). The 1.4% power uprate RT_{PTS} values for all beltline materials must not exceed the screening criteria as specified in 10 CFR 50.61. Specifically, the RT_{PTS} values of the base metal (plates or forgings) shall not exceed 270 °F, while the girth weld metal RT_{PTS} values shall not exceed 300 °F through the EOL. The critical material in the beltline region is the intermediate to lower shell girth weld. A conservative power level was assumed in the calculation of the end of license fluence value. The RT_{PTS} was estimated to be 246 °F which shows a considerable margin to the 10 CFR 50.61 screening value of 270 °F.

The NRC staff has determined that the RT_{PTS} value was estimated using an acceptable methodology, is conservative, and therefore, concludes it is acceptable.

Upper Shelf Energy (USE)

With USE analyses for the IP2 RPV, the licensee stated that the EOL USE values for all reactor beltline materials meet the requirements of 10 CFR Part 50, Appendix G, in that all beltline materials are expected to have a USE greater than 50 ft-lbs. through EOL. The licensee also stated that the bounding EOL (32 EFPY) USE was predicted using the EOL 1/4T fluence projection and the predictions are shown in Table 7-3 of the application report.

The NRC staff has evaluated the information provided by the licensee as well as information contained in the NRC staff's Reactor Vessel Integrity Database. Based on the revised fluence values noted in the table above, the NRC staff independently confirmed that the IP2 RPV materials would continue to meet the USE requirements of Appendix G to 10 CFR Part 50 though EOL.

Regarding the IP2 RPV surveillance program and capsule withdrawal schedule, the licensee concluded in Section 7.2. and also in 7.2.1:

> Assessment of the reactor vessel surveillance capsule removal schedule in the current Technical Specifications to confirm that the uprated fluence projections do not change the required number of capsules to be withdrawn from IP2 reactor. A calculation of ΔRT_{NDT} at 32 EFPY was performed to determine the number of capsules to be withdrawn for IP2. This calculation determined that the maximum ΔRT_{NDT} using the uprated fluences corresponding to 3216 MWt for IP2 at 32 EFPY is greater than 200^0 F. These ΔRT_{NDT} values require 5 capsules to be withdrawn from Unit 2 (Reference 7-6). This is consistent with the current withdrawal schedule contained in the IP2 Technical Specifications.

So far, four capsules have been withdrawn (Capsule 1:1976, Capsule 2: 1978, Capsule 3: 1982, Capsule 4:1987). IP2 has a total of eight capsules, four remaining in the reactor vessel.

Based on the revised RPV fluence information submitted by the licensee and the NRC staff's review of the current IP2 surveillance capsule withdrawal schedule, the staff finds the reactor vessel material surveillance program requirements specified in Appendix H to 10 CFR Part 50 will continue to be met.

3.3.5 Reactor Core Support Structures and Vessel Internals

The changes in the reactor coolant system (RCS) temperature produce changes in the boundary conditions experienced by the reactor internal components. Also, increases in core power will increase nuclear heating rates in the lower core plate, upper core plate, and baffle-barrel former region (including core barrel, baffle plate, baffle/barrel region bolts). The licensee evaluated the reactor internals to ensure their structural integrity at the 1.4% power uprate conditions. The reactor internal components for IP2 were not licensed to the ASME Code. However, the design of the IP2 reactor internals was evaluated in accordance with the requirements of Subsection NG of the 1986 Edition of the ASME Code, Section III.

The licensee evaluated critical reactor internal components considering the revised design conditions provided in Table 2-1 of the application report for plant operation at the proposed power uprate level. The licensee verified that for the baffle-barrel region components, the current structural and thermal analysis results are still bounding for the revised design conditions associated with the 1.4% power uprate. The licensee also indicated that for all of the reactor internal components, except the lower core plate and the upper core plate, the stresses and CUFs are unaffected by the proposed 1.4% power uprate, because the existing design basis analyses remain bounding. The licensee performed structural evaluations to demonstrate the design adequacy of the lower and upper core plate for the revised design conditions. Table 7-4 of the application report provides the maximum calculated stress intensities and CUFs for the lower and upper core plates. The calculated stresses are shown to be less than the Code allowable stress limits and the CUFs are less than the limit of 1.0. In addition, the licensee evaluated the hydraulic uplift force and flow induced vibration due to the vessel/core inlet coolant temperature decrease of 1 °F for the power uprate. It is noted that the decrease of 1 °F in coolant temperature does not result in a significant increase in fluid density. Therefore, the licensee concluded that there is no adverse impact on the design basis of the reactor internals with regard to flow induced vibration and that the net reactor internals hold-down forces are not affected by the increased flow as a result of the 1.4% power uprate.

In addition, the licensee stated in Section 7.3.3 of the Application Report:

> Evaluations were performed to demonstrate that structural integrity of the reactor components is not adversely affected by the 1.4-percent power uprate conditions. The presence of heat generated in reactor internal components, along with the various fluid temperatures, results in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth, which must be accounted for in the design and analysis of various components.
>
> The core support structure components affected by the 1.4-percent power uprate are discussed below. The primary inputs to the evaluations are the revised RCS temperature (as discussed in Section 2) and the gamma heating rates. The gamma heating rates took into account the 1.4-percent increase in core power.
>
> The reactor internals components subjected to heat generation effects (either directly or indirectly) are the upper core plate, the lower core plate, and the baffle-barrel region. For all of the reactor internal components, except the lower core plate and the upper core plate, the stresses and cumulative fatigue usage factors were unaffected by the 1.4-percent power uprate conditions, because the previous analyses remain bounding.

Based on the information provided by the licensee regarding the results of its structural evaluations with the changes to operating temperature, flow rates, and neutron fluences that result from the power uprate, the NRC staff agrees that the integrity of the RPV internals will be maintained. Therefore, the licensee's ability to meet the regulatory requirements in 10 CFR 50.46 regarding ECCS performance and maintaining a coolable core geometry will not be adversely impacted by the proposed uprate and is acceptable.

3.3.6 Control Rod Drive Mechanisms (CRDMs)

The pressure boundary portion of the CRDMs is that portion exposed to the vessel/core inlet fluid. The licensee evaluated the adequacy of the CRDMs by reviewing the IP2 current CRDM design specifications and stress report to compare the design-basis values of input parameters against the revised design conditions in Table 2-1 of the application report for the power uprate. The comparison shows that the current design temperature (650 °F) and pressure (2500 psia) are bounding for the 1.4% power uprate. Therefore, the existing stress and fatigue usage are not affected for the power uprate and continue to satisfy the allowable stress and fatigue usage limits.

Based on the information provided by the licensee regarding the CRDMs, the NRC staff agrees that the existing stress and fatigue usage are not affected by the uprate because no values of input parameters will be changed. The integrity of the CRDMs will be maintained such that the licensee will be able to meet the regulatory requirements in GDC-14 and other applicable portions of Appendix A to 10 CFR Part 50. The NRC staff concludes that the licensee has demonstrated the acceptability of the CRDMs for the 1.4% power uprate conditions, and therefore, the licensee's assessment is acceptable.

3.3.7 Reactor Coolant Pumps (RCPs)

The licensee assessed the existing design-basis analyses of the IP2 RCPs to determine the impact of the revised design conditions on them in Table 2-1. The proposed core power uprate would leave the RCS pressure unchanged. The limiting design parameter of RCP outlet temperature (reactor vessel inlet temperature), as provided in Table 2-1 of the application report, was decreased for the power uprate condition, in comparison to the design temperature defined in the RCP equipment specification. Typically, a higher RCP outlet temperature results in a greater actual stress. As a result of the evaluation, the licensee indicated that the current stress and CUFs in the stress reports for the IP2 RCPs remain bounding for the 1.4% power uprate. Based on the current design basis steam generator (SG) outlet temperature (515.5 °F) and the current flow rate of 85,800 gpm per loop, which are bounding for the power uprate, the licensee concluded that the current design-basis motor loads remain bounding and applicable for the proposed 1.4% power uprate.

On the basis of its review, the staff concurs with the licensee's conclusion that the RCPs, when operating at the proposed conditions with a 1.4 percent power increase from the current rated power, are bounded by the current design-basis motor loads and will remain in compliance with the requirements of the codes and standards under which the IP2 were originally licensed.

3.3.8 Pressurizer

The licensee evaluated the structural adequacy of the pressurizer and components for limiting locations at the pressurizer spray nozzle, the surge nozzle, and upper shell for operation at the uprated conditions. The evaluation was performed by comparing the values of key parameters in the current IP2 pressurizer stress report with the revised design conditions in Table 2-1 for the proposed power uprate. Section 7.8 of the application report provides the comparison of the values of current and uprated pressurizer design parameters. The analysis results demonstrate that the proposed IP2 1.4% power uprate will have a minimal effect on the pressurizer components. Table 7-8 of the application report compares the fatigue usages calculated with those from the original design basis. The licensee concluded that the pressurizer components meet the stress/fatigue analysis requirement of the ASME Code, Section III, for plant operation at the 1.4% power uprate conditions. Therefore, the NRC staff finds that the existing design-basis analyses remain valid for the proposed power uprate and is acceptable.

3.3.9 NSSS Piping and Pipe Supports

The proposed power uprate of IP2 involves an increase in the temperature difference across the RCS. The licensee evaluated the NSSS piping and supports by assessing the existing design-basis analysis as compared to the uprated power conditions, with regard to the system design parameters, transients, and the LOCA dynamic loads. The evaluation was performed for the reactor coolant loop (RCL) piping, primary equipment nozzles, primary equipment supports, and the pressurizer surge line piping. The licensee indicated that the NRC-approved methods, criteria and standards used in the existing design basis analysis for IP2 were used for the power uprate evaluation.

The licensee also indicated that the design transients used in the evaluation of the RCS piping systems and equipment nozzles are unchanged for the proposed power uprate. The potential for a slight increase in loop hydraulic forces due to the decrease in the cold leg temperature and the increase in water density at the power uprate conditions was offset by the existing margin in the current design-basis analysis. The RCL piping evaluation was performed based on United States of America Standard (USAS) B31.1, “Power Piping,” 1955 Code, which is the Code of record. However, the design-basis analysis was performed in accordance with the standards of USAS B31.1-1973 Edition. In addition, the pressurizer surge line was evaluated in accordance with the ASME Code, Section III, Subsection NB, 1986 Edition. The support structures were evaluated in accordance with the American Institute of Steel Construction (AISC) Specification, 1963 Edition, which is the Code of record. As a result of its evaluation, the licensee concluded that the existing stresses and loads remain bounding for the power uprate for the NSSS components including the reactor cooling loop piping, the primary equipment nozzles, the primary equipment supports, pipe supports and the auxiliary equipment (i.e. heat exchangers, pumps, valves and tanks).

On the basis of its review of the licensee’s submittal, the NRC staff concurs with the licensee’s conclusion that the existing analyses of stresses and loads on NSSS piping and supports, primary equipment nozzles, primary equipment supports, and auxiliary lines connecting to the primary loop piping will continue to show that these systems, structures, and components meet the design-bases criteria, as defined in the IP2 UFSAR, because sufficient margin exists. The existing analyses are, therefore, acceptable for the proposed power uprate.

Leak Before Break (LBB) Methodology

In Section 7.4.3 of the Application Report, the licensee concluded that:

> There is an insignificant change in loads due to the 1.4-percent power uprate parameters as indicated in Section 7.4.1. The effect of material properties due to the changes in RCS temperature, shown in Table 2-1, will have a negligible impact on the LBB margins shown in References 7-11 and 7-12. The existing conclusions of the LBB analyses discussed in References 7-11 and 7-12 remain applicable for the 1.4-percent power uprate for IP2.

> Therefore, based on the above assessment, it is concluded that the LBB margins will not change significantly and the conclusions of References 7-11 and 7-12 remain unchanged for the 1.4-percent power uprate for IP2.

Based on the changes in pressure, temperature, and operating loads expected to result from the proposed 1.4% power uprate, the NRC staff agrees with the licensee's conclusion that the effect of the proposed power uprate on the facility's existing LBB evaluations will be insignificant. However, due to recent events concerning primary water stress corrosion cracking (PWSCC) of Inconel 82/182 material, the NRC staff is in the process of examining the significance of this issue with respect to existing LBB evaluations. Currently, the NRC staff is evaluating what licensee actions, if any, are necessary to ensure that the technical bases for existing LBB approvals remain valid, and any concerns regarding the effect of PWSCC on existing LBB evaluations will be resolved separately from this power uprate submittal. As noted above, the NRC staff also expects the impact of the proposed power uprate, as well as the small projected increase in T_{hot}, on the PWSCC susceptibility of any Alloy 82/182 materials in lines approved for LBB at IP2, to be insignificant.

3.3.10 Steam Generators

The licensee assessed the existing structural and fatigue analyses of the Model 44F SGs at IP2. The revised design conditions in Table 2-1 for the power uprate are compared against the design parameters in the Model 44F SG stress reports. Based on the comparison of values of key input parameters, the licensee developed scaling factors which were used to scale up the original stress ranges and fatigue usage factors for the power uprate conditions. The evaluation was performed in accordance with the requirements of the ASME Code, Section III, 1965 Edition through the Summer 1966 Addenda, which is the Code of record for the SGs at IP2.

The calculated maximum stresses and cumulative fatigue usage factors for the critical SG components are provided in Table 7-6 of the application report. The results indicate that most of the calculated ranges of stress intensities are below the allowable limit of $3S_m$. For those components (i.e., divider plate, tube/sheel junction, etc.) where the ranges of primary plus secondary stress intensities exceed $3S_m$, a simplified elastic plastic analysis was done to justify the exceeding of $3S_m$ limit in compliance with the ASME Code. The calculated CUFs shown in Table 7-6 are below the allowable limit of unity. In its application, the licensee provided its evaluation of the flow-induced vibration for U-bend tubes due to the increased feedwater flow rate. The evaluation showed that the maximum fluid-elastic stability ratio and the maximum vibration induced displacement are within the allowable limits and therefore, acceptable for the proposed power uprate.

On the basis of its review, the staff concludes that the licensee has used acceptable methods and values of analysis inputs and has demonstrated the maximum stresses and CUFs for the critical SG components are within the Code allowable limits. Therefore, the staff finds them acceptable for the proposed 1.4% power uprate.

<u>Structural Integrity Evaluation</u>

The licensee performed a structural integrity evaluation for their SGs that focused on the primary-side components and secondary-side components. The main primary-side components that were assessed were the divider plate, tubesheet and shell junctions, tube-to-tubesheet weld and the SG tubes. The main secondary-side components that were assessed were the feedwater nozzle, secondary-side manway studs and the steam nozzle.

The values of RCS parameters for primary-side transients from current and power uprated conditions were compared to determine scale factors that were applied to the baseline analyses maximum stress ranges and fatigue usage factors. The baseline analyses results for various components were then updated for the 1.4% power uprate conditions.

The 1.4% power uprate structural evaluation was performed for the 3127 MWt NSSS power and 25% Steam Generator Tube Plugging (SGTP) conditions. The licensee compared the results of the structural evaluations with ASME Code Section III acceptance criteria and concluded that all components analyzed meet applicable ASME Code Section III limits.

The NRC staff finds the licensee's evaluation to be acceptable since the Code limits continue to be met.

Evaluation of Primary-to-Secondary-Side Pressure Differential

The licensee performed an analysis to determine if the applicable ASME Code limits for Model 44F SG design primary-to-secondary ΔP are exceeded for any applicable transient conditions for the 1.4% power uprate parameters. The licensee indicated that the maximum allowable normal ΔP is limited to 1700 psi and under upset conditions is limited to 1870 psi. The evaluation was performed for the 25% SGTP condition which results in the maximum primary-to-secondary side pressure differential.

The licensee compared the results of the analyses using uprate parameters with the ASME Code limits (normal and upset) and concluded the requirements of the ASME Code continue to be satisfied. Therefore, the NRC staff finds the licensee's evaluation to be acceptable.

Evaluations for Repair Hardware

The IP2 replacement SGs were placed in service in 2000 and these SGs have several Westinghouse shop welded plugs installed. The licensee re-evaluated the installed welded plugs for the operating conditions and transients associated with the 1.4% uprated power operation. In anticipation of future needs, the licensee also evaluated mechanical plugs, tube undercuts and a collar-cable tube stabilizer for the 1.4% power uprated conditions.

The licensee performed analyses for mechanical plugs (both the long and short 7/8-inch ribbed plugs) and concluded that the design met the applicable stress and retention acceptance criteria for use under the operating conditions associated with the 1.4% power uprate. The shop welded plug design was analyzed and determined to satisfy all ASME Code, Section III allowable values for the 1.4% power uprated condition. The stress evaluation for tube undercuts, if needed to support tube plugging and sleeving operations, indicated that stresses are within the ASME Code allowable values and that fatigue usage factors are acceptable. The Westinghouse collar-cable stabilizer design is a stainless steel wire cable within a series of stainless steel collars to provide both flexibility and damping. The licensee's analysis showed that if a random wear couple were to exist between a severed tube and the cable, the central wire cable would remain intact for the life of the installation. Deleterious contact between the stabilizer and adjacent tubes was also demonstrated not to occur.

The NRC staff finds the licensee’s evaluation for SG repair hardware acceptable since the stresses are within Code allowances and the power uprate conditions are not predicted to have deleterious effects on repair hardware.

RG 1.121 Analysis

NRC RG 1.121, “Bases for Plugging Degraded PWR Steam Generator Tubes”, describes an acceptable method for establishing the limiting safe condition of degradation in the tubes beyond which tubes found defective by the established inservice inspection shall be removed from service. The allowable tube repair limit, in accordance with RG 1.121, is obtained by incorporating, into the resulting structural limit, a growth allowance for continued operation and an allowance for eddy current measurement uncertainty.

The licensee performed an analysis to define the structural limits for an assumed uniform thinning mode of degradation in both the axial and circumferential directions. The assumption of uniform thinning is generally regarded to result in a conservative structural limit for all flaw types occurring in the field. In addition, the licensee used ASME Code minimum material properties which it determined provides added conservatism. The licensee factored in information based on the predicted growth rates and non-destructive evaluation uncertainties and concluded that the current TS repair limit of 40% through wall is adequate.

The NRC staff finds the licensee’s evaluation and reasoning to be acceptable because it follows the guidance in RG 1.121.

Tube Vibration and Wear

The licensee evaluated the potential effects of the 1.4% power uprate on SG tube vibration and wear.

The original vibration and wear baseline analysis results were modified to account for anticipated changes in the thermal-hydraulic characteristics of the secondary side of the SG resulting from the 1.4% power uprate. The analysis indicated that large amplitudes of vibration and tube-to-tube contact would not occur. The licensee’s analysis indicated that a slight increase in tube wear was possible, but that the increased level of wear was not significant and would occur over many cycles easily detectable by normal SG inspection techniques. The licensee concluded that the increase in wear rate resulting from the 1.4% power uprate will not result in unacceptable wear.

The NRC staff finds the licensee’s evaluation and reasoning to be acceptable because large amplitudes of vibration tube-to-tube contact or significant increases in tube wear from support structures are not predicted to occur.

Tube Integrity

The licensee discussed the effects of local tube degradation resulting from operating loads and the chemical environment in the SG. These effects were discussed relative to the TS repair limit and SG inspection frequency. The licensee concluded that the 1.4% power uprate is not expected to have a significant impact on corrosive SG tube degradation. This is based on: the expected corrosion resistance of thermally treated alloy 600 SG tubing; IP2's low relative operating temperature; the relatively small increase in operating temperature as a result of the 1.4% power uprate and industry operating experience with thermally treated alloy 600 SG tubing. The licensee also concluded that the TS repair limit of 40% was adequate and that the required frequency of inspection was not affected significantly by the 1.4% power uprate.

The NRC staff finds the licensee's evaluation and reasoning to be acceptable the power uprate should not have a significant impact on parameters that affect SG tube corrosion.

Flow-Accelerated Corrosion (FAC)

Flow-accelerated corrosion (FAC) is a corrosion mechanism causing wall thinning of high energy pipes in the power conversion system which may lead to their failure. Since failure of these pipes may result in undesirable challenges to the plant's safety systems, the licensee has a program for predicting, inspecting, and repairing or replacing the components whose wall thinning exceeds the values required for their safe operation. The program uses the Electric Power Research Institute (EPRI) developed CHECKWORKS computer code for predicting thinning of the walls in the components subjected to FAC. In the submittal, the licensee stated they would revise the code to incorporate flow and process system conditions as determined for the 1.4% power uprate conditions, and that the results of the upgraded code would be factored into future surveillance/pipe repair plans.

The NRC staff considers the licensee's FAC program to be adequate for ensuring integrity of the high energy pipes because it uses an acceptable computer code for predicting wall thinning.

Summary

For the reasons set forth above, the NRC staff finds the licensee's evaluation and reasoning to be acceptable and, therefore, concludes that the proposed 1.4% power uprate for IP2 will not have significant impact on SG structural integrity or flow accelerated corrosion.

3.4 Electrical Equipment Design

The electrical distribution system has been previously evaluated to conform to GDC-17. The offsite power system includes two or more physically independent circuits capable of operating independently of the onsite standby power sources. The NRC staff's review covers the information, analyses and documents for the offsite power system and the stability studies for the electrical transmission grid. The focus of the review relates to the basic requirement that loss of the nuclear unit, the largest operating unit on the grid, or the loss of the most critical transmission line will not result in the loss of offsite power to the plant. Branch Technical Position (BTP) ICSB-11, "Stability of Offsite Power Systems," and GDC-17 outline an acceptable approach to addressing the issue of stability of offsite power systems. Acceptance criteria are based on GDC-17. Specific review criteria are contained in SRP Sections 8.1 and 8.2, Appendix A to 8.2 and BTPs PSB-1 and ICSB-11.

As described in the licensee's application, the main generator is rated at 1439.2 MVA at a 0.91 power factor. The main generator provides power through the isolated phase bus at 22 kV to both the main transformer and the unit auxiliary transformer. The generator voltage is stepped up through the main transformer to a 345 kV system. The preferred ac power source provides offsite ac power to the auxiliary power distribution system for the startup, operation, or shutdown of the station. The preferred ac power also provides a source of offsite ac power to all emergency loads necessary for the safe shutdown of the reactor.

3.4.1 AC Onsite Power Systems

The onsite ac power system includes those standby power sources, distribution systems, and auxiliary supporting systems provided to supply power to the safety-related equipment. The NRC staff's review covers the descriptive information, analyses, and referenced documents for the ac onsite power system. Acceptance criteria are based on GDC-17 as it relates to the capability of the ac onsite power system to perform its intended functions during all plant operating and accident conditions. Specific review criteria are contained in SRP Section 8.1 and 8.3.1.

The onsite emergency power supply consists of three independent emergency diesel generators (EDGs). The emergency bus loading was evaluated to determine any load increases that would affect it as a result of the 1.4% power uprate. A review of the electrical loading associated with each EDG determined that the loads are unaffected by 1.4% power uprated conditions. Since no new loads or EDG changes were identified, the existing EDG protection schemes are similarly unaffected. Therefore, the EDGs are not affected by uprate conditions.

The NRC staff has reviewed the licensee's submittal for the effect of the proposed power uprate on the onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed power uprate on the system's functional design since the safety system loads are unaffected by the uprate conditions. For the same reason, the NRC staff further concludes that the ac onsite power system will continue to meet the requirements of GDC-17 following implementation of the proposed power uprate. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the onsite ac power system.

3.4.2 Onsite Direct Current (DC) Power Systems

The dc power systems include those dc power sources and their distribution systems and auxiliary supporting systems provided to supply motive or control power to safety-related equipment. The NRC staff's review covers the information, analyses, and referenced documents for the dc onsite power system. Acceptance criteria are based on GDC-17 and 10 CFR 50.63 as they relate to the capability of the onsite electrical power to facilitate the functioning of structures, systems, and components important to safety. Specific review criteria are contained in SRP Sections 8.1 and 8.3.2.

The NRC staff has reviewed the licensee's analyses for the effect of the proposed power uprate on the dc onsite power system and concludes that the licensee has adequately accounted for the effects of the proposed power uprate on the system's functional design. The NRC staff further concludes that the dc onsite power system will continue to meet the requirements of GDC-17 following implementation of the proposed power uprate since no new loads were added to the system. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to the dc onsite power system.

3.4.3 Station Blackout (SBO)

SBO refers to the complete loss of ac electric power to the essential and nonessential switchgear buses in a nuclear power plant for a specified duration. SBO involves the loss of offsite power concurrent with turbine trip and failure of the onsite emergency ac power system. SBO does not include the loss of available ac power to buses fed by station batteries through inverters or the loss of power from “alternate ac sources” (AAC). The NRC staff’s review focuses on the impact of the proposed power uprate on the plant’s ability to cope with and recover from an SBO event. Requirements for SBO are set forth in 10 CFR 50.63. Specific review criteria are contained in SRP Section 8.1 and Appendix B to SRP 8.2.

IP2 uses internal combustion gas turbines as an AAC power source to operate systems necessary for the required SBO coping and recovery. The AAC power sources have sufficient capacity and capability to provide power to the shutdown buses within 1 hour of the SBO event for a duration of 8 hours.

The methodology and assumptions associated with the SBO analysis with regard to equipment operability are unchanged with the uprate. There is no change in the ability of the turbine-driven auxiliary feedwater pumps, supplied with steam from the SGs, to support reactor heat removal due to the 1.4% power uprate. The licensee analyzed the following topics:

For the current power rating, the stated volume of water needed for 8 hours of decay heat removal and primary system cooldown is 142,850 gallons. For the power uprate, a small increase (less than 1%) in the volume is needed. Since the TSs require that a minimum of 360,000 gallons of water must be available in the condensate storage tank (CST) during plant operation, there continues to be a large margin between the minimum required volume of water in the CST and the volume of water needed for coping with an SBO. The licensee evaluated the shutdown loads following a trip and a loss of offsite ac power at 1.4% power uprate conditions and showed no increases in load on the station batteries. Accordingly, there is no change in the ability of the plant to cope with an SBO event under the 1.4% power uprated condition. The 1.4% power uprate will have no effect on air-operated valve operation during an SBO event. The licensee also evaluated the effect of loss of ventilation in the auxiliary feedwater pump area and determined that it bounds the conditions identified for the 1.4% power uprate. Containment isolation valves needed to maintain containment integrity are closed and locked and covered administratively. The 1.4% power uprate will have no effect.

The NRC staff has reviewed the licensee’s analyses of the effect of the proposed power uprate on the plant’s ability to cope with and recover from an SBO event for the period of time established on the plant’s licensing bass. For the reasons set forth above, the NRC staff concludes that the licensee has adequately evaluated the effects of the proposed power uprate on SBO and demonstrated that the plant will continue to meet the requirements of 10 CFR 50.63 following the implementation of the proposed power uprate. Therefore, the NRC staff finds the proposed power uprate acceptable with respect to SBO.

3.4.4 Environmental Qualification (EQ) of Electrical Equipment

The term “environmental qualification” applies to equipment important to safety to assure this equipment meets its specified performance requirements when it is subjected to the conditions predicted to be present when it must perform its safety function up to the end of its qualified life. The NRC staff’s review covers the environmental conditions that could affect the design and safety functions of electrical equipment including instrumentation and control. The NRC staff’s review is to ensure compliance with the acceptance criteria, thus ensuring that the equipment continues to be capable of performing its design safety functions under the most severe accident and post-accident environmental conditions. Acceptance criteria are based on 10 CFR 50.49 as it relates to specific requirements regarding the qualification of electrical equipment important to safety that is located in a harsh environment. Specific review criteria are contained in SRP Section 3.11.

In accordance with 10 CFR 50.49, safety-related electrical equipment must be qualified to survive the radiation environment at their specific location during accident and post-accident operating conditions. The licensee assessed the electrical equipment important to safety to ensure that the existing qualification remain adequate. In its April 3, 2003, letter, the licensee provided additional information in support of the NRC staff's review of EQ issues. The licensee evaluated equipment qualification for safety-related electrical equipment inside containment for a LOCA. The resultant temperature, pressure, humidity and radiation profiles due to the 1.4% power uprate are bounded by environmental conditions used in the current analysis at 102% power. The licensee evaluated EQ for safety-related electrical equipment located outside the containment based on high energy line break (HELB) conditions. The proposed 1.4% power uprate does not create any new high-energy lines and the temperature, pressure, humidity and radiation profiles remain bounded by environmental conditions used in the current analysis at 102% power. Lines in which 1.4% power uprate conditions are less severe than the current operation (i.e., temperature and pressure) are considered acceptable for the uprate. For systems that are considered unaffected by the 1.4% power uprate, the current HELB analysis is also considered unaffected. The 1.4% power uprate does not introduce any additional HELB areas. Therefore, the EQ of the electrical components will not be affected by the 1.4% power uprate.

3.4.5 Grid Stability

The licensee evaluated the impact of power uprate on grid stability in January 2001 and determined that there is no anticipated significant effect on grid stability. In its April 3, 2003, letter, the licensee provided additional information in support of the NRC staff's review of the grid stability issue. The licensee analyzed the grid stability by using the stability data provided by the New York State ISO (Independent System Operator). Stability plots compared the response of several IP2 generator variables before and after the uprate, as well as selected 345 kV voltages. The study concluded that the system is shown to be stable for all the contingencies, and the plots indicate a very similar response at IP2 before and after the uprate. Therefore, the proposed 1.4% power uprate will not impact grid stability analysis.

The NRC staff reviewed the licensee's submittal and, for the reasons set forth above, concludes that grid stability will be maintained with the anticipated power uprate and, therefore, the design is acceptable.

.4.6 Plant Equipment

Main Generator

The main generator is rated at 1439.2 MVA (based on 75 psig hydrogen pressure) at 0.91 power factor (pf). The current power level of 3071.4 MWt is 1022 MWe. For the 1.4% power uprate, at 3114.4 MWt, the main generator is anticipated to operate at an output as high a 1042.4 MWe. The generator's capability curves show that at 1042.4 MWe the generator is capable of exporting 630 megavolt-amp reactive (MVAR) (lagging power factor of 0.856) and capable of importing 510 MVAR (leading power factor of 0.898). Therefore, the generator is capable of operating at approximately 1218 MVA lagging and 1160.5 MVA leading when operating at a power uprate level of 1042.4 MWe. The exciter has the capability to support machine operation within its nameplate rating and within the capability curve of the machine for the leading and lagging case of VAR production. The applied main generator protection schemes are intended to limit machine damage for internal fault conditions and to prevent machine damage during abnormal operating or external fault conditions. A review of one-line diagrams and protective relay settings confirms that the applied schemes are dependent upon machine ratings and design parameters, and the design of the connected system. The main generator performance is bounded by existing design and is not impacted by the power uprate.

The NRC staff reviewed the licensee's application and, for the reasons set forth above, concludes that it will continue to operate safely at the anticipated power uprate and, therefore, the design is acceptable.

<u>Main Transformer (MT)</u>

The main generator delivers its power output to 2 MTs (MT 21 and 22). Main transformers 21 and 22 are Westinghouse transformers, nameplate rating 20.3/345 kV, 542 MVA FOA @ 55 °C, 3-phase, 60 Hz and 607 MVA FOA @ 65 °C, 3-phase, 60 Hz. MTs 21 and 22 each have an impedance of 16.08% at the 55 °C rating, therefore, the load will divide evenly between the transformers.

The total capacity of the MT bank is 1214 MVA at the 65 °C rating. With the generator operating at the 1.4% power uprate level, with a lagging power factor (pf), and allowing for the load of the unit auxiliary transformer (UAT) and MT losses, the required capacity of the MTs at the 1.4% power uprate level is within the 65 °C rating. Therefore, the MTs are adequate for the 65°C rating when the generator is operating at the 1.4% power uprate level at lagging pf. With the generator operating at a leading pf, and allowing for the load of the UAT and MT losses, the MVAR imported by the generator will be limited to keep the loading on the MTs within the 65°C rating at the 1.4% power uprate level.

The NRC staff reviewed the licensee's submittal and concluded that the MTs will continue to operate within the nameplate rating at 65 °F under the anticipated power uprate loading and, therefore, the design is acceptable.

<u>Isophase Bus</u>

The isolated phase bus duct (Iso-Phase) connects the main generator to the primary windings of the MTs and the UAT. The Iso-Phase Bus system is organized into segments. The first segment runs from the generator terminals to the point where the main bus splits into the 2 segments that run to the 2 MTs. This first segment has a forced air-cooled rating of 32 kA at 22 kV, 65 °C. The second segment of the main bus runs from the split to each MT. These segments have a forced air-cooled rating of 16 kA at 22 kV, 65 °C. The third segment runs from the main bus tap to the UAT. This segment has a self-cooled rating of 1.5 kA at 22 kV. This segment does not have a forced-cooled rating.

The transformer test report shows the 2 MTs have identical MVA ratings and impedances. Since the current splits evenly between the transformers in proportion to the impedance, current to each MT primary winding will be the same. The 16 kA portion of the bus between the split and UAT tap is the most limiting since it carries the generator output to one MT plus the UAT load. The licensee will limit the amount of MVAR exported by the generator to maintain the loading within the rating of the most limiting section of iso-phase bus at the 1.4% power uprate level. Therefore, the Iso-phase Bus system will remain within its design parameters and is found to be acceptable.

Unit Auxiliary Transformer

The UAT nameplate rating is 22/6.9 kV, 43 MVA FOA @ 55°C, 3-phase, 60 Hz. The transformer is equipped with a +10/-5 % load tap changer. The UAT supplies power to balance-of-plant systems under normal operating conditions. The BOP systems most impacted are the feedwater system, the condensate system, and the heater drains system. The analysis of these systems at the increased power level produced new pump operating points. The main feedwater pumps are turbine driven so their new operating point does not affect the station electrical distribution system. The combined total brake horsepower (BHP) of the heater drain pumps and condensate pumps, used in the load flow calculation, envelopes the change in operating points and there would be no net station electrical load increase for these pumps. Therefore, the NRC staff finds this acceptable.

Station Auxiliary Transformer (SAT)

The SAT nameplate rating is 138/6.9 kV, 43 MVA FOA @ 55°C, 3-phase, 60 Hz. The transformer is equipped with a +10/-5 % load tap changer. The SAT provides power to BOP systems under abnormal operating conditions. The BOP systems most impacted are the feedwater system, the condensate system, and the heater drains system. The analysis of these systems at the increased power level produced new pump operating points. The main feedwater pumps are turbine driven so their new operating point does not affect the station’s electrical distribution system. The combined total BHP of the heater drains pumps and condensate pumps, used in the load flow calculation, envelopes the change in operating points and there would be no net station electrical load increase for these pumps. Therefore, the NRC staff finds this acceptable.

Motor-Driven Pumps

The electrical equipment that supports the mechanical systems are typically motors, cables, and circuit breakers. The licensee has determined that some medium voltage motors on non-safety-related 6.9 kV switchgear have revised operating points. The condensate pumps, rated at 3000 HP each, and the heater drain pumps, rated at 1000 HP each, experience a brake horsepower increase. Since the existing motor drives will operate at a brake horsepower less than the design rating during full load conditions at the 1.4% power uprate, no motor replacements will be required at the 1.4% power uprate.

The NRC staff has reviewed the licensee’s submittal and, for the reasons set forth above, concludes that the brake horsepower of the condensate and heater drain pumps remain below the design rating and the design is, therefore, acceptable.

3.4.7 Summary

The NRC staff has evaluated the effect of power uprate on the necessary electrical systems and environmental qualification of electrical components. Results of these evaluations, as set forth above, show that the increase in a core thermal power would have negligible impact on grid stability, SBO, or the environmental qualification of electrical components. This is consistent with GDC 17, 10 CFR 50.63, and 10 CFR 50.49, and the proposed change is, therefore, acceptable.

3.5 System Design

3.5.1 Reactor Coolant System (RCS)

The RCS operating conditions are changed slightly at uprated power. The steady-state RCS pressure (2235 psig), no-load RCS temperature (547 ^{o}F), RCS flow (80,700 gpm per loop) and hot-leg temperature have not changed. However, the Tcold is decreased from 547.4 °F to 546.4 °F. Therefore, the RCS temperatures associated with the power uprate are still within the bounds of the original design temperature of 650 °F for RCS and 680 °F for pressurizer. Sufficient core cooling under power uprate conditions is verified by various plant transient and safety analyses. The performance of natural circulation cooldown was analyzed with a 2% power measurement uncertainty with acceptable results which is applicable to the uprated power conditions. The NRC staff find that the changes of RCS operating parameters associated with power uprate are acceptable based on the acceptable results of the safety analyses addressed in Section 3.2.

3.5.2 Safety Injection System (SIS)

The adequacy of the safety injection system (SIS) during the injection and sump recirculating phases following a LOCA was verified in the current licensing analysis at the current power level with the existing 2% uncertainty allowance and is applicable to the uprated power conditions. For the non-LOCA events, the performance of the SIS is verified by various safety analyses performed in support of the power uprate. There are no system modifications needed to support power uprate. The NRC staff agrees with the licensee’s assessment based on the acceptable results of the safety analyses addressed in Section 3.2.

3.5.3 Residual Heat Removal (RHR) System

The 1.4% power uprate affects the plant cooldown time since 100% power level was assumed in the cooldown analysis of record. Therefore, updated cooldown cases to account for the 1.4% uprate conditions were analyzed. The licensee has calculated the ability of the RHR system to achieve cold shutdown under the power uprate conditions with the spent fuel pool (SFP) heat load. In accordance with the methodology described in the UFSAR, the licensee calculated that cold shutdown can be achieved (1) within 33 hours assuming both RHR trains operable with SFP heat load and (2) within 71.89 hours assuming only one RHR train operable without SFP heat load. The results of the licensee’s new calculation confirms that the RHR cooldown capacity meets the 72-hour Appendix R requirement, and the normal plant cooldown time changes do not affect plant safety. Based on its evaluation, the licensee has concluded that system modifications are not required to accommodate the power uprate. The NRC staff reviewed the licensee’s submittal and, for the reasons set forth above, agrees with the licensee’s assessment.

3.5.4 Spent Fuel Pool Storage and Cooling

The design basis for the SFP cooling system includes the capability to maintain the SFP temperature below 140 °F following the discharge of 72 fuel assemblies, and below 180 °F following the discharge of a full core. The design criteria are based, in part, on the calculated time it would take for the SFP to boil in the event of loss of SFP cooling.

The required post-accident functions are analyzed for 102% of the current rated core thermal power. That analysis bounds the 1.4% power uprate conditions; therefore, it continues to be acceptable. Accordingly, the staff concludes that the power uprate is acceptable with respect to SFP cooling.

3.5.5 Balance of Plants (BOP) Systems and Motor-Operated Valves (MOVs)

The licensee stated that the IP2 BOP systems were reviewed for potential effects due to the 1.4% power uprate to 3114.4 MWt reactor core power. The BOP systems that could potentially be affected by the 1.4% power uprate are the:

- Main Steam and Steam Dump System (SDS)
- Condensate and Feedwater Systems (C&FS)
- Condenser/Circulating Water
- Extraction Steam System
- Feedwater Heaters and Drains
- Service Water System (SWS)
- Component Cooling Water System (CCWS)
- Containment Cooling and Filtration (CC&F) Systems
- Other Heating, Ventilation and Air Conditioning (HVAC) Systems
- Instrumentation and Controls (I&C)
- Piping and Support Evaluation
- Spent Fuel Pool Cooling System
- Main Turbine

The licensee evaluated the adequacy of the BOP systems based on comparing the existing values of design bases parameters with the uprated values of input parameters in Table 2-1 for the core power uprate conditions. The BOP piping systems evaluated for the power uprate are the main steam and steam dump, condensate and feedwater, condenser and circulating water, extraction steam, feedwater heaters and drains, service water, component cooling water, containment cooling and filtration, spent fuel pool cooling, and main turbine systems. In its April 3, 2003, response to the staff's request for additional information, the licensee provided change factors for the affected BOP piping. The change factors are defined as the ratios of current design-basis temperatures minus 70 °F, pressures, or flow rates to the corresponding values for the power uprate conditions. The licensee also indicated that the review of a sample of representative calculations shows that at least a 5% margin to the design limit exists to accommodate the changes (<3 percent) due to the proposed power uprate. As a result, the licensee concluded that the existing design basis analyses, using maximum differential temperatures and pressures for normal operation and worst case conditions, for the BOP piping, pipe supports, and components remain bounding for the uprated power level of 3114.4 MWt at IP2.

The licensee also reviewed the programs and generic letter issues as they pertain to the power uprate. In its application report, the licensee confirmed that there are no changes to the IP2 MOV program as a result of the 1.4% power uprate. Safety-related valves were not found to be impacted by the 1.4% power uprate and are, therefore, acceptable. This determination was confirmed by verifying that changes in system operating parameters, such as temperature, pressure and flow rate, were bounded by the design-basis values. Additionally, in its application report, the licensee assessed the impacts of the 1.4% power uprate on the GL 89-10 and GL 96-05 programs and found them to be acceptable.

The licensee indicated that the current evaluation of GL 95-07 associated with the pressure locking and thermal binding was based on the worst-case for the containment pressure and temperature calculation, which remains bounding for the 1.4% power uprate. In its April 3, 2003, response to the staff's request for additional information, the licensee indicated that the existing evaluation for GL 96-06 was performed at 102% of the current power and is therefore, bounding for the proposed power uprate of 101.4%. For the reasons set forth above, the NRC staff concurs with the licensee's conclusions that the power uprate will have no adverse effects on the performance of safety-related valves and that conclusions reached based on implementation of provisions in GL 95-07, GL 96-06, GL 89-10 and GL 96-05 programs remain valid.

As a result of the above evaluation, the staff concludes that the BOP piping, pipe supports and equipment nozzles, and valves remain acceptable and will continue to function within the existing design basis for the proposed 1.4% power uprate.

3.5.6 Radioactive Waste Processing Systems

Since reactor operation at the increased power level after the 1.4% power uprate will increase the release rate of radioactive isotopes into the reactor coolant, the activity levels associated with liquid and gaseous effluents will increase proportionally. However, the volume of solid waste would not be expected to increase proportionally since equipment performance and system operation is not appreciably changing. The radioactive wastes are processed through the solid, liquid, and/or gaseous radioactive waste system. The licensee states that the wastes generated will be processed within the plant and there would be minimal effects from the additional waste generation. The 10 CFR Part 50, Appendix I evaluation for IP2 was based on a power level of 3216 MWt and thus encompasses plant operation at a core power level of 3133.1 MWt (3114.4 MWt plus 0.6% power level uncertainty).

Based on this statement and experience gained from the review of power uprate applications for similar PWR plants, the NRC staff concludes that the solid, liquid, and gaseous radioactive waste systems are acceptable for the plant power uprate operations.

3.5.7 Containment System

The licensee assessed the existing containment integrity analysis to ensure that the maximum pressure inside the containment would not exceed the containment design pressure if a design-basis LOCA or MSLB inside containment should occur during plant operation. The review also established the pressure and temperature for environmental qualification and operation of safety-related equipment located inside the containment. The results of the review follow.

<u>Containment Integrity Analysis - Loss-of-Coolant Accident</u>

The licensee states that the current mass and energy release data for input into the containment response analysis assumed a core thermal power of 3071.4 MWt, plus an additional 2% power measurement uncertainty allowance. The results of this analysis bounds the power uprate level of 3114.4 MWt, a 1.4% uprate with a 0.6% uncertainty. Therefore, the mass and energy release data for the LOCA bound the power uprate conditions, and the peak LOCA containment pressure and temperature will not be affected by the power uprate.

The licensee also conducted a short-term LOCA mass and energy release calculation to support the reactor cavity and loop subcompartment pressurization analyses. These analyses are performed to ensure the structural integrity of walls in the immediate proximity of the postulated break location to withstand the short pressure pulse (generally less than 3 seconds) that accompanies a LOCA within the region. The analysis inputs that may potentially change with the 1.4% power uprate conditions are the initial RCS fluid temperatures. Since the critical portion of this event last for less than 3 seconds, the single effect of reactor power is not significant. The licensee found that the critical flow correlation used in the mass and energy releases for this analysis provides an increase in the mass and energy release for a slightly lower fluid temperature. The licensee states that the limiting RCS conditions for pressure and temperature bound the proposed 1.4% uprate. Therefore, the current licensing basis for the short-term subcompartment pressurization analysis is unaffected by the power uprate.

<u>Containment Integrity Analysis - Main Steam Line Break (MSLB)</u>

The licensee states that mass and energy releases and containment integrity were evaluated for an MSLB based on the 1.4% power uprate conditions. As indicated by the values for design parameters for the 1.4% power uprate in Table 2-1 in its application report, the licensee determined that the parameters either remain unchanged or are bounded by the current analysis values in the mass and energy calculations. Therefore, the results of the containment integrity analysis for the MSLB is not affected by the 1.4% power uprate.

Based on review and assessment of the information provided in the licensee’s submittal, the NRC staff has determined that the peak LOCA, as well as MSLB containment pressure and temperature, will not be affected by the power uprate, and that the containment integrity analysis at the proposed uprated power is bounded by current LOCA and MSLB analysis. Therefore, the staff concludes that the amendment is acceptable with respect to these issues.

3.6 Other Areas of Review

3.6.1 10 CFR Part 50, Appendix R, Fire Protection

The licensee stated that the emergency lighting and RCP oil collection sections are not affected by the 1.4% power uprate. However, the 1.4% power uprate will affect plant cooldown times since no additional margin has been applied to the core power level assumed in the cooldown analysis of record. The licensee performed a single-train cooldown analysis to support the worst-case scenario for the 10 CFR Part 50, Appendix R fire hazards analysis. The worst-case scenario included in part: reduced RHR system cooldown capacity due to RHR pump miniflow, and increased heat load on the component cooling water system due to additional heat generated in the SFP from the 1.4% increase in core power. The analysis indicated that the RHR system was capable of achieving RCS cold shutdown (below 200 °F) in less than 72 hours after reactor shutdown, as is required by Appendix R (see Section 6.1.3 of the Application Report included as part of the December 12, 2002, application and Section 3.5.3 of this SE).

For a postulated fire with a loss of offsite power (LOOP), IP2 utilizes one of three internal combustion gas turbines as an ac power source to operate the applicable safe shutdown systems. An Appendix R gas turbine is also available at IP2 as a permanently installed alternate AC power supply to enhance the plant's alternate safe shutdown capability. The licensee stated that the Appendix R gas turbine loads are not affected by the 1.4% power uprate.

Since the 72-hour cooldown requirement is maintained for the power uprate conditions and the Appendix R gas turbine loads are not affected by the 1.4% power uprate, the NRC staff concludes that the safe shutdown capability, with regard to Appendix R requirements, is not affected by the power uprate.

3.6.2 Human Factors

The NRC staff reviewed the following operator performance topics discussed in the licensee's application.

3.6.2.1 Plant Procedures

The licensee stated that the power uprate has no significant effect on plant operating procedures. Where changes are necessary, the procedures will be revised or updated in a manner consistent with any other plant modification. The licensee also stated that procedure limitations on power operations due to BOP equipment unavailability, such as updated neutron flux trip setpoints with inoperable MSSVs, will be revised as necessary to account for the increase in core power to 3114.4 MWt. Those procedures needed for the operation and maintenance of Caldon LEFM Check System are being revised as necessary to reflect installation of the Check System. Specific actions to be taken when the Caldon LEFM Check System is inoperable are addressed in Section 3.1 above.

The NRC staff finds that the licensee's response is satisfactory because the procedures will be revised to incorporate the Caldon LEFM Check System prior to implementation of the power uprate. In addition, the licensee will treat plant procedure changes due to the power uprate in a manner consistent with any other plant procedure change.

3.6.2.2 Changes to Risk-Important Operator Actions Sensitive to Power Uprate

The licensee stated that engineered safety features (ESF) system design and setpoints and procedural requirements already bound the proposed power uprate. The responses of the reactor operators to any event will be essentially unaffected by a change in RTP.

There will be minimal impact on alarms, controls, and displays for the 1.4% uprate. The Caldon LEFM Check System will have alarms in the control room to alert operators of conditions that impair its availability and accuracy. No other alarm impacts are expected. It is not anticipated that any existing alarms will be modified or deleted. Alarms will be re-calibrated as necessary to reflect small setpoint changes. However, no significant or fundamental setpoint changes are anticipated. Also, the operator response to existing alarms is anticipated to remain as before.

When the power uprate is implemented, the nuclear instrumentation system will be adjusted to indicate the new 100% RTP in accordance with TS requirements and plant administrative controls. Since the power uprate is predicated on the availability of the system, procedural guidance will be implemented to facilitate operation when the Caldon LEFM Check System is unavailable. The reactor operators will be trained on the changes in a manner consistent with any other design modification.

The power uprate will be reflected in the plant simulator. These changes should be virtually transparent to the reactor operators.

The NRC staff finds that the licensee’s response is satisfactory because the licensee has adequately addressed the question of operator actions sensitive to the power uprate and shown that the responses of the reactor operators to any event will be essentially unaffected by 1.4% increase in the rated thermal power. The licensee will implement procedures and guidance as necessary for operator actions when Caldon LEFM Check System is unavailable.

3.6.2.3 Changes to the Plant Integrated Computer System (PICS)

The licensee stated that only process parameter scaling changes will be made, as necessary, to the PICS, and that there are no other impacts to the PICS due to the 1.4% power uprate.

The NRC staff finds that the licensee’s response is satisfactory because the licensee will identify and make the necessary scaling changes to the PICS as a result of the power uprate.

3.6.2.4 Changes to the Operator Training Program and the Control Room Simulator

The licensee's response to this question is included in the discussion in Section 3.6.2.2. The NRC staff finds the licensee’s response satisfactory, because the licensee has adequately described how the changes to operator actions will be addressed by training and how the simulator will accommodate the changes.

3.6.2.5 Human Factors Evaluation Summary

The NRC staff concludes that the previously discussed review topics associated with the proposed power uprate have been satisfactorily addressed. The NRC staff further concludes that the power uprate will not adversely affect simulation facility fidelity, operator performance, or operator reliability.

3.7 Technical Specification Changes

The licensee submitted to the NRC the proposed TS changes in support of safe operations in the IP2 plant at an uprated power level of 3114.4 MWt. The following is the NRC staff review of the TS changes.

<u>Licensed Core Thermal Power Levels - TS Section 1.1</u>

The rated thermal power level would be changed from 3071.4 MWt to 3114.4 MWt. The TS change reflects the actual proposed change in the plant and it is consistent with the results of the licensee’s supporting safety analyses. Consistent with its conclusions set forth throughout this SE, the NRC staff finds this proposed change acceptable.

Safety Limits - TS 2.1.1, Figure 2.1-1

IP2 TS Figure 2.1-1 regarding Safety Limits shows the loci of points of rated power, reactor coolant system pressure and average temperature for which the calculated DNBR is no less than the safety limit value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. The licensee proposed changes to Figure 2.1-1 to reflect the 1.4% power uprate. The value for 100% rated thermal power is changed from 3071.4 to 3114.4 MWt in the revised safety limits. The licensee has demonstrated in its supporting safety analyses that with the plant protection system functioning, the results of transient and accident analyses show that the proposed safety limits are not violated. Consistent with its conclusions set forth throughout this SE, the NRC staff finds the proposed change acceptable.

Limiting Safety System Settings, Protective Instrumentation - TS 2.3

The licensee's proposed change will modify the design full power reactor coolant system Tavg at rated power to $<579.2\ ^{\circ}$F from the current value of $<579.7\ ^{\circ}$F in TSs 2.3.1.B(4) and 2.3.1.B(5). This value is used in the overtemperature and overpower delta-T algorithm. This modified Tavg value is consistent with the new Tavg developed at 1.4% uprated power level of 3114.4 MWt. Therefore, the NRC staff finds the proposed change acceptable.

Reactor Coolant System Pressure, Temperature and Flow Rate - TS 3.1.G

The licensee's proposed change would modify the design full power reactor coolant system Tavg at rated power to $< 586.7\ ^{\circ}$F from the current value of $<587.2\ ^{\circ}$F in TS 3.1.G. This value is specified relative to DNB parameters for 4-loop steady-state operation at a power level greater than 98% of rated full power. Instrument uncertainties are not accounted for in determining the DNB parameter limits on temperature and pressure. Since the proposed change is in the conservative direction, the NRC staff finds the proposed change acceptable.

Operable MSSV Versus Applicable Neutron Flux Trip Setpoints, TS, Table 3.4-1

The licensee's proposed change would decrease the allowable values for the neutron flux trip setpoint during operation with one or more main steam safety valves (MSSVs) inoperable. The licensee indicated that these changes are appropriate due to the slight increase of rated thermal power level due to the proposed 1.4% power uprate. The changes to TS Table 3.4-1 will provide more conservative trip setpoints at different MSSV inoperable conditions. The NRC staff finds the licensee's proposal acceptable because the changes are in the conservative direction from the current TS at IP2.

List of Reference Documents applicable for Core Operating Limits - TS 6.9.1.9

The licensee's proposed change would modify TS 6.9.1.9 to add Caldon Engineering Report-80P and Caldon Engineering Report-160P which are a part of the documentation relative to developing core operating limits at IP2 under uprated power conditions. This change will establish a basis for performing reload analyses at the uprated power, and the NRC staff finds it acceptable.

TS Bases Associated with the Proposed TS Changes

The licensee proposed a change to the Bases for TS 3.4, “Steam and Power Conversion System,” which would revise it to reflect the updated values of steam flow parameters. This proposed change is consistent with the TS changes discussed above. Therefore, the NRC staff does not object to the proposed change to the TS Bases.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (68 FR 801). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

7.0 REFERENCES

1. ENO Letter, Fred R. Dacimo to NRC, “Proposed Changes to Technical Specifications: Measurement Uncertainty Recapture Power Uprate Increase of Licensed Thermal Power (1.4%),” dated December 12, 2002.

2. ENO Letter, Fred R. Dacimo to NRC, “Reply to Request for Additional Information Regarding Proposed Licensing Amendment for 1.4% Measurement Uncertainity Recapture Power Uprate,” dated April 3, 2003.

3. ENO Letter, Fred R. Dacimo to NRC, “Reply to Request for Additional Information Regarding Proposed Licensing Amendment for 1.4% Measurement Uncertainity Recapture Power Uprate,” dated May 2, 2003.

4. Westinghouse Report WCAP-10263, “A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant,” dated January 1993.

5. Caldon, Inc. Topical Report ER-80P, “Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System,” NRC approved March 8, 1999.

6. Caldon, Inc. ER-157P Topical Report, Supplement to Topical Report ER-80P, “Basis for a Power Uprate with the LEFM Check or LEFM CheckPlus System,” NRC approved December 20, 2001.

7. Caldon, Inc. ER-160P Topical Report, Supplement to Topical Report ER-80P, “ Basis for a Power Uprate with the LEFM Check System,” NRC approved January 19, 2001, as part of the Watts Bar SE approval.

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