ATTACHMENT I TO NL-02-155

ANALYSIS OF PROPOSED

TECHNICAL SPECIFICATION CHANGE REGARDING

INCREASE OF LICENSED THERMAL POWER, 1.4%

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2
DOCKET NO. 50-247

1.0 DESCRIPTION

This letter is a request to amend Operating License DPR-26, Docket No. 50-247 for Indian Point Nuclear Generating Unit No. 2.

The proposed changes to the Indian Point 2 (IP2) Technical Specifications (TS) are based upon the application of a 1.4% Measurement Uncertainty Recapture (MUR) Power Uprate analysis in support of a new Caldon Leading Edge Flow Meter (LEFM) measurement system. This new Caldon measurement system will improve accuracy in feed water flow and temperature measurement. This increase in accuracy will allow IP2 to operate at a 1.4% higher reactor thermal power level, improving core power output from 3071.4. MWt to 3114.4 MWt. This 1.4% core power uprate is effectively achieved by recapturing excess uncertainty currently included in the power uncertainty allowance originally required for Emergency Core Cooling System (ECCS) evaluation models performed in accordance with the requirements set forth in 10 CFR 50, Appendix K. Improvement in core power level measurement accuracy is possible through the reduction in feed water flow measurement uncertainty used in the reactor power calorimetric calculation. The feed water flow measurement uncertainty is reduced through the installation and use of a Caldon, Inc. LEFM Check 2000FC Cabinet.

Analysis work in the support of this 1.4% MUR power uprate amendment request has been performed consistent with the NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002.

2.0 PROPOSED CHANGE

a. Paragraph 2.C.1 of the Facility Operating License will be changed to allow reactor core power levels not in excess of **3114.4** MWt. The current value is 3071.4 MWt.

b. Technical Specification 1.1 will be revised to define 'rated power' as a steady state reactor thermal power of **3114.4** MWt. The current value is 3071.4 MWt.

c. Technical Specification Figure 2.1-1, "Reactor Core Safety Limit – Four Loops in Operation," will be replaced with a new figure updated to reflect the new maximum power level.

d. Technical Specification 2.3.1.B (4) will be revised to state the design full power Tavg at rated power as $\leq$ **579.2** ^{O}F. This value is used in the Overtemperature Delta-T algorithm. The current value is 579.7 ^{O}F

e. Technical Specification 2.3.1.B (5) will be revised to state the design full power Tavg at rated power as $\leq$ **579.2** ^{O}F. This value is used in the Overpower Delta-T algorithm. The current value is 579.7 ^{O}F.

f. Technical Specification 3.1.G.a will be revised to state the Reactor Coolant System Tavg related to DNB parameters for four loop steady state operation at power levels greater than 98% of rated full power as $\leq$ **586.7** ^{O}F. The current value is 587.2 ^{O}F.

g. Technical Specification Table 3.4-1 will be revised to state the new power range neutron (PRN) flux high setpoints corresponding to inoperable secondary safety valves:

   With 1 safety valve inoperable, the PRN setpoint is reduced from 64 to **59** (% RTP)
   With 2 safety valves inoperable, the PRN setpoint is reduced from 44 to **40**
   With 3 safety valves inoperable, the PRN setpoint is reduced from 24 to **21**

h. Technical Specification 6.9.1.9, which lists the reference documents applicable for determining core operating limits, will be revised by adding Caldon Engineering Report-80P and Caldon Engineering Report-160P.

i. The Bases for Technical Specification section 3.4, "Steam and Power Conversion System" will be revised to reflect the updated steam flow parameters in the discussion of the main steam safety valves.

3.0 BACKGROUND

Indian Point Nuclear Generating Unit No. 2 is presently licensed for a full core power rating of 3071.4 MWt. Through the use of more accurate system to measure feed water flow, Caldon LEFM Check System, improved accuracy of core thermal power is obtained by way of a more precise determination of plant secondary calorimetric power. Approval is requested to increase IP2 core thermal power by 1.4% to 3114.4 MWt by use of a Caldon LEFM Check System for improved accuracy of the feed water system variables that are used in the secondary plant calorimetric calculation. ENO evaluated the impact of this 1.4% core power uprate on plant systems, components, and safety analyses. Results of this evaluation are summarized in the sections that follow and primarily in Attachment III of this license amendment submittal.

4.0 TECHNICAL ANALYSIS

ENO has evaluated the impact of the proposed new maximum core thermal power on the nuclear steam supply system and the balance-of-plant . The engineering report provided in Attachment III summarizes the analyses and evaluations performed for this project. The evaluation is consistent with the methodology described in WCAP-10263, "A Review Plan for Uprating the Licensed Power of a PWR Power Plant" (Reference 2). The supporting analyses and evaluations are also consistent with NRC Regulatory Issue Summary (RIS) 2002-03 (Reference 1). The site-specific installation of the improved feedwater flow measuring instrumentation (Caldon LEFM Check system) follows the guidelines of Caldon Topical Report ER-80P (Reference 3)

The analyses and evaluations demonstrate that the proposed 1.4% increase in core thermal power can be accommodated by existing plant structures, systems, and components. Applicable limits and acceptance criteria for normal operating and postulated off-normal transient conditions will continue to be met.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Entergy Nuclear Operations, Inc. has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

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

Response: No.

The proposed 1.4% increase in maximum core thermal power is based on the use of instrumentation that supports a reduction in the measurement uncertainty value assumed in certain safety analyses. The affected analyses now use an uncertainty value of 2% which was required by 10 CFR 50 Appendix K at the time that the plant was originally licensed. At that time, measurement of feedwater flowrate in the plant secondary side used differential pressure-type flow venturis. The plant secondary side thermal calorimetric is used to determine reactor thermal power. A June 2000 revision to 10 CFR 50 Appendix K permitted the use of lower uncertainty values in the affected analyses, if the reduced value can be justified. Entergy Nuclear Operations (ENO) has implemented the use of Caldon, Inc. Leading Edge Flowmeter (LEFM) technology to measure feedwater flowrate. The LEFM measures fluid velocity by measuring the transit time of ultrasonic pulses introduced into the fluid stream. The LEFM Check System implemented at Indian Point 2 has a demonstrated measurement accuracy of 0.6%. Based on this measurement accuracy, the licensed thermal power can be increased 1.4% by reducing the assumed uncertainty used in safety analyses with respect to core thermal power from 2.0% to 0.6%. This results in a net increase in licensed reactor core thermal power; from 3071.4 MWt to 3114.4 MWt. The LEFM and the flow venturi instrumentation are used to collect data and there is no automatic initiation function performed by this instrumentation. Use of the LEFM instrumentation is therefore not an accident initiator and does not increase the probability of occurrence of an existing analyzed accident. Also, the LEFM instrumentation and the venturi instrumentation do not mitigate accidents so that the consequences of previously analyzed accidents are not increased.

Analyses and evaluations associated with the proposed change to core thermal power have demonstrated that applicable acceptance criteria for plant systems, components, and analyses (including the Final Safety Analysis Report Chapter 14 safety analyses) will continue to be met for the proposed 1.4% increase in licensed core thermal power for Indian Point 2. The subject increase in core thermal power will not result in conditions that could adversely affect the integrity (material, design, and construction standards) or the operational performance of any potentially affected system, component or analysis. Therefore, the probability of an accident previously evaluated is not affected by this change. The subject increase in core thermal power will not adversely affect the ability of any safety-related system to meet its intended safety function. Further, the radiological dose evaluations in support of this power uprate effort show that the current FSAR Chapter 14 radiological analyses are unaffected, and that the current dose analyses of record bound plant operation with the subject increase in licensed core thermal power level.

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

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

Response: No.

The proposed license amendment increases the maximum allowed core thermal power through the use of feedwater flow instrumentation that supports a reduction in the measurement uncertainty assumed in certain safety analyses. The LEFM Check System instrumentation has greater measurement accuracy than the differential pressure-type flow venturi instrumentation that was originally used so that the measurement uncertainty assumed in certain analyses can be correspondingly reduced. Both the venturi and LEFM flow instrumentation provide data that is used by plant operators to monitor the thermal output of the plant. The instrumentation does not perform an automatic actuation function and there are no output signals to plant safety systems or control systems. Therefore, instrumentation malfunction or failure does not introduce new accident scenarios or equipment failure mechanisms. Operation, maintenance, or failure of either instrumentation system does not have an adverse effect on safety-related systems or any structures, systems, and components required for transient or accident mitigation.

Operating the plant at a new maximum core thermal power of 3114.4MWt, which is 1.4% greater than the current maximum of 3071.4 MWt, is bounded by existing or updated analyses which demonstrate that established limits and acceptance criteria continue to be met. Operating at the new power level does not create new or different accident initiators and existing credible malfunctions are bounded by existing or updated analyses or evaluations.

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

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The evaluations and analyses associated with the proposed increase in maximum core thermal power demonstrate that applicable acceptance criteria will continue to be met. The existing licensed maximum core thermal power level incorporates a 2% measurement uncertainty for the analysis of loss-of-coolant-accidents as originally required by Appendix K of 10 CFR 50. The regulations have subsequently been revised to allow the option of justifying smaller measurement uncertainties by using more accurate instrumentation to calculate reactor thermal power. Certain analyses that already assume a bounding core power level because of the 2% measurement uncertainty are not changed as a result of the proposed increase in core thermal power. Use of the LEFM instrumentation with improved measurement accuracy supports the use of a smaller measurement uncertainty assumption in the safety analyses. Other analyses were updated or evaluations were performed to demonstrate that nuclear steam supply and balance-of-plant systems and components will continue to perform, under normal and credible transient conditions, within established limits.

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

Based on the above, Entergy Nuclear Operations, Inc. concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92 (c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements / Criteria

The proposed change has been evaluated in accordance with NRC guidance provided in Regulatory Issue Summary 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002. The analyses and evaluations completed to support the proposed increased in core thermal power demonstrate that acceptance criteria including those established by regulatory requirements continue to be met.

The proposed increase in core thermal power is being accomplished based on the revised requirements (June 2000) of 10 CFR 50, Appendix K, which allows for a reduction of measurement uncertainty that was previously required in the analysis of loss-of-coolant-accidents. The reduced measurement uncertainty analysis assumption is supported by the implementation of feedwater flow instrumentation with improved accuracy. The analysis criteria and reporting requirements of 10 CFR 50.46 will continue to be satisfied at the new maximum power level.

The affect of the new maximum power level on structures, systems, and components of the nuclear steam supply system and the balance-of-plant was evaluated to assure that applicable regulatory requirements and criteria are met. A description of the analyses and evaluations performed is provided in the measurement uncertainty recapture report provided with this application for amendment.

ENO has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than those changes proposed in the IP2 TS. Additionally, this change does not affect conformance with any General Design Criteria differently than described in the FSAR.

5.3 Environmental Considerations

The proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

6.0 PRECEDENCE

The NRC has previously approved similar applications for amendment which use improved instrumentation to support a reduction in the assumed core power measurement uncertainty and a corresponding increase in maximum licensed core thermal power. The concept of measurement uncertainty recapture was a result of a change to Appendix K of 10 CFR 50. The final rule was issued in June 2000. The specific instrumentation to be implemented at Indian Point 2 is the Caldon Leading Edge Flowmeter (LEFM) Check System. The design and performance of this system has previously been approved by the NRC as documented in topical reports; references 3, 4, and 5. The evaluations performed for Indian Point 2 demonstrate that the site-specific parameters are appropriately bounded by the topical reports. Prior NRC approvals for the measurement uncertainty recapture power uprate include licensees that have implemented the Caldon instrumentation systems (LEFM Check or LEFM CheckPlus), including Waterford approved in March 2002, Sequoyah approved in April 2002, and H. B. Robinson approved in November 2002.

7.0 REFERENCES

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

2. Westinghouse WCAP-10263, " A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," dated January 1983.

3. Caldon, Inc. Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level using the LEFM Check System," approved by NRC SER dated March 8, 1999.

4. Caldon, Inc. Topical Report ER-157P, Supplement to Topical Report ER-80P, "Basis for a Power Uprate with the LEFM Check or LEFM CheckPlus System," approved by NRC SER dated December 20, 2001.

5. Caldon, Inc. Topical Report ER-160P, Supplement to Topical Report ER 80-P, "Basis for a Power Uprate with the LEFM Check System," approved by NRC SER on January 19, 2001 as part of the Watts Bar license amendment MUR power uprate approval.

ATTACHMENT II TO NL-02-155

MARKUP OF TECHNICAL SPECIFICATION AND BASES

FOR PROPOSED CHANGES REGARDING

INCREASE OF LICENSED THERMAL POWER, 1.4%

Facility Operating License, page 3
Technical Specification Pages:

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- Figure 2.1-1
- Page 2.3-1
- Page 2.3-3
- Page 3.1.G-1
- Table 3.4-1
- Page 6-10
- Page 3.4-3 (Bases change)

ENTERGY NUCLEAR OPERATIONS, INC.
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instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) ENO pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; Amdt. 42 10-17-78

(5) ENO pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70 32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) <u>Maximum Power Level</u>

ENO is authorized to operate the facility an steady state reactor core power levels not in excess of ~~3071.4~~3114.4 megawatts thermal. Amdt. 148 3-7-90

(2) <u>Technical Specifications</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 234, are hereby incorporated in the license. ENO shall operate the facility in accordance with the Technical Specifications.

D. (1) Deleted per Amdt. 82, 12-11-82.

(2) Secondary Water Chemistry Monitoring Amdt. 60 1-28-80

ENO shall implement a secondary water chemistry monitoring program to inhibit steam generator tube degradation. The program shall include:

(a) Identification of a sampling schedule for the critical parameters and control points for these parameters;

1.0 DEFINITIONS

The following terms are defined for uniform interpretation of the specifications.

1.1 a. RATED POWER

A steady state reactor thermal power of ~~3071.4~~ 3114.4 MWT.

b. THERMAL POWER

The total core heat transfer rate from the fuel to the coolant.

1.2 REACTOR OPERATING CONDITIONS

1.2.1 Cold Shutdown Condition

When the reactor is subcritical by at least 1% $\Delta k/k$ and Tavg is $\leq$ 200°F*.

1.2.2 Hot Shutdown Condition

When the reactor is subcritical, by an amount greater than or equal to the margin as specified in Technical Specification 3.10 and T_{avg} is > 200°F* and $\leq$ 555°F.

1.2.3 Reactor Critical

When the neutron chain reaction is self-sustaining and $k_{eff} = 1.0$.

1.2.4 Power Operation Condition

When the reactor is critical and the neutron flux power range instrumentation indicates greater than 2% of rated power.

* For the one time, fuel out, chemical decontamination program only, this value will be 250°F.

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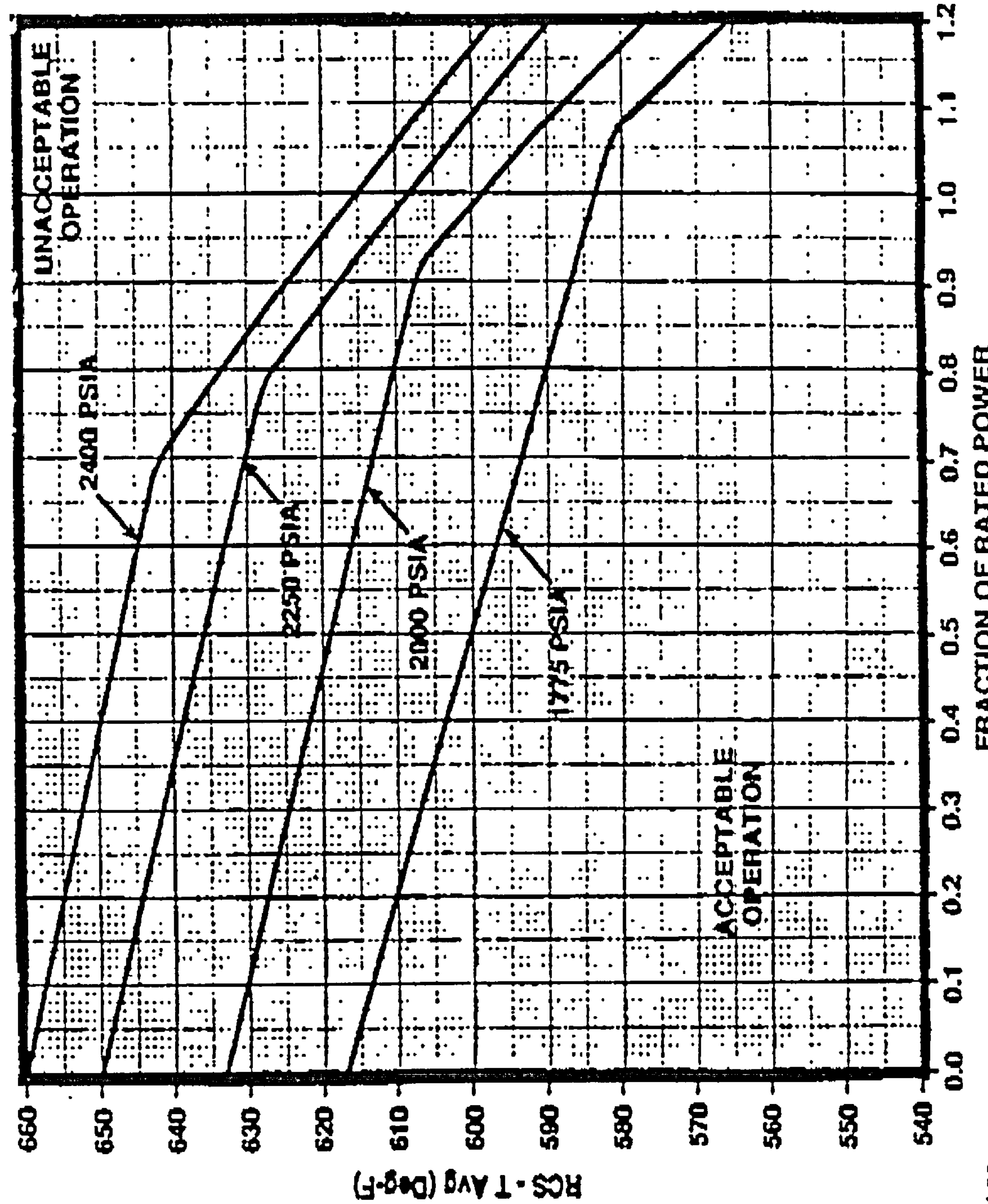
Figure 2.1-1
Reactor Core Safety Limit – Four Loops in Operation



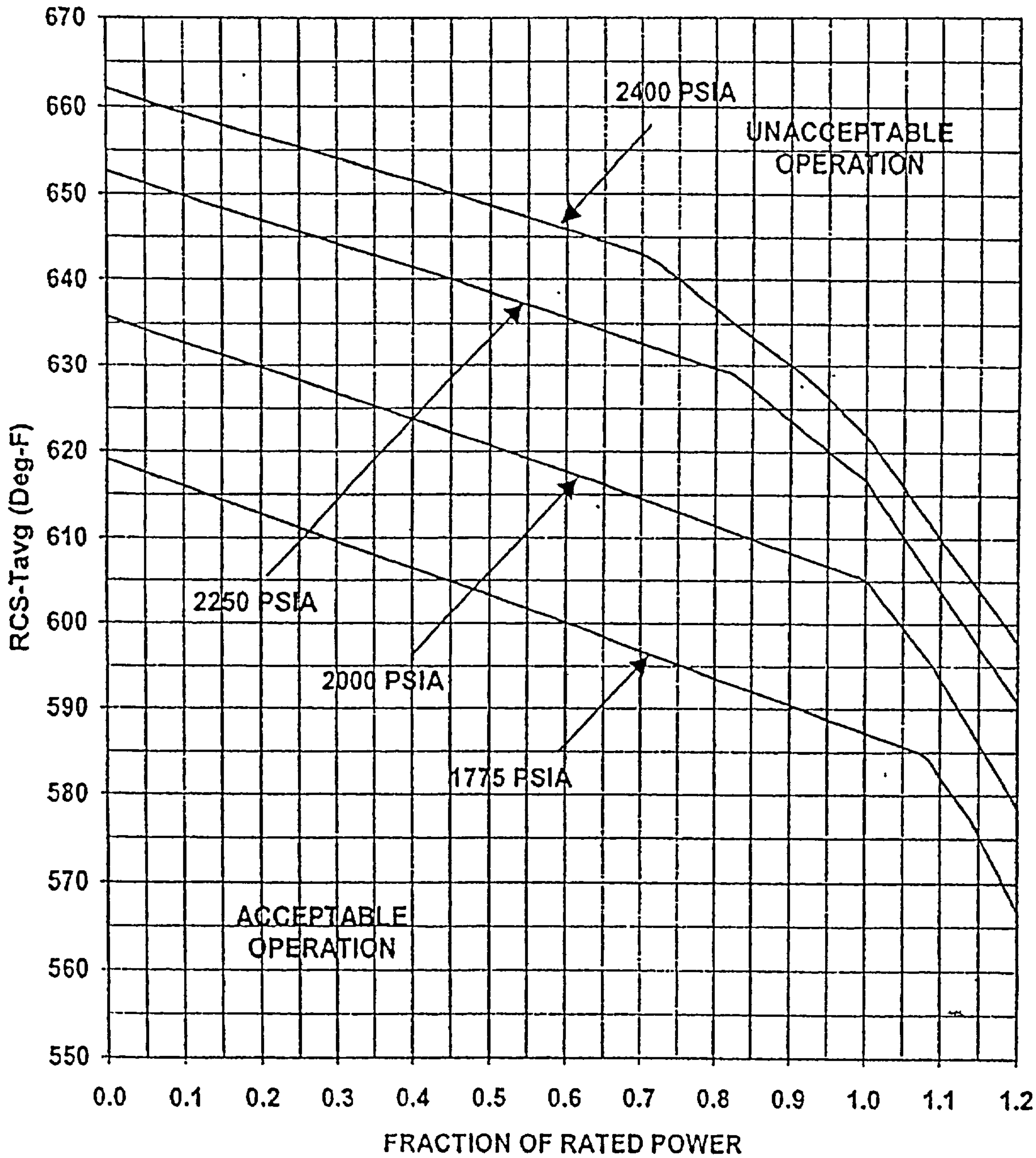


Figure 2.1-1

Reactor Core Safety Limit – Four Loops in Operation

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, flow, and pressurizer level.

Objective

To provide for automatic protective action such that the principal process variables do not exceed a safety limit.

Specifications

1. Protective instrumentation for reactor trip settings shall be as follows:

   A. Startup protection

      (1) High flux, power range (low setpoint): ≤ 25% of rated power.

   B. Core limit protection

      (1) High flux, power range (high setpoint): ≤ 109% of rated power.

      (2) High pressurizer pressure: ≤ 2363 psig.

      (3) Low pressurizer pressure: ≥ 1928 psig.

      (4) Overtemperature ΔT:

      $$\Delta T \leq \Delta T_o \left[K_1 - K_2 (T - T') + K_3 (P - P') - f(\Delta I)\right]$$

      where:

      ΔT = Measured ΔT by hot and cold leg RTDs, °F

      ΔT_o ≤ Indicated ΔT at rated power, °F

      T = Average temperature, °F

      T' = Design full power T_{avg} at rated power, ≤ ~~579.7~~579.2°F

P = Pressurizer pressure, psig

P' = 2235 psig

$K_1 \leq$ 1.22

K_2 = 0.022

K_3 = 0.00095

and $f(\Delta I)$ is a function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(i) For $q_t - q_b$ between -36% and +7%, $f(\Delta I) = 0$, where q_t and q_b are percent rated power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total power in percent of rated power;

(ii) For each percent that the magnitude of $q_t - q_b$ exceeds -36%, the ΔT trip setpoint shall be automatically reduced by 2.14% of its value at rated power; and

(iii) For each percent that the magnitude of $q_t - q_b$ exceeds +7%, the ΔT trip setpoint shall be automatically reduced by 2.15% of its value at rated power.

(5) Overpower ΔT:

$$\Delta T \leq \Delta T_o \left[K_4 - K_5 \frac{dT}{dt} - K_6 (T - T'')\right]$$

where:

ΔT = Measured ΔT by hot and cold leg RTDs, °F

$\Delta T_o \leq$ Indicated ΔT at rated power, °F

T = Average temperature, °F

T'' = Indicated full power T_{avg} at rated power ≤ ~~579.7~~579.2°F

K_4 ≤ 1.074

K_5 = Zero for decreasing average temperature

K_5 ≥ 0.188, for increasing average temperature (sec/°F)

K_6 ≥ 0.0015 for $T \geq T''$; $K_6 = 0$ for $T < T''$

$\frac{dT}{dt}$ = Rate of change of T_{avg}

(6) Low reactor coolant loop flow:

(a) ≥ 92% of normal indicated loop flow.

(b) Low reactor coolant pump frequency: ≥ 57.5 cps.

(7) Undervoltage: ≥ 70% of normal voltage.

C. Other reactor trips

(1) High pressurizer water level: ≤ 90% of span.

(2) Low-low steam generator water level: ≥ 7% of narrow range instrument span.

2. Protective instrumentation settings for reactor trip interlocks shall satisfy the following conditions:

A. The reactor trips on low pressurizer pressure, high pressurizer level, and low reactor coolant flow for two or more loops shall be unblocked when:

(1) Power range nuclear flux ≥ 10% of rated power, or

(2) Turbine first stage pressure ≥ 10% of equivalent full load.

B. The single loop loss of flow reactor trip may be bypassed when the power range nuclear instrumentation indicates ≤ 60% of rated power.

G. REACTOR COOLANT SYSTEM PRESSURE, TEMPERATURE, AND FLOW RATE

Specifications

The following DNB related parameters pertain to four loop steady-state operation at power levels greater than 98% of rated full power:

a. Reactor Coolant System $T_{avg} \leq$ ~~587.2~~586.7°F

b. Pressurizer Pressure $\geq$ 2190 psia

c. Reactor Coolant System Total Flow Rate $\geq$ 331,840 gpm

Item (b), pressurizer pressure, is not applicable during either a thermal power change in excess of 5% of rated thermal power per minute, or a thermal power step change in excess of 10% of rated thermal power.

Under the applicable operating conditions, should reactor coolant temperature, T_{avg}, or pressurizer pressure exceed the values given in items (a) and (b), the parameter shall be restored to its applicable range within 2 hours.

Basis

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions that would result in a DNBR of less than the safety limit DNBRs.

The limits on reactor coolant system temperature, pressure and loop coolant flow represent those used in the accident analyses and are specified to assure that the values assumed in the accident analyses are not exceeded during steady-state four loop operation. Indicator uncertainties have not been accounted for in determining the DNB parameter limits on temperature and pressure.

TABLE 3.4-1

Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steam Line Safety Valves During 4-Loop Operation

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of Rated Thermal Power)
1	~~64~~59
2	~~44~~40
3	~~24~~21

If these requirements cannot be met, then:

1. maintain the plant in a safe, stable mode which minimizes the potential for a reactor trip, and

2. continue efforts to restore water supply to the auxiliary feedwater system, and

3. notify the NRC within 24 hours regarding the planned corrective action.

Basis

Reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condensers. Thereafter, core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to feed the steam generators is provided by operation of the turbine cycle feedwater system.

The operability of the twenty main steam line code safety valves ensure that the secondary system pressure will be limited to within 110% of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% Rated Thermal Power coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The total relieving capacity of the twenty main steam safety valves is 15,108,000 lbs/hr which is ~~114~~ 111.2 percent of the total secondary steam flow of ~~13,310,000~~13,580,000 lbs/hr at 100% NSSS Power (~~3083.4~~3126.4 Mwt). Startup and/or power operation is allowable with main steam safety valves inoperable within the limitations of Table 3.4-1 on the basis of the reduction in secondary system steam flow and thermal power required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are based on the heat removal capacity of the remaining operable steam line safety valves. The maximum thermal power corresponding to the heat removal capacity of the remaining operable steam line safety valves is determined via a conservative heat balance calculation as described in the attachment to Ref. 2 with an appropriate allowance for calorimetric power uncertainty.

e. Control Bank Insertion Limits for Specification 3.10.4.

6.9.1.9 The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Methodology for Specification 3.10.4 - Shutdown Bank Insertion Limit, Control Bank Insertion Limits and 3.10.2 - Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT", September 1974 (W Proprietary). (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

c. T.M. Anderson to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 - Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package. (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

d. NUREG-0800, Standard Review Plan, US Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981. (Methodology for Specification 3.10.2 - Axial Flux Difference (Constant Axial Offset Control).)

e. WCAP-10266-P-A Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987, (W Proprietary). (Methodology for Specification 3.10.2 Height Dependent Heat Flux Hot Channel Factor.)

f. WCAP-12945-P, Westinghouse "Code Qualification Document for Best Estimate LOCA Analyses", July, 1996

g. Caldon, Inc. Engineering Report-80P, " Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM √™ System," Revision 0, March 1997, and Caldon, Inc. Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM √™ System," Revision 0, May 2000.

6.9.1.10 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling System (ECCS) limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

ATTACHMENT III TO NL-02-155

1.4% MEASURMENT UNCERTAINTY RECAPTURE

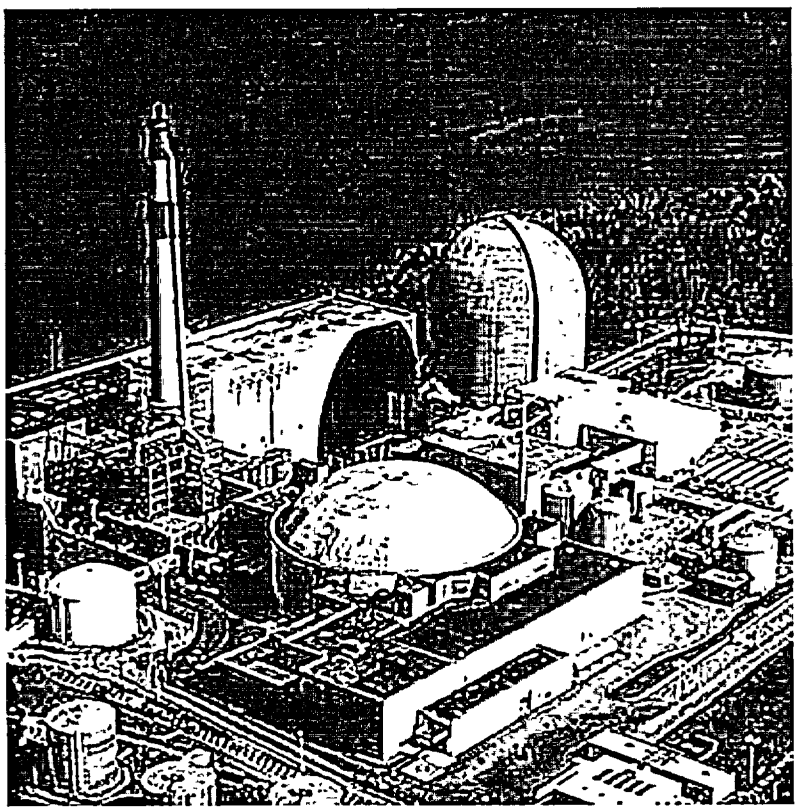
POWER UPRATE APPLICATION REPORT

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2
DOCKET NO. 50-247

November 2002

Entergy Nuclear Operations, Incorporated

Indian Point Nuclear Generating Unit No. 2



1.4-Percent Measurement Uncertainty Recapture

Power Uprate License Amendment Request Package

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LIST OF ACRONYMS

AAC	Alternate Alternating Current
AC	Alternating Current
AEVS	Auxiliary Feedwater Electrical Vent System
AFW	Auxiliary Feedwater
AFWS	Auxiliary Feedwater System
AISC	American Institute of Steel Construction
ALARA	As Low As is Reasonably Achievable
AMSAC	ATWS Mitigation System Actuation Circuitry
ANS	American Nuclear Society
ANSI	American National Standards Institute
AO	Axial Offset
AOO	Anticipated Operational Occurrences
ARL	Alden Research Laboratory
ARV	Atmospheric Relief Valves
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient Without Scram
AVB	Anti-Vibration Bar
B&PV	Boiler and Pressure Vessel Code
BEF	Best-Estimate Flow
BOL	Beginning of Life
BOP	Balance of Plant
C&FS	Condensate and Feedwater System
CCR	Central Control Room
CCW	Component Cooling Water
CCWS	Component Cooling Water System
CF	Chemistry Factor
CFC	Containment Fan Cooler
CFR	Code of Federal Regulations
COMS	Cold Overpressure Mitigation System
CRDM	Control Rod Drive Mechanism
CST	Condensate Storage Tank
CUF	Cumulative Usage Factor
CVCS	Chemical and Volume Control System
DC	Direct Current
DBA	Design-Basis Accident
DBE	Design-Basis Event
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DOR	Division of Operating Reactors

LIST OF ACRONYMS (CONT.)

ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EDS	Electrical Distribution System
EFPY	Effective Full-Power Year
EOL	End of Life, or License
EPRI	Electrical Power Research Institute
EQ	Environmental Qualification
ERG	Emergency Response Guideline
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Feature Actuation System
FA	Forced-Air Cooled
FAC	Flow Accelerated Corrosion
FCV	Feedwater Control Valves
$F_{\Delta H}$	Hot Channel Enthalpy Rise
FDB	Flow Distribution Baffle
FES	Final Environmental Statement
FF	Fluence Factor
FOA	Forced-Oil-Air Cooled
FON	Fraction of Normal
FR	Federal Register
FRV	Feedwater Regulating Valve
HELB	High-Energy Line Break
HFP	Hot Full Power
HLSO	Hot-Leg Switch Over
HVAC	Heating, Ventilation, and Air Conditioning
HZP	Hot Zero Power
I&C	Instrumentation and Control
IEEE	Institute of Electrical and Electronics Engineers
IFM	Intermediate Flow Mixer
IP2	Indian Point Unit 2
ISA	Instrument Society of America
LAR	License Amendment Request
LBB	Leak Before Break
LBLOCA	Large-Break Loss-of-Coolant Accident
LEFM	Leading Edge Flow Meter
LOAC	Loss of All AC
LOCA	Loss-of-Coolant Accident
LOL	Loss of Load

LIST OF ACRONYMS (CONT.)

LONF	Loss of Normal Feedwater
LOOP	Loss-of-Offsite Power
LTCC	Long-Term Core Cooling
LTOPS	Low-Temperature Overpressure Protection System
MA	Mill Annealed
MCO	Moisture Carryover
MFP	Main Feedwater Pump
MOV	Motor-Operated Valve
MRIL	Maximum Reliable Indicated Level
MSIV	Main Steam Isolation Valve
MSLB	Main Steamline Break
MSS	Main Steam System
MSSV	Main Steam Safety Valve
MT	Main Transformer
MTC	Moderator Temperature Coefficient
NIS	Nuclear Instrumentation System
NPSH	Net Positive Suction Head
NPSHa	Net Positive Suction Head Available
NRC	Nuclear Regulatory Commission
NRS	Narrow Range Span
NSAL	Nuclear Service Advisory Letter
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
ODSCC	Outside-Diameter Stress Corrosion Cracking
OFA	Optimized Fuel Assembly
OPΔT	Overpower Delta Temperature
OTΔT	Overtemperature Delta Temperature
P&I	Proportional and Integral
PCWG	Performance Capability Working Group
PICS	Plant Integrated Computer System
PORV	Power-Operated Relief Valve
PMA	Process Measurement Accuracy
PRA	Probabilistic Risk Assessment
PRT	Pressurizer Relief Tank
PSS	Primary Sampling System
P-T	Pressure-Temperature
PTC	Pressure Test Code
PTLR	Pressure Temperature Limit Report
PTS	Pressurized Thermal Shock

LIST OF ACRONYMS (CONT.)

PWR	Pressurized Water Reactor
PWSCC	Primary Water System Stress Corrosion Cracking
QA	Quality Assurance
RCCA	Rod Cluster Control Assembly
RCL	Reactor Coolant Loop
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RG	Regulatory Guide
RHR	Residual Heat Removal
RHRS	Residual Heat Removal System
RIS	Regulatory Issue Summary
RPS	Reactor Protection System
RSE	Reload Safety Evaluation
RTDP	Revised Thermal Design Procedure
RT_{NDT}	Reference Temperature for Nil Ductility Transition
$RT_{NDT(u)}$	Reference Temperature for Nil Ductility Transition (Unirradiated)
RTP	Rated Thermal Power
RT_{PTS}	Reference Temperature, Pressurized Thermal Shock
RTS	Reactor Trip System
RTSR	Reload Transition Safety Report
RWFS	Rod Withdrawal From Subcritical
RWST	Refueling Water Storage Tank
SAL	Safety Analysis Limit
SAT	Station Auxiliary Transformer
SBLOCA	Small-Break Loss-Of-Coolant Accident
SBO	Station Blackout
S/C	Surveillance Capsule
SDS	Steam Dump System
SER	Safety Evaluation Report
SFP	Spent Fuel Pool
SFPCS	Spent Fuel Pool Cooling System
SG	Steam Generator
SGBS	Steam Generator Blowdown System
SGPER	Steam Generator PERformance
SGTP	Steam Generator Tube Plugging
SGTR	Steam Generator Tube Rupture
SPDES	State Pollutant Discharge Elimination System
SSE	Safe Shutdown Earthquake
SST	Station Service Transformer
STDP	Standard Thermal Design Procedure

LIST OF ACRONYMS (CONT.)

TDF	Thermal Design Flow
TSP	Tube Support Plate
TT	Thermally Treated
UAT	Unit Auxiliary Transformer
UFSAR	Updated Final Safety Analysis Report
USAS	United States of America Standard
USE	Upper Shelf Energy
V+	VANTAGE+
V&V	Verification and Validation
VAR	Volt-Ampere Reactive
WOG	Westinghouse Owners Group
X/X_{DNB}	Ratio of the Local Quality to the Estimated Quality at DNB Transition

1 INTRODUCTION

1.1 BACKGROUND

The purpose of this report is to support the United States Nuclear Regulatory Commission (NRC) review and approval of the Indian Point Unit 2 (IP2) 1.4-Percent Measurement Uncertainty Recapture Power Uprate License Amendment Request (LAR). Indian Point Unit 2 is presently licensed for a core power rating of 3071.4 MWt (see Section 1.4). The 1.4-percent power uprate, which is enabled through the use of more accurate feedwater flow measurement techniques, will increase the IP2 licensed core thermal power to 3114.4 MWt.

The June 1, 2000 NRC rulemaking regarding the Code of Federal Regulations (CFR) 10 CFR 50, Appendix K (Federal Register (FR) 65 FR 34913, June 1, 2000) allows licensees to use a power uncertainty of less than 2 percent in loss-of-coolant accident (LOCA) analyses. This rulemaking provides licensees with the option of either maintaining the 2-percent power allowance between the licensed core power level and the core power level assumed in the plant licensing basis LOCA analyses, or applying a reduced allowance that accounts for more accurate feedwater flow measurement techniques.

The 1.4-percent core power uprate is effectively achieved by recapturing excess uncertainty currently included in the power uncertainty allowance originally required for Emergency Core Cooling System (ECCS) evaluation models performed in accordance with the requirements set forth in 10 CFR 50, Appendix K. Improvement in core power measurement accuracy is possible through the reduction of the feedwater flow measurement uncertainty used in the power calorimetric calculation. The feedwater flow measurement uncertainty is reduced through the use of improved measurement instrumentation. Since most of the current IP2 licensing bases analyses already include a 2-percent core power allowance, a demonstrated core power uncertainty of 0.6 percent effectively enables a 1.4-percent increase in licensed core thermal power - with limited effect on most plant analyses and equipment.

This report summarizes the various evaluations and analyses of the potential effects of the 1.4-percent core power uprate on plant systems, components, and analyses.

1.2 LICENSING APPROACH

All work supporting the IP2 1.4-percent power uprate, and the preparation of this report, has been performed consistent with the NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002. Affected and unaffected plant systems, components, and analyses have been clearly distinguished throughout the report according to the RIS 2002-03 guidance. Affected systems, components, and safety analyses are those having current design and licensing bases analyses and calculations that do not bound the potential effects of the 1.4-percent power uprate. Unaffected systems, components, and safety analyses are those having current design and licensing bases analyses and calculations that bound the potential effects of the 1.4-percent power uprate.

Furthermore, Westinghouse has addressed the potential effects of the 1.4-percent power uprate on Nuclear Steam Supply System (NSSS) systems, components, and safety analyses consistent with the Westinghouse methodology established in WCAP-10263, "A Review Plan for Uprating the Licensed

Power of a PWR Power Plant," dated January, 1983. Since submittal to the NRC, the WCAP-10263 methodology has been successfully used as the basis for power uprate projects for over 30 pressurized water reactor (PWR) units.

The methodology in WCAP-10263 establishes the general approach and criteria for uprate projects, including the broad categories that must be addressed, such as NSSS design parameters, design transients, systems, components, accidents, and nuclear fuel, as well as the interfaces between the NSSS and Balance-of-Plant (BOP) systems. The methodology includes the use of well-defined analysis input assumptions/parameter values, use of currently approved analytical techniques, and use of currently applicable licensing criteria and standards. A comprehensive engineering review program consistent with the WCAP-10263 methodology has been performed for IP2 to evaluate the increase in the licensed core power from 3071.4 MWt to 3114.4 MWt.

1.3 EVALUATION APPROACH

Either the licensed core thermal power, or the associated NSSS thermal power, is used as one of the inputs to most plant system, component, and safety analyses in one of the following four ways:

1. A relatively small number of IP2 analyses assume either a nominal core or nominal NSSS power level. These analyses have either been evaluated or revised for the 1.4-percent power uprate. The results of these evaluations and analyses demonstrate that the applicable analysis acceptance criteria will continue to be met at the 1.4-percent power uprate conditions.

2. Some IP2 analyses assume a core power level in excess of the 1.4-percent uprate core power level of 3114.4 MWt. These analyses were previously performed at a higher power level that bounds the current IP2 power level and the 1.4-percent uprate power level. This higher power level is typically the original IP2 design basis core thermal power level of 3216 MWt. For these analyses, some of this existing excess margin was used to accommodate the 1.4-percent uprate.

3. Most IP2 analyses already add a 2-percent uncertainty allowance to the nominal power level to account solely for power measurement uncertainty. These analyses have not been revised for the 1.4-percent uprate power level conditions because the sum of increased core power level (1.4-percent) and the improved power measurement accuracy (uncertainty less than 0.6 percent) is already bounded by the currently analyzed 2-percent uncertainty allowance.

   The power calorimetric uncertainty calculation described in Section 4 demonstrates that, with the Caldon Leading Edge Flow Meter (LEFM) Check System instrumentation installed, the power measurement uncertainty (based on a 95-percent probability at a 95-percent confidence interval) is less than 0.6 percent. Since these analyses only need to account for the 0.6-percent power measurement uncertainty, the existing 2-percent uncertainty allowance can be allocated to account for the 0.6-percent uncertainty in the analyses and enable the 1.4-percent increase in licensed core thermal power. In addition, these analyses also employ other conservative assumptions that are unaffected by the 1.4-percent increase in core thermal power. Therefore, the use of the calculated 95/95 power measurement uncertainty, and retention of other existing conservative assumptions ensure that the margin of safety for these analyses will not be reduced.

4. Some analyses are performed at 0-percent power conditions, or do not model power at all. By definition, these analyses are unaffected by the 1.4-percent increase in core thermal power and have not been revised.

1.4 SUMMARY OF TECHNICAL SPECIFICATION CHANGES

The primary IP2 Technical Specification and Bases changes associated with 1.4-percent core thermal power uprate project are:

- A change to the core power from 3071.4 MWt to 3114.4 MWt on page 3 of the operating license and in the definition of Rated Power on page 1-1 of the Technical Specifications

- A change to Figure 2.1-1, Reactor Core Safety Limit

- A change to T' on page 2.3-1 and T" on page 2.3-3 of the Technical Specifications to reflect the T_{avg} change for the uprate

- A change to the T_{avg} value on page 3.1.G-1 to reflect the revised T_{avg} for the uprate

- A change to page 3.4-3, Steam and Power Conversion System Bases

- A change to Table 3.4-1, Maximum Allowable Power Range Neutron Flux High Setpoint with Inoperable Steam Line Safety Valves- During 4-Loop Operation

- A change to Specification 6.9.1.9, Core Operating Limits Report, to add Caldon Topical Reports ER-80P and ER-160P

1.5 SCOPE SUMMARY AND LICENSE AMENDMENT REPORT STRUCTURE

This LAR package is structured as follows:

- **Section 2** presents the primary and secondary system design performance conditions (parameters) that were developed based on the 1.4-percent power uprate. These design performance conditions form the basis for all of the NSSS analyses and evaluations contained herein.

- **Section 3** addresses the performance of the Caldon LEFM Check System that provides the more accurate feedwater flow measurement.

- **Section 4** discusses the Revised Thermal Design Procedure (RTDP) uncertainties that support the 0.6-percent power calorimetric uncertainty which, in turn, justifies the 1.4-percent power uprate. This section also addresses the potential effects on Reactor Protection System/Engineered Safety Features Actuation System uncertainties and setpoints.

- **Section 5** concludes that current design transients accommodate the revised NSSS design conditions.

- **Sections 6 and 7** present the NSSS systems (e.g., safety injection, residual heat removal, and control systems) and components (e.g., reactor vessel, pressurizer, reactor coolant pumps, steam generator, and NSSS auxiliary equipment) evaluations completed for the revised design conditions.

- **Section 8** provides the results of the accident analyses and evaluations performed for the various analyses area (e.g., steam generator tube rupture, mass and energy release, LOCA and non-LOCA).

- **Section 9** addresses the potential effects of the uprate on the plant electrical system.

- **Section 10** addresses the potential effects of the uprate on the BOP systems.

- **Section 11** summarizes radiological evaluations for normal operation, environmental qualification, and post-LOCA access to vital areas.

- **Section 12** addresses the potential effects of the uprate in the areas of plant programs and operations, and environmental impact.

The analyses and evaluations described herein demonstrate that all applicable acceptance criteria will continue to be met based on operation at the 1.4-percent power uprate conditions at 3114.4 MWt, and that there are No Significant Hazards related to this power uprate according to the regulatory criteria of 10 CFR 50.92.

2 NUCLEAR STEAM SUPPLY SYSTEM PARAMETERS

2.1 INTRODUCTION

The Nuclear Steam Supply System (NSSS) design parameters are the fundamental parameters used as input in all of the NSSS analyses. They provide the primary- and secondary-side system conditions (temperatures, pressures, and flow) that are used as the basis for all of the NSSS analyses and evaluations.

It was necessary to revise these parameters due to the 1.4-percent increase in licensed core power from 3071.4 MWt to 3115 MWt (which conservatively bounds the proposed amendment value of 3114.4 MWt). The new parameters are identified in Table 2-1. These parameters have been incorporated, as required, into the applicable NSSS systems and components evaluations, as well as safety analyses, performed in support of the 1.4-percent power uprate.

2.2 INPUT PARAMETERS AND ASSUMPTIONS

The NSSS design parameters are determined based on conservative inputs, such as a conservatively low thermal design flow (TDF) and bounding steam generator tube plugging (SGTP) levels, which yield primary- and secondary-side conditions that bound plant operation.

The code used to determine the NSSS design parameters was SGPER (Steam Generator PERformance). There is no explicit Nuclear Regulatory Commission (NRC) approval for the code since it is used to facilitate calculations that could be performed by hand. It uses basic thermal-hydraulic calculations, along with first principles of engineering, to generate the temperatures, pressures, and flows shown in Table 2-1. The code and method used to calculate these values have been successfully used to license all previous uprates for Westinghouse plants.

Four cases are provided for Indian Point Unit 2 (IP2) with the following assumptions common to all cases:

- Westinghouse Model 44F steam generators
- An NSSS uprated power level of 3127 MWt (3115 MWt core power + 12 MWt Reactor Coolant System net heat input)
- A nominal feedwater temperature (T_{feed}) of 431.8°F
- 15x15 VANTAGE+ (V+) fuel
- A core bypass flow of 6.5 percent that accounts for intermediate flow mixing grids

Table 2-1 1.4-Percent Uprate NSSS Design Parameters – Indian Point Unit 2				
	Case 1	Case 2	Case 3	Case 4
Thermal Design Parameters				
NSSS Power, %	101.4	101.4	101.4	101.4
MWt	3127	3127	3127	3127
10^6 Btu/hr	10,670	10,670	10,670	10,670
Reactor Power, MWt [1]	3115	3115	3115	3115
10^6 Btu/hr	10,629	10,629	10,629	10,629
Thermal Design Flow, Loop gpm	80,700	80,700	80,700	80,700
Reactor 10^6 lb/hr	126.6	126.6	121.9	121.9
Reactor Coolant Pressure, psia	2250	2250	2250	2250
Core Bypass, %	6.5	6.5	6.5	6.5
Reactor Coolant Temperature, °F				
Core Outlet	587.4	587.4	615.7	615.7
Vessel Outlet	583.0	583.0	611.7	611.7
Core Average	552.9	552.9	583.0	583.0
Vessel Average	549.4	549.4	579.2	579.2
Vessel/Core Inlet	515.8	515.8	546.7	546.7
Steam Generator Outlet	515.5	515.5	546.4	546.4
Steam Generator [2]				
Steam Temperature, °F	490.4	479.6	522.6	512.0
Steam Pressure, psia	624	564	831	758
Steam Flow, 10^6 lb/hr total	13.48	13.47	13.56	13.53
Feedwater Temperature, °F	431.8	431.8	431.8	431.8
Moisture, % max.	0.25	0.25	0.25	0.25
Tube Plugging, %	0	25	0	25
Zero Load Temperature, °F	547	547	547	547
Hydraulic Design Parameters				
Mechanical Design Flow, gpm	97,700			
Minimum Measured Flow, gpm total	330,000			

Note:

1. Conservatively bounds the proposed value of 3114.4 MWt in this License Amendment Request
2. For analyses limited by high steam pressure, conditions corresponding to a maximum steam pressure of 855 psia, steam temperature of 525.9°F, and steam flow of 13.58 x 10^6 lb/hr are assumed. This covers the possibility that the plant could operate with better-than-expected steam generator performance.

2.3 DISCUSSION OF PARAMETER CASES

Table 2-1 provides the NSSS design parameter cases generated and used as the basis for the 1.4-percent uprate. The cases are defined as follows:

- Cases 1 and 2 represent parameters that incorporate a minimum vessel average temperature (T_{avg}) of 549.4°F while maintaining a minimum vessel/core inlet temperature (T_{cold}) of 515.8°F. Case 2 yields the minimum secondary-side steam generator pressure, temperature, and flow. Note that all primary-side temperatures are identical for these two cases.

- Cases 3 and 4 represent parameters that incorporate a maximum T_{avg} of 579.2°F while maintaining a maximum vessel outlet temperature (T_{hot}) of 611.7°F. Case 3 provides the highest secondary-side performance conditions since it is based on 0-percent SGTP and a higher T_{avg}. Note that all primary-side temperatures are identical for these two cases.

- The core/vessel inlet temperature and steam generator outlet temperature are both referred to as "T_{cold}" interchangeably, in subsequent sections of this report, since they are only 0.3°F different (Table 2-1).

The 1.4-percent power uprate results in minor changes to some of the NSSS design parameters. These changes were evaluated by each of the analytical areas discussed in this report.

- Minimum T_{avg} increased by 0.4°F
- Maximum T_{avg} decreased by 0.5°F
- Steam pressure decreased by 9 psia

2.4 CONCLUSIONS

The four cases of NSSS design parameters identified in Table 2-1 were used to evaluate the impact of the 1.4-percent power uprate on IP2.

The appropriate design parameters were used for each NSSS analysis.

3 CALDON LEFM CALCULATION

3.1 CALDON LEFM ULTRASONIC FLOW MEASUREMENT

The Caldon Leading Edge Flow Meter (LEFM) Check System is used to measure feedwater flow. Feedwater flow is an input for determining the plant secondary calorimetric power, which is used in turn to verify the core thermal power output. The Caldon LEFM Check System uses the transit times of ultrasonic pulses traveling upstream and downstream to calculate the fluid velocity along each of four chords of the circular cross-section of the feedwater pipe. Each of the four velocities is then numerically integrated to determine the volumetric flow, which is then combined with pressure and temperature conditions to determine mass flow through the feedwater pipe. This flow measurement method yields highly accurate flow readings and has been approved by the Nuclear Regulatory Commission (NRC) for power uprate applications as documented in Caldon Topical Reports ER-80P and ER-160P (References 3-1 and 3-2). At Indian Point Unit 2 (IP2), there are LEFM Check System flow elements in each of the four feedwater lines, located at approximately 10 to 15 diameters downstream from two non-planar bends.

3.2 USE OF CALDON LEFM TO DETERMINE CALORIMETRIC POWER

The LEFM Check System measurements of feedwater mass flow and temperature are transmitted to the plant computer for real-time calculation of reactor thermal power. The mass flow and temperature outputs are also used to trend delta pressure (ΔP) readings generated by the feedwater nozzles to be used as a backup for the calorimetric power calculation in the event that the LEFM is out of service.

The trend-based benchmarking of backup instrumentation provides justification for operation if the LEFM Check System is out of service as described in Section 3.3.

3.3 CALDON LEFM CHECK SYSTEM OUT OF SERVICE

As described in Topical Report ER-80P, the LEFM Check System contains self-diagnostics that detect all possible system failures and changes in hydraulic velocity profiles that affect the accuracy of ultrasonic flow measurement devices. Alarm thresholds are set to provide notification prior to a condition that may lead to operation outside its design-basis accuracy. The LEFM Check System does not perform any safety function, and is not used to directly control any plant systems. Therefore, LEFM Check System inoperability has no immediate effect on plant operation.

If the LEFM Check System becomes unavailable, plant operation at a core thermal power level of 3114.4 MWt may continue for the allowed outage time. The allowed outage time for operation at the 1.4-percent power uprate level with an LEFM Check System out of service is 7 days, as long as steady-state conditions persist during the 7 days (i.e., no power changes in excess of 10 percent during the period). There are 5 bases for this proposed time period:

- Indian Point Unit 2 Operations personnel operate based on alternate plant instruments, which is benchmarked to the LEFM's last good reading as soon as the LEFM Check System becomes unavailable. This alternate instrumentation has been subject to programmatic, extensive trending relative to LEFM flow and temperature outputs.

- While recognizing that the accuracy of the alternate instruments may degrade over time, it is considered likely that any degradation as a result of nozzle fouling, drift, and the like, would be imperceptible for the 7-day period as long as steady-state conditions persist.

- It is considered prudent to provide IP2 Operations personnel time to become accustomed to operation with the alternate plant instruments prior to requiring a de-rating should the allowed outage time be exceeded.

- Given that most repairs can be made within 24 hours, the 7 days gives plant personnel ample time to trouble shoot, repair, and verify normal operation of the LEFM Check System within its original uncertainty bounds at the same power level as before the failure.

- A 7-day period will be adequate in most cases to affect an LEFM Check System return to service. Therefore, unnecessary de-rate evolutions would be avoided almost entirely.

If the plant experiences a power decrease of greater than 10 percent during the 7-day period, the permitted maximum power level would be reduced upon return to full power, in accordance with the power levels described below, since a plant transient may result in calibration changes of the alternate instruments.

If the 7-day outage period is exceeded, then the plant would operate at a power level consistent with the accuracy of the alternate plant instruments. These alternate plant instruments are feedwater flow venturis and feedwater temperature detectors. The plant implements procedures and guidance as required according to operator actions when the LEFM Check System is unavailable.

The LEFM Check System at IP2 is installed in each of the 4 feedwater lines. Failure of any 1 of the LEFMs will result in calculation of thermal power based on operation of the operable LEFMs and on operation of alternate plant instruments.

3.4 MAINTENANCE AND CALIBRATION

Maintenance of the Caldon LEFM Check System is performed in accordance with the guidelines established in the referenced Topical Report ER-80P and the User's Manual. Proper maintenance is assured through both automatic and manual checks of the system. Manual checks are performed using site-specific procedures developed from Topical Report ER-80P and the User's Manual.

Calibration and maintenance are performed by qualified personnel using site procedures. The site procedures are developed using the Caldon technical manuals. All work is performed in accordance with site work-control procedures.

Routine preventive maintenance procedures, which include physical inspections, power supply checks, backup battery replacements, and internal oscillator frequency verification, are performed by Caldon.

Ultrasonic signal verification and alignment is performed automatically with the LEFM Check System. Signal verification is possible by review of signal quality measurements performed and displayed by the LEFM Check System. Signal verification status is also provided serially to the online calorimetric program.

Indian Point Unit 2 Instrumentation and Control (I&C) personnel are trained per the I&C training Program on the LEFM Check System before work or calibration may be performed. Formal training by Caldon is provided to site personnel.

The LEFM Check System is designed and manufactured in accordance with Caldon's Code of Federal Regulations (CFR) 10 CFR 50 Appendix B Quality Assurance Program and its Verification and Validation (V&V) Program. Caldon's V&V Program fulfills the requirements of American National Standards Institute/Institute of Electrical and Electronics Engineers - American Nuclear Society (ANSI/IEEE-ANS) Std. 7-4.3.2, 1993, "IEEE Standard Criteria for Digital Computers in Safety Systems of Nuclear Power Generating Stations," Annex E, and American Society of Mechanical Engineers (ASME) NQA-2a-1990, "Quality Assurance Requirements for Nuclear Facility Applications." In addition, the program is consistent with guidance for software V&V in the Electric Power Research Institute (EPRI) TR-103291, "Handbook for Verification and Validation of Digital Systems," December 1994. Specific examples of quality measures undertaken in the design, manufacture, and testing of the LEFM Check System are provided in Section 6.4 and Table 6-1 of Topical Report ER-80P.

Corrective actions involving maintenance are performed by qualified personnel. At IP2, the LEFM Check System will be included in the preventive maintenance program. As a plant system, all equipment problems fall under the site work-control process. All conditions that are adverse to quality are documented under the corrective action program. The software falls under IP2's software quality assurance (QA) program currently in place. Procedures are maintained for notification of deficiencies and error reporting.

In addition to the calibration and maintenance of the LEFM Check System (which also supplies feedwater temperature values), all other instrument components that provide fluid condition data for calculation of rated thermal power is controlled, calibrated, and performance monitored to the conditions represented in the overall calorimetric uncertainty evaluation done for the IP2 1.4-percent power uprate.

The IP2 LEFM Check System is under Caldon's V&V Program, and procedures are maintained for user notification of deficiencies that could affect the accuracy and reliability of mass flow and temperature measurements.

3.5 OPERATIONS AND MAINTENANCE HISTORY OF THE INSTALLED CALDON LEFM INSTRUMENTATION

The LEFM Check System was originally installed at IP2 in 1980. The original electronic unit was upgraded with the Caldon LEFM 8300 electronic unit in 1995. The upgrade to the LEFM Check System Electronic Unit, which meets the requirements of the approved Topical Reports ER-80P and ER-160P, was installed in October 2002. Since installation, the LEFM Check System has been used to provide trend-basis data for tracking the alternate plant instruments to be used when the LEFM Check System is out of service.

The Caldon LEFM Check System was installed at IP2 in the Fall of 2002. The installations were performed in accordance with Caldon's installation and commissioning procedures. These procedures were produced in accordance with the descriptions and criteria established by the referenced Topical Report ER-80P.

The Caldon LEFM Check System installed at IP2 is representative of the Caldon LEFM Check System discussed in the Topical Report ER-80P, and is bounded by the requirements set forth in this topical report.

3.6 UNCERTAINTY DETERMINATION METHODOLOGY

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.

With respect to the Caldon LEFM Check System uncertainties, uncertainty calculations have been performed and determined a mass flow accuracy of better than 0.5 percent of rated flow for IP2.

Additionally, the Caldon LEFM Check System uncertainty calculations have been performed to achieve a 95-percent confidence interval, 95-percent probability flow measurement.

Indian Point Unit 2 maintenance procedures and Caldon LEFM Check System operating instructions ensure that the assumptions and requirements of the uncertainty calculation remain valid.

3.7 SITE-SPECIFIC PIPING CONFIGURATION

The plant-specific installation follows the guidelines of Topical Report ER-80P. Two IP2 LEFM Check System flow elements, loops 21 and 22, were calibrated at Alden Research Laboratory (ARL). Loop 23 and 24 calibration coefficients are based upon ARL testing of a population of 7 flow elements with similar inside diameters and dimensions. The LEFM Check System flow elements at IP2 are installed 10 (loop 21), 12 (loop 22), 15 (loop 23), and 13 (loop 24) diameters downstream from 2 non-planar elbows separated by 10.3 diameters. The uncertainty analysis expressly considers the additional uncertainty for these features, and their effects on the LEFM Check System flow measurement. Further, the actual plant velocity profiles at IP2 have been compared to straight pipe profiles at ARL and the effects on the LEFM Check System measurement have been thoroughly addressed in Caldon Topical Report ER-262. These measurements assure that the actual LEFM Check System measurements at IP2 are bounded by the uncertainty analysis and addressed appropriately in Topical Report ER-80P.

3.8 REFERENCES

3-1 Caldon, Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM $\sqrt{}$™ System," Revision 0, March 1997.

3-2 Caldon, Inc. Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM $\sqrt{}$™ System," Revision 0, May 2000.

4 CONTROL AND PROTECTION SYSTEM SETPOINTS AND UNCERTAINTIES

Westinghouse WCAP-13825, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology Consolidated Edison – Indian Point 2," dated February 1994 (Reference 4-1), provides the basis for the Revised Thermal Design Procedure (RTDP) uncertainties that are used in the Indian Point Unit 2 (IP2) Updated Final Safety Analysis Report (UFSAR) Chapter 14 safety analyses. These include T_{avg} (rod) control, pressurizer pressure control, Reactor Coolant System (RCS) flow measurement (calorimetric), and power measurement (calorimetric). The effects of the 1.4-percent power uprate on the power calorimetric uncertainty, as well as the RTDP and Reactor Trip System (RTS) / Engineered Safety Feature Actuation System (ESFAS) uncertainties, are discussed in the following subsections.

4.1 POWER CALORIMETRIC

Typical plant safety analysis evaluations for Condition II non-departure from nucleate boiling (DNB), Condition III, and Condition IV events assume a power calorimetric uncertainty of 2 percent of rated thermal power (RTP). The 1.4-percent power uprate is based on a reduction in the power calorimetric uncertainties, such that the calculated uncertainties, plus the magnitude of the power uprate, remains within the 2-percent RTP assumption of these evaluations. Therefore, the final calculated uncertainties determine the magnitude of the power uprate. The primary means of reducing the power calorimetric uncertainties is a reduction in the uncertainties associated with the measurement of secondary-side feedwater flow. New calculations were performed to determine the uncertainties for the daily power calorimetric assuming the use of the Caldon Leading Edge Flow Meter (LEFM) Check System to determine total feedwater flow. The uncertainty allowance for feedwater system flow is ±0.38 percent. The flow error, in combination with the remaining uncertainty components, results in a total 95/95 power measurement uncertainty of ±0.6 percent RTP. A power measurement uncertainty of ±0.6 percent allows a power uprate of 1.4-percent RTP. The methodology used to determine the power calorimetric uncertainties is documented in WCAP-15904 (Reference 4-2).

4.2 REVISED THERMAL DESIGN PROCEDURE UNCERTAINTIES

4.2.1 T_{avg} (Rod) Control and Pressurizer Pressure Control

The uncertainties associated with the T_{avg} and pressurizer pressure control systems are not affected by changes in the IP2 design parameters for the 1.4-percent power uprate conditions. Therefore, the 1.4-percent power uprate does not require changes to the uncertainties documented in WCAP-13825 for these parameters.

4.2.2 Reactor Coolant System Flow Calculation

The RCS flow uncertainty calculation uses nominal plant conditions for feedwater temperature and steam pressure as part of the input assumptions for the calculations. The small changes in these plant parameters due to the 1.4-percent power uprate conditions do not change the final calculated RCS flow uncertainties. Therefore, the 1.4-percent power uprate does not require changes to these uncertainties as documented in WCAP-13825.

4.3 RTS/ESFAS UNCERTAINTIES

The licensing basis accident analyses, which model RTS and ESFAS initiations for mitigation, have been reviewed. It has been confirmed that their results remain acceptable (see Section 8) using the currently applied safety analysis assumptions and the initial condition inputs based on the 1.4-percent increase in nominal core power. As such, an additional review was conducted to assess the influence of the slight fluid condition changes associated with the 1.4-percent power uprate on total uncertainties calculated as the basis for the implemented RTS/ESFAS setpoints. The fluid condition changes assumed for this review are given in Section 2 of this report. The results of this uncertainty evaluation are provided below.

4.3.1 RTS Functions

Power Range Neutron Flux – High

As T_{avg} is essentially unchanged by the 1.4-percent power uprate condition, aggregate neutron leakage (vessel) characteristics are expected to be unchanged and flux monitoring sensitivity is also unchanged. The existing total uncertainty for this protection function includes a full 2-percent calorimetric uncertainty allowance and this value has not been reduced as a result of the 1.4-percent power uprate. Therefore, the existing uncertainty value for this function is unaffected as a result of the 1.4-percent power uprate.

Power Range Neutron Flux – Low

The conditions described above for the high trip function of the Nuclear Instrumentation System (NIS) power range are also applicable to the low trip function. Power ascension to an unchanged full-power T_{avg} value will ensure similar sensitivity at low power when transitioned from the current licensed power level to the 1.4-percent uprate power level. The calorimetric uncertainty allowance is also the same. Therefore, the existing uncertainty value for this function is unaffected as a result of the 1.4-percent power uprate.

Overtemperature ΔT (OTΔT)

The RCS full-power T_{hot} and T_{cold} values will change slightly as a result of the 1.4-percent power uprate, and this will have a slight effect on the indicated values for delta temperature (ΔT) and T_{avg}. These small changes are accommodated by existing plant setpoints. Full-power ΔT will increase slightly, which will have a conservative effect on the uncertainty calculations. Therefore, the existing uncertainty value for this function is unaffected as a result of the 1.4-percent power uprate.

Overpower ΔT (OPΔT)

As discussed for OTΔT, the small changes to T_{hot}, $T_{cold,}$ $T_{avg,}$, and ΔT do not affect the existing uncertainty value for this function. Therefore, this function is unaffected as a result of 1.4-percent power uprate.

Pressurizer Pressure – Low and High

Nominal pressurizer pressure (i.e., 2250 psia) is unaffected by the 1.4-percent power uprate and the existing uncertainty values for these functions are unaffected as a result of the 1.4-percent power uprate.

Pressurizer Water Level – High

Although the nominal operating conditions for the 1.4-percent power uprate are bounded by the previously evaluated design conditions, a small change to the design pressurizer level versus T_{avg} program was made to accommodate the potential operation at the upper end of the design T_{avg} window. This change did not affect the uncertainties associated with the pressurizer water level high trip, but did result in a slight increase in the level uncertainties associated with the accident analysis initial conditions. The revised uncertainty is bounded by the allowances incorporated in the accident analysis. Therefore, the pressurizer level uncertainties are consistent with operation at a power increase of 1.4 percent.

Reactor Coolant Flow – Low

The small changes in the plant parameters due to the 1.4-percent power uprate do not affect the final calculated RCS flow calorimetric uncertainties. Therefore, the low RCS flow trip function is unaffected by the 1.4-percent power uprate.

Underfrequency and Undervoltage Reactor Coolant Pumps (6.9 kV Bus)

Bus underfrequency and undervoltage relay uncertainties are not subject to influences associated with fluid parametric changes caused by the 1.4-percent power uprate. Also, nominal frequency and voltage conditions both inside the plant and on the local grid are unaffected by any minor load changes that exist in the plant as a result of the 1.4-percent power uprate. Therefore, the underfrequency and undervoltage functions are unaffected by the 1.4-percent uprate.

Steam Generator Narrow Range Water Level – Low-Low

Process Measurement Accuracies (PMAs) considered in the steam generator level calculation take into account specific effects such as fluid pressure/specific gravity variations and reference leg temperature effects. The final calculated results, which account for the minor variations (approximately 2 psi decrease in minimum steam pressure) associated with the revised operating parameters, are unaffected by the 1.4-percent power uprate.

4.3.2 ESFAS Functions

Containment Pressure – High and High-High

Containment pressure instrument uncertainties are not affected by the fluid system parametric changes associated with the 1.4-percent power uprate. Therefore, the containment pressure functions are unaffected by the 1.4-percent power uprate.

Pressurizer Pressure – Low

Nominal pressurizer pressure (i.e., 2250 psia) and the existing uncertainty value for this function are unaffected by the 1.4-percent power uprate.

High Steam Line Differential Pressure

Steam line pressure instrument uncertainties are not affected by the fluid system parametric changes associated with the 1.4-percent power uprate. Therefore, the steam line differential pressure uncertainties are unaffected by the 1.4-percent power uprate.

High Steam Flow in Two Lines

The uncertainty calculations associated with the high steam flow instrumentation were reviewed based on the operating parameters associated with the 1.4-percent power uprate. Several of these parameters (steam flow, steam pressure, and turbine first-stage pressure, which defines the variable flow setpoint based on power) provide input to the scaling process used to define the variable trip setpoint for this function. The uncertainty calculations, which include scaling tolerances that are consistent with operation at the 1.4-percent power uprate conditions, demonstrate margin between the setpoint and analytical limits.

T_{avg} – Low

The RCS T_{hot} and T_{cold} values will change slightly at the 1.4-percent power uprate conditions, but the uprate T_{avg} is approximately the same as at the previous licensed power level. Therefore, the Low T_{avg} ESFAS function is unaffected by the 1.4-percent power uprate.

Steam Line Pressure – Low

Steam line pressure instrument uncertainties are not affected by the fluid system parametric changes associated with the 1.4-percent power uprate. Therefore, the seam line pressure low uncertainties are unaffected by the 1.4-percent power uprate.

Steam Generator Water Level – Low-Low

The PMAs considered in the steam generator water level calculation take into account specific effects such as fluid pressure/specific gravity variations and reference-leg temperature effects. The final calculated results, which account for the minor variations (approximately 2 psi decrease in minimum steam pressure) associated with the revised operating parameters, are unaffected by the 1.4-percent power uprate.

Degraded Voltage, Loss of Voltage, and Station Blackout (480 VAC)

The 480 VAC Emergency Bus Undervoltage relay and timer uncertainties are not subject to influences associated with fluid parametric changes caused by the 1.4-percent power uprate. Therefore, the 480 VAC ESFAS functions are unaffected by the 1.4-percent power uprate.

4.4 STEAM GENERATOR WATER LEVEL SETPOINT UNCERTAINTY ISSUES

Westinghouse recently issued three Nuclear Service Advisory Letters (NSALs), NSAL 02-3 and Revision 1, NSAL 02-4, and NSAL 02-5, to document the problems with the Westinghouse designed steam generator water level setpoint uncertainties. The NSAL 02-3 and its revision, issued on

February 15, 2002, and April 8, 2002, respectively, deal with the uncertainties caused by the mid-deck plate located between the upper and lower taps used for steam generator measurements, affecting the low-low level trip setpoint (used in the analyses for events such as the feedwater line break, Anticipated Transient Without Scram (ATWS) and steam line break). The NSAL 02-4, issued on February 19, 2002, deals with the uncertainties created because the void content of the two-phase mixture above the mid-deck plate was not reflected in the calculation, affecting the high-high level trip setpoint. The NSAL 02-5, issued on February 19, 2002, deals with the initial conditions assumed in the steam generator water level related safety analyses. The discussion below addresses how IP2 accounts for these uncertainties as documented in the NSALs in determining the steam generator water level setpoints.

Indian Point Unit 2 is operating with Westinghouse Model 44F steam generators. In comparison to other Westinghouse steam generator models, the Model 44F steam generator has a relatively large flow area through the mid-deck plate region of the steam generator and, as such, there is essentially no pressure drop across the mid-deck plate region of the Model 44F steam generators. This is shown for IP2 in the attachment to NSAL 02-3, Revision 1. Therefore, with respect to NSAL 02-3 and NSAL 02-5, IP2 is not affected and the current safety analyses remain limiting.

For NSAL-02-4 (Maximum Reliable Indicated Level - MRIL), the IP2 safety analysis limit for steam generator level high is 80-percent span. This value is considerably lower than the MRIL calculated in accordance with NSAL-02-04. Since the setpoint evaluation of the 73-percent nominal trip setpoint was based on the limit of 80 percent, the concerns identified in the NSAL have been addressed for IP2.

4.5 REFERENCES

4-1 WCAP-13825, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology Consolidated Edison – Indian Point 2, T. P. Williams, February 1994.

4-2 WCAP-15904, "Power Calorimetric Uncertainty for the 1.4% Uprating for Entergy Indian Point Unit 2," M. D. Coury.

5 DESIGN TRANSIENTS

5.1 NUCLEAR STEAM SUPPLY SYSTEM DESIGN TRANSIENTS

The Indian Point Unit 2 (IP2) 1.4-percent power uprate results in a slight change in the Nuclear Steam Supply System (NSSS) design parameters from those that were used for the existing 3083.4 MWt design transient and Model 44F Steam Generator Replacement Programs. These include slight changes to the parameters that are important to the analysis of the NSSS design transients used for structural fatigue analysis of the various NSSS components. These particular parameters are shown in Table 5-1, along with the current and 1.4-percent power uprate values. This section of the report summarizes the review of the NSSS design transients and the potential need to revise some transient definitions to account for the 1.4-percent power uprate.

5.1.1 NSSS Design Transient Background

The NSSS design transients are included in the various component design specifications, and are used to perform fatigue stress analyses on these NSSS components. These transients include the transient profile (i.e., parameter variation during the transient) and the number of assumed occurrences of the transient over the plant design lifetime. The transient profiles show the variations in the following parameters:

- Reactor Coolant System (RCS) T_{hot} (generally reported as "variation" or "change from initial")
- RCS T_{cold} (generally reported as "variation" or "change from initial")
- RCS pressure (generally reported as "variation" or "change from initial")
- RCS flow (generally reported as "normalized" or "fraction of nominal")
- Pressurizer pressure (generally reported as "variation" or "change from initial")
- Pressurizer surge flow (generally reported as "normalized")
- Pressurizer spray flow (generally reported as "normalized")
- Steam generator steam temperature (generally reported as "variation" or "change from initial")
- Steam generator steam and feedwater flows (generally reported as "normalized" or "fraction of nominal")

In addition, the pressurizer design transients include additional information such as temperature differential and transient duration for the pressurizer spray and surge nozzles.

The NSSS design transients for IP2 were initially generated back in the late 1960s to early 1970s. They were then revised as necessary as part of the 3083.4 MWt rerating and Replacement Stream Generator Programs in 1988-1989 with revisions to reflect the steam generator primary-to-secondary pressure limit in 2000-2001.

5.1.2 NSSS Design Transient Evaluation

This section provides an evaluation of the continued applicability of the current NSSS design transients for the 1.4-percent uprate conditions.

The plant conditions used for the NSSS design transient evaluations are based primarily on the NSSS Performance Capability Working Group (PCWG) design parameters. Table 5-1 provides a comparison of the PCWG parameters for the 1.4-percent power uprate against those assumed in the current NSSS design transients for IP2. As shown in Table 5-1, there are some minor differences in the T_{cold} and T_{steam} values at the High T_{avg} conditions and in the T_{hot} for the Low T_{avg} conditions. The feedwater temperature increased slightly for both the High and Low T_{avg} conditions. As discussed below, these small differences in the PCWG parameters for the 1.4-percent uprate versus the ones applicable for the current NSSS design transients will have no significant impact on the current NSSS design transients. Furthermore, because of the analysis conservatisms included in the development of the NSSS design transients, the current NSSS design transients continue to represent a conservative set of transients for use in the NSSS component fatigue evaluation for the 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program.

The NSSS design transients are traditionally developed for fatigue stress analyses of the various NSSS components. Conservatism is generally included in them via the analysis assumptions associated with either frequency of occurrences or the transient assumptions. These conservatisms include:

- Frequency of occurrences are developed conservatively. For example, while the plants are operated in a base-loaded fashion, it is assumed that a plant loading from 0-percent to 100-percent power followed by an unloading from 100-percent to 0-percent power occurs every day. For the upset transients, it is assumed a reactor trip from 100-percent power occurs 400 times over the plant life (i.e., 10 times each year for every year of operation). A loss of load is assumed to occur 80 times over the plant life (i.e., 2 times each year for every year of operation). These are conservative in comparison to actual plant operating experience.

- Conservatisms are also included in the transient analysis assumptions. For example, the normal condition design transients are analyzed assuming they are all at beginning-of-life (BOL) conditions, resulting in the minimum reactivity feedback and maximum parameter (i.e., RCS and pressurizer pressure and temperature) transient variations. The loss-of-load transient is analyzed like a conservative Anticipated Transient Without Scram (ATWS) event, with no reactivity feedback, no credit for any control systems, and no reactor trip until the pressurizer is nearly water solid. The reactor trip transient is assumed to occur at BOL core conditions to result in the minimum decay heat and the maximum RCS cooldown.

- The design transients are generally analyzed assuming a 2-percent power uncertainty allowance, which bounds the 1.4-percent power uprate plus the 0.6-percent power measurement uncertainty.

- The existing design transients have been developed using a conservative starting point that results in a conservative parameter transient variation. The 25-percent steam generator tube plugging (SGTP) case results in the maximum change in steam temperature (and consequently steam pressure) from 0-percent to 100-percent power. Also, any transient that results in a steam dump opening or a challenge to the steam generator safety valves would have a larger parameter change if initiated from the 25-percent tube plugging condition. It also results in the maximum steam generator primary-to-secondary pressure differential.

- The steam generator primary-to-secondary pressure differential must stay below the design limit of 1700 psid during any normal condition transient and must not be exceeded by more than 110-percent during any upset condition transient.

Therefore, based on the limiting values of a maximum T_{hot}, minimum T_{cold}, and minimum T_{steam} shown in Table 5-1 being unchanged for the uprating from the present design values, the existing design transients remain valid for the uprating.

The review of the primary-to-secondary pressure differential is contained in Section 7.7.

Table 5-1
IP2 Plant Operating Conditions (25% SGTP)

	Present Power Rating (Reference 5-1)		1.4% Power Uprate	
	Low T_{avg}	High T_{avg}	Low T_{avg}	High T_{avg}
NSSS Power, MWt	3083.4	3083.4	3127	3127
Reactor Coolant Flow, gpm/loop	80,700	80,700	80,700	80,700
Reactor Vessel Outlet Temperature, T_{hot}, °F	582.2	611.7	583.0	611.7
Reactor Vessel Average Temperature, T_{avg}, °F	549.0	579.7	549.4	579.2
Steam Generator Outlet* Temperature, T_{cold}, °F	515.5	547.4	515.5	546.4
Steam Temperature, °F	482.2**	513.5	479.6**	512.0
Steam Pressure, psia	578**	768	564**	758
No-Load Temperature, °F	547	547	547	547
Feedwater Temperature, °F	430	430	431.8	431.8
Feedwater/Steam Flow (Total), 10^6 lb/hr	13.25	13.30	13.47	13.53

Notes:

* Reactor vessel inlet is only 0.4°F higher

** Minimum steam pressure (and resulting steam temperature) to avoid violating the steam generator primary-to-secondary pressure differential is 650 psia. This 650 psia limit (and corresponding steam temperature of 494.9°F) was used for the steam generator design transients only; other NSSS components were evaluated based on the steam temperatures and pressures as shown.

5.2 AUXILIARY EQUIPMENT DESIGN TRANSIENTS

5.2.1 Introduction

The IP2 auxiliary equipment design specifications included transients that were used to design and analyze the Class 1 auxiliary nozzles connected to the RCS and certain NSSS auxiliary systems piping, heat exchangers, pumps, and tanks. These transients are described by variations in pressure, fluid temperature, and flow and represent bounding cases for operational events postulated to occur during the plant lifetime. To a large extent, the transients are based on engineering judgement and experience and are considered to be of such magnitude and/or frequency as to be significant in the component design and fatigue evaluation processes. The transients are sufficiently conservative, such that, when used as a basis for component fatigue analysis, they provide confidence that the component will perform as intended over the operating license period of the plant. For purposes of analysis, the number of transient occurrences was based on an operating license period of 40 years.

As part of the IP2 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program, the auxiliary equipment design transients were reviewed to assess continued applicability.

5.2.2 Input Parameters and Assumptions

The review of the auxiliary equipment design transients was performed based on the range of NSSS design parameters developed to support an NSSS power level of 3127 MWt (Table 2-1).

The approved range of NSSS design parameters for the 1.4-percent power uprate were compared with the NSSS design parameters used to developed the current design-bases transients.

5.2.3 Description of Analyses and Evaluation

An evaluation of the current design transients was performed to determine which transients could be potentially impacted by the 1.4-percent power uprate. The evaluation concluded that the only design transients that could potentially be impacted by the power uprate are those temperature transients impacted by full-load RCS design temperatures.

These temperature transients are defined by the differences between the temperature of the coolant in the RCS loops and the temperature of the coolant in the auxiliary systems connected to the RCS loops. The greater the temperature difference, the greater the impact these temperature transients have on auxiliary component design and fatigue evaluation processes. Since the operating coolant temperatures in the auxiliary systems are not impacted by the 1.4-percent power uprate, the temperature difference between the coolant in the auxiliary systems and the coolant in the RCS loops is only impacted by changes in the RCS operating temperatures.

The current design temperature transients are based on a full-load T_{hot} of 630°F and a full-load T_{cold} of 560°F. These full-load temperatures were assumed for equipment design to ensure that the temperature transients would be conservative for a wide range of NSSS design parameters.

5.2.4 Acceptance Criteria and Results

A comparison of the range of NSSS design temperatures for the 1.4-percent power uprate at full-load (that is, T_{hot} (583.0 - 611.7°F) and T_{cold} (515.8 – 546.7°F), with the T_{hot} and T_{cold} values used to develop the current design transients) indicates that the power uprate temperature ranges are lower. These lower full-load operating temperatures result in less severe transients, since the temperature differences between RCS loop temperatures and the lower operating temperatures in the auxiliary systems connected to the RCS are less. For example, the temperature transients imposed on the Chemical and Volume Control System letdown and charging nozzles associated with starting and stopping letdown and charging flow would be less severe, since the temperature differences are less. Therefore, the current body of auxiliary design transients are conservative for the 1.4-percent power uprate.

5.2.5 Conclusions

The only auxiliary equipment transients that can potentially be impacted by a 1.4-percent power uprate are those temperature transients related to full-load NSSS design temperatures. A review of these temperature transients indicates that, if these transients were based on the 1.4-percent power uprate design parameters, they would be less severe. Therefore, the current auxiliary equipment design transients for IP2 remain bounding for the proposed 1.4-percent power uprate.

5.3 REFERENCES

5-1 WCAP-12187, "Consolidated Edison Company of New York, Inc., Indian Point Unit 2 3083.4 MWt Stretch Rating Engineering Report," March 1989.

6 NUCLEAR STEAM SUPPLY SYSTEMS

This section discusses the evaluations performed on the Nuclear Steam Supply System (NSSS) systems in support of the revised design parameters discussed in Section 2. The systems that could potentially be affected by the Indian Point Unit 2 (IP2) 1.4-percent power uprate that are discussed in this section are the NSSS fluid systems, the NSSS/Balance-of-Plant (BOP) interface systems, and the NSSS control systems. The performance and integrity of these systems, except Residual Heat Removal System (RHRS) performance, are unaffected by the 1.4-percent power uprate. For RHRS performance, the IP2 plant cooldown cases were analyzed based on the 1.4-percent power uprate and shown to still meet applicable acceptance criteria.

6.1 NSSS FLUID SYSTEMS

6.1.1 Reactor Coolant System

Reactor Coolant System Design Parameters

The NSSS design parameters at the uprated power level are discussed in Section 2. The primary change in parameters that affect Reactor Coolant System (RCS) performance are core power and the resulting full-load T_{cold} and T_{hot} temperatures. The steady-state RCS pressure (2235 psig), no-load RCS temperature (547°F), and RCS flows have not changed. The change in full-load RCS temperatures are shown below:

RCS Temperatures	Current Parameters	Uprated Parameters
T_{cold} (Steam Generator Outlet)	547.4°F	546.4°F
T_{hot} (Vessel Outlet)	611.7°F	611.7°F

RCS Design Temperature and Pressure

The RCS is specified with a design pressure of 2485 psig and a nominal operating pressure of 2235 psig. The RCS design temperature is 650°F with the exception of the pressurizer, which is designed to 680°F. Based on the uprated RCS parameters discussed above, the RCS design pressure and temperature continue to bound the uprated operating conditions. Therefore, it is concluded that the RCS design temperature and pressure are not affected by the uprated conditions, and the integrity of the RCS pressure boundary is maintained within the original design limits.

RCS Heat Capacity

The RCS heat capacity is defined as the amount of heat (Btu) to raise or lower the RCS temperature 1°F (i.e., Btu/°F), or, the amount of sensible heat that must be removed or added to the RCS for a given change in RCS temperature. The RCS heat capacity is derived from the composite of RCS fluids and component masses, both of which are not changing as a result of the 1.4-percent power uprate. Therefore, it is concluded that the RCS heat capacity is not affected by the 1.4-percent power uprate.

Reactor Coolant Pump Net Positive Suction Head

Adequate reactor coolant pump (RCP) net positive suction head (NPSH), at the RCP suction, is monitored by using the RCS wide-range pressure instrument. Since the RCS wide-range pressure instrument is somewhat removed from the RCP suction point (e.g., wide-range pressure instrument located in the RCS hot leg), the pressure drop from the RCS wide-range pressure transmitter to the RCP suction is accounted for when using this instrument for RCP NPSH. This pressure drop is a function of RCS flow, in addition to other plant physical parameters such as RCS component and piping losses. As indicated by the RCS design parameters, RCS flow does not change as a result of the 1.4-percent power uprate. Since there are no plant changes for the 1.4-percent power uprate that could effect the RCS hydraulic performance for RCP NPSH, including RCS flow, it is concluded that RCP NPSH is not affected by the 1.4-percent plant uprate.

Pressurizer Spray Flow

The driving head for pressurizer spray is a function of RCS flow and temperature. Since the changes in RCS temperatures are negligible (see above) at the uprated conditions, there is no impact on pressurizer spray performance as a result of the RCS temperature changes at the uprated conditions. The reactor vessel flow of 343,136 gpm was used for determining pressurizer spray flow performance. The best-estimate reactor vessel flow for the uprated conditions is 343,600 gpm. Since the best-estimate flow for the uprated conditions is greater than the 343,136 gpm flow assumed in the spray performance analysis, there is no impact on pressurizer spray flow performance as a result of the uprated power conditions.

Pressurizer Spray and Surge Line Low-Temperature Alarms

These instruments are provided to indicate that the minimum spray and surge line flows are met, so that thermal shock to these lines is minimized when these lines are in use. Since the changes in 1.4-percent uprated no-load and full-power RCS hot- and cold-leg temperatures (see above RCS temperatures) are insignificant, the setpoints of these instruments are not affected by the uprated conditions.

Pressurizer Relief Tank

The pressurizer relief tank (PRT) limiting design basis is to accept and quench the design-basis discharge from the pressurizer steam space. The PRT is sized to condense and cool a discharge of steam equivalent to 110 percent of the full-power pressurizer steam volume. The amount of energy absorbed by the PRT is related to the volume and pressure of the steam discharged. As discussed in the above sections for RCS design parameters and RCS design temperature and pressure, RCS pressure does not change for the 1.4-percent uprated conditions. However, pressurizer level changes as a result of the uprated conditions. The containment maximum ambient temperature of 130°F is also addressed.

The sensitivity of the PRT initial water temperature on the PRT design-basis performance has been previously evaluated. Specifically, a 130°F initial PRT water temperature was evaluated. Acceptable PRT performance was demonstrated for 130°F, with the PRT setpoints and parameters validated for the 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program. Therefore, it is concluded that the PRT performance remains acceptable with a 130°F initial PRT temperature and assuming the design-basis discharge defined above.

The loss-of-load transient was re-analyzed (using a pressurizer level based on the uprated conditions), which is associated with the design PRT steam discharge from the pressurizer. According to this analysis, the pressurizer steam released remains bounded by the PRT design conditions described above. Therefore, it is concluded that the PRT is not affected by the 1.4-percent power uprate, including the changes in pressurizer level and the maximum ambient containment temperature of 130°F.

6.1.2 Natural Circulation Cooldown Capability

The loss of all alternating current (AC) power to the station auxiliaries analysis (Updated Final Safety Analysis Report (UFSAR) Section 14.1.12), which takes credit for natural circulation, was analyzed with 2-percent power measurement uncertainty. This is discussed in Section 8.3. The use of the 2-percent power uncertainty, combined with the current power level, is equivalent to modeling the plant at the 1.4-percent uprated power level with the reduced uncertainty of 0.6 percent.

6.1.3 NSSS Auxiliary Systems

The following NSSS Auxiliary Systems are addressed in this section:

- Chemical and Volume Control System (CVCS)
- Emergency Core Cooling System (ECCS)
- Residual Heat Removal System
- Primary Sampling System (PSS)
- Component Cooling Water System (CCWS)

Chemical and Volume Control System

Regenerative Heat Exchanger

The regenerative heat exchanger cools the normal letdown flow from the RCS, which is at RCS T_{cold} temperature. The design inlet (RCS T_{cold}) temperature is 554.8°F, which bounds the highest RCS T_{cold} temperature associated with the RCS no-load temperature of 547°F. (See above RCS section.) Since the no-load RCS temperature has not changed, and the full-load uprated T_{cold} temperature has decreased by a negligible amount, there is negligible impact on the performance of the regenerative heat exchanger at uprated conditions due to any minor change in letdown flow (from the slight change in RCS T_{cold} temperature) or RCS T_{cold} temperature.

Non-Regenerative Heat Exchanger

The non-regenerative (letdown) heat exchanger cools the letdown flow from the regenerative heat exchanger. Since the change in performance of the regenerative heat exchanger is negligible at uprated conditions, as discussed in the previous section, there will be a negligible impact on the performance of the non-regenerative heat exchanger. The minor difference in performance can easily be accommodated by the non-regenerative heat exchanger cooling water temperature control valve AC-TCV-130.

Excess Letdown Heat Exchanger

The excess letdown heat exchanger cools the excess letdown flow from the RCS, which is at RCS T_{cold} temperature. The design inlet (RCS T_{cold}) temperature is 554.8°F, which bounds the highest RCS T_{cold} temperature associated with the RCS no-load temperature of 547°F. (See above RCS section.) Since the no-load RCS temperature has not changed, and the full-load uprated T_{cold} temperature has decreased by a negligible amount, there is negligible impact on the performance of the excess letdown heat exchanger at uprated conditions due to the change in RCS T_{cold} temperature.

Seal Water Heat Exchanger

The seal water heat exchanger cools the seal return flow from the four RCP number one seals, in addition to the excess letdown flow (from the excess letdown heat exchanger) if in service. The RCP heat load (including the thermal barrier heat exchanger) is a function of RCS T_{cold} temperature, while the excess letdown flow heat load is a function of excess letdown heat exchanger performance. (See above discussion for the excess letdown heat exchanger.) Since the no-load RCS temperature has not changed, and the full-load uprated T_{cold} temperature has decreased by a negligible amount, there is a negligible effect on the performance of the seal water heat exchanger at uprated conditions due to the change in RCS T_{cold} temperature.

Charging, Letdown, and RCS Makeup (Boration, Dilution and N-16 Delay Time)

As discussed in the above sections for the various CVCS heat exchangers, there is negligible effect on their performance as a result of the uprated conditions. Therefore, it can be concluded that there will also be a negligible impact on the charging (including RCP seal injection) and letdown performance provided by the CVCS. The flow capacity performance of the RCS makeup system is independent of the change in RCS conditions resulting from the uprated conditions. However, the makeup system also relies on storage capacity of various sources of water including primary makeup water and boric acid solutions from both the boric acid storage tanks and the refueling water storage tank (RWST).

Primary makeup water is used to dilute RCS boron, for purposes of providing positive reactivity control (e.g., increasing core reactivity), or for blending concentrated boric acid to match the prevailing RCS boron concentration during RCS inventory makeup operations (e.g., to maintain volume control tank level). Since the flow capacity performance of the RCS makeup system is independent of the change in RCS conditions resulting from the uprated conditions as discussed above, the plant uprate does not affect the capability of the makeup system to perform these makeup system functions.

The boric acid storage tanks and RWST provide the sources of boric acid for purposes of providing negative reactivity control (e.g., decreasing core reactivity), in addition to the reactor control rods. The plant uprate (i.e., increased core power level) is expected to have a small impact on the boration requirements that must be met by the CVCS boration capabilities. However, this capability is beyond the scope of this report. The Westinghouse Reload Safety Evaluation (RSE) process is designed to address boration capability due to routine plant changes, such as core reloads, and infrequent plant changes such as a plant uprate that results in a change to core operating conditions. Therefore, boration capability will be addressed as part of a future RSE revision in support of this plant uprate for the current cycle core. Future RSEs will consider the uprated power condition and, therefore, will address CVCS boration for future core reloads.

The letdown flow path is routed inside containment such that there is adequate decay of N-16 before the letdown fluid leaves the Containment Building. Since the change in letdown flow is considered negligible, as discussed in the previous paragraphs (e.g., due to the slight change in RCS T_{cold} temperature), this radiation protection feature of the CVCS is not affected by the plant uprate.

Emergency Core Cooling System

The scope of this discussion regarding the ECCS includes the Safety Injection Systems (both low-head and high-head systems) and Containment Spray System performance. Subsequent to ECCS actuation, the Safety Injection System draws water from the RWST during the injection phase and delivers to the RCS, while the Containment Spray System simultaneously draws from the RWST and sprays the containment atmosphere. At the conclusion of RWST drain down, operation of the Containment Spray System is terminated. Also, the Safety Injection System is switched over to the containment recirculation alignment, drawing fluid from the containment sump. The safety injection system can also provide recirculation spray to the Containment Spray System, if required for continued containment cooling, during the recirculation phase.

The plant changes associated with the 1.4-percent power uprate do not impact the hydraulic performance of these systems during the injection phase since the RWST temperature is not changed. There could be a small impact (a slight increase in sump fluid temperature) during recirculation, since decay heat slightly increases with power level. However, the post-LOCA containment sump temperature performance analysis has been determined to bound the uprated conditions. Therefore, it is concluded that ECCS hydraulic performance is not affected by the plant uprate.

Residual Heat Removal System

The 1.4-percent power uprate affects the plant cooldown time since no additional margin (e.g., 102-percent reactor power) has been applied to the core power level assumed in the cooldown analysis of record. Therefore, updated cooldown cases to account for the 1.4-percent uprate conditions were analyzed. The two-train system alignment case was considered to address the design and UFSAR bases cases. In addition, a single-train cooldown analysis was performed to support the worst-case scenario for the Code of Federal Regulations (CFR) 10 CFR 50 Appendix R fire hazards analysis. The following considerations were applied to these cooldown analyses.

The Component Cooling Water (CCW) and Residual Heat Removal (RHR) heat exchanger data assumes five-percent steam generator tube plugging (SGTP), as was used for the previous cooldown analyses of record (resulting in slightly degraded normal cold shutdown and Appendix R cooldown performance).

The evaluation accounts for the RHR system cooldown capacity loss (extended the cooldown time to the refueling temperature - 140°F) due to the RHR pump miniflow, which always remains open in the IP2 RHRS design.

For normal cooldown, the spent fuel pool (SFP) heat load was increased by 1.4 percent to account for the uprated core power, thus maintaining the same level of margin as in the previous analysis of record.

For Appendix R cooldown, the SFP is assumed to be isolated at the time RHR is initiated, which is consistent with the assumptions from the previous Appendix R analysis of record.

The RHR cooldown analyses results are given in Table 6-1.

Table 6-1
RHR Cooldown Analyses Results

Cases	Mode 6 Time to Cool Down to 140°F (Hours after Reactor Shutdown)	Mode 5 Time to Cool Down to 200°F (Hours after Reactor Shutdown)	Mode 4 Time RHR Initiated @ 350°F (Hours after Reactor Shutdown)
1. Normal Cooldown with SFP Heat Load	101.1	33.0	10.0
2. Appendix R Cooldown without SFP Heat Loads	N/A	71.89	46.0 [1]

Note:

1. RHR is initiated at this time to prevent RCS heatup (above 350°F) after RHR cut-in.

Case 1 is the 2-train normal cooldown case described in the UFSAR. Case 1 includes the SFP heat load as part of the CCW system auxiliary heat loads, while Case 2 does not. Case 2 is the single train Appendix R cooldown case. To meet the 72-hour cold shutdown time limit for Appendix R cooldown, the SFP heat load is assumed to be isolated at the time RHR is initiated (at 46 hours after reactor shutdown). The UFSAR will need to be updated to reflect these revised cooldown times. The UFSAR normal cooldown time is no longer 72.6 hours, but is now 101.1 hours after reactor shutdown to reach 140°F. Additionally, the time to cool to 200°F (cold shutdown) will need to be revised to 33 hours, as opposed to 19.4 hours currently reported in the UFSAR. It is also noted that the Case 2 cooldown meets the 72-hour Appendix R requirement, and the normal plant cooldown time changes do not affect plant safety.

Primary Sampling System

The scope of this evaluation is limited to the high-pressure, remotely obtained samples from the RCS since these sample locations set the limiting process conditions that govern the design of the Primary Sampling System (PSS) and associated sample coolers. The limiting duty for the RCS sample coolers is based on the capability of the cooler to condense and cool a sample stream from the pressurizer steam space. The maximum normal steam condition within the pressurizer is based on the saturation steam temperature at normal operating RCS pressure, since the pressurizer is maintained at saturation conditions for RCS pressure control. As discussed in the RCS section above, the RCS operating pressure has not changed at the uprated conditions. Therefore, the design duty of the PSS is not affected as a result of the 1.4-percent plant uprate.

Component Cooling Water System

Normal Plant Operations (at Power and Refueling)

The normal plant heat loads on the CCWS are as follows:

- Residual heat exchangers
- Reactor coolant pumps
- Nonregenerative heat exchanger
- Excess letdown heat exchanger
- Seal water heat exchanger
- Sample heat exchangers
- Waste gas compressors
- Reactor vessel support pads
- Residual heat removal pumps
- Safety injection pumps
- Recirculation pumps
- Spent fuel pool heat exchanger
- Charging pumps, fluid drive coolers, and crankcase

Of the CCWS heat loads discussed above, the SFP is the only heat load with a potential to affect the CCWS during normal plant operation. The SFP cooling system is addressed in Section 10.10. All other heat loads are not affected by the plant uprate during normal (at-power) plant operation. Therefore, it is concluded the CCWS is not affected by the 1.4-percent power uprate during normal plant operation.

Normal and 10 CFR 50 Appendix R (Fire Protection) Plant Cooldown

The CCWS provides cooling to the RHR heat exchangers during plant cooldown. (See the RHRS section above for further details of plant cooldown performance.) During plant cooldown, the RHR heat exchanger heat load is controlled (by throttling RCS flow) so that an acceptable CCW supply temperature is maintained to the CCW-serviced equipment. Based on the results of the updated RHR cooldown work described in the RHRS section above, the same CCW supply temperatures of record have been maintained. For normal cooldown, the CCW supply temperature is limited to 120°F while for Appendix R cooldown, the CCW supply temperature is limited to 125°F as in the past. Therefore, it is concluded that the CCWS is not affected by the plant uprate during plant cooldown.

Post-Loss-of-Coolant Accident Plant Cooldown

The CCWS supports post-loss-of-coolant accident (LOCA) ECCS operation during recirculation by providing cooling to the RHR heat exchangers. As described in the above ECCS section, there could be a small impact (a slight increase in sump fluid temperature) during recirculation, since decay heat slightly increases with power level. However, the post-LOCA containment sump temperature performance has been determined to be unaffected by the uprated conditions, since uprated reactor power remains bounded the containment analysis of record. Therefore, it is concluded that the CCWS is not affected by ECCS performance at the plant uprated conditions.

6.2 NSSS/BALANCE-OF-PLANT INTERFACE SYSTEMS

6.2.1 Introduction and Background

As part of the IP2 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program, the following balance-of-plant (BOP) fluid systems were reviewed to assess compliance with Westinghouse NSSS/BOP interface guidelines (Reference 6-1):

- Main Steam System (MSS)
- Steam Dump System (SDS)
- Condensate and Feedwater System (C&FS)
- Auxiliary Feedwater System (AFWS)
- Steam Generator Blowdown System (SGBS)

The review was performed based on the range of NSSS design parameters approved for an NSSS power level of 3127 MWt (Table 2-1). The various interface systems were reviewed with the purpose of providing interface information, which could be used in the more detailed BOP analyses.

6.2.2 Input Parameters and Assumptions

A comparison of the uprate design parameters (Reference 6-2) with the non-uprate design parameters previously evaluated for systems and components indicates differences that could impact the performance of the BOP systems. For example, the increase in core power of 1.4 percent (from 3071.4 to 3115 MWt) coupled with zero SGTP and a 1.8°F increase in feedwater temperature (from 430° to 431.8°F) would result in about a 1.7-percent increase in steam/feedwater mass flow rates. Additionally, the average SGTP level of 25 percent in combination with the upper limit on T_{avg} (579.2°F) would result in a reduction in full-load steam pressure from 768 to 758 psia.

6.2.3 Description of Analyses and Results

Evaluations of the above BOP systems relative to compliance with Westinghouse NSSS/BOP interface guidelines (Reference 6-1) were performed to address the NSSS design parameters for the 1.4-percent power uprate, which include ranges for parameters such as T_{avg} (549.4°F to 579.2°F) and SGTP (0 to 25 percent). These ranges on NSSS design parameters result in ranges on BOP parameters such as steam generator outlet pressure (564 psia to 855 psia) and steam/feedwater mass flow rates (13.47×10^6 lb/hr to 13.58×10^6 lb/hr). The NSSS/BOP interface evaluations were performed to address these NSSS and BOP design parameters. The results of the NSSS/BOP interface evaluations are delineated below.

Main Steam System

The uprating coupled with the potential reduction in full-load steam pressure to the design value of 758 psia (Table 2-1) adversely impacts main steam line pressure drop. At the reduced full-load steam pressure of 758 psia coupled with a feedwater temperature of 431.8°F, the steam line volumetric flow rate would increase by approximately 3.1 percent and steam line pressure drop would increase by approximately 4.8 percent due to lower density steam coupled with increased feedwater mass flow rate.

The following subsections summarize the evaluation of the MSS major components relative to the 1.4-percent uprate parameters. The major components of the MSS are the steam generator main steam safety valves (MSSVs), the steam generator power-operated atmospheric relief valves (ARVs), and the main steam isolation valves (MSIVs) and non-return valves.

Steam Generator Main Steam Safety Valves

The setpoints of the MSSVs are determined based on the design pressure of the steam generators (1085 psig) and the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code. Since the design pressure of the steam generator has not changed with the 1.4-percent power uprate, there is no need to revise the setpoints of the safety valves.

The MSSVs must have sufficient capacity so that main steam pressure does not exceed 110 percent of the steam generator shell-side design pressure (the maximum pressure allowed by the ASME B&PV Code) for the worst-case loss-of-heat-sink event (Reference 6-2). Based on this requirement, Westinghouse applies the conservative criterion that the valves should be sized to relieve 100 percent of the maximum calculated steam flow at an accumulation pressure not exceeding 110 percent of the MSS design pressure (Reference 6-1).

Indian Point Unit 2 has 20 safety valves with a total rated capacity of 15.108×10^6 lb/hr, which provides about 111.2 percent of the maximum uprated full-load steam flow of 13.58×10^6 lb/hr (Table 2-1). Therefore, based on the range of NSSS design parameters for the uprate, the capacity of the installed MSSVs meets the Westinghouse sizing criterion.

The original design requirements for the MSSVs (as well as the ARVs and steam dump valves) included a maximum flow limit per valve of 890,000 lb/hr at 1085 psig. Since the actual capacity of any single MSSV, ARV, or steam dump valve is less than the maximum flow limit per valve, the maximum capacity criteria are satisfied.

Steam Generator Power-Operated ARVs

The ARVs, which are located upstream of the MSIVs and adjacent to the MSSVs, are automatically controlled by steam line pressure during plant operations. The ARVs automatically modulate open and exhaust to the atmosphere whenever the steam line pressure exceeds a predetermined setpoint to minimize safety valve lifting during steam pressure transients. As the steam line pressure decreases, the ARVs modulate closed and reseat at a pressure below the opening pressure. The ARV set pressure for these operations is between zero-load steam pressure and the setpoint of the lowest-set MSSVs. Since neither of these pressures changes for the proposed range of NSSS design parameters, there is no need to change the ARV setpoint.

The primary function of the ARVs is to provide a means for decay heat removal and plant cooldown by discharging steam to the atmosphere when either the condenser, the condenser circulating water pumps, or steam dump to the condenser is not available. Under such circumstances, the ARVs in conjunction with the Auxiliary Feedwater System (AFWS) permit the plant to be cooled down from the pressure setpoint of the lowest-set MSSVs to the point where the RHRS can be placed in service. During cooldown, the ARVs are either automatically or manually controlled. In automatic, each ARV proportional and integral (P&I) controller compares steam line pressure to the pressure setpoint, which is manually set by the plant operator.

In the event of a steam generator tube rupture event in conjunction with loss-of-offsite power (LOOP), the ARVs are used to cool down the RCS to a temperature that permits equalization of the primary and secondary pressures at a pressure below the lowest-set MSSV. Reactor Coolant System cooldown and depressurization are required to preclude steam generator overfill and to terminate activity release to the atmosphere (Reference 6-2).

The steam generator ARVs are sized to have a capacity equal to about 10 percent of rated steam flow at no-load pressure (Reference 6-1). This capacity permits a plant cooldown to RHRS operating conditions (350°F) in 4 hours (at a rate of about 50°F/hr) assuming cooldown starts 2 hours after reactor shutdown. This sizing is compatible with normal cooldown capability and minimizes the water supply required by the AFWS. This design basis is limiting with respect to sizing the ARVs and bounds the capacity required for tube rupture.

An evaluation of the installed capacity (2,467,000 lb/hr at 1020 psia) indicates that the original design bases, in terms of plant cooldown capability, can still be achieved for the range of NSSS design parameters for the 1.4-percent power uprate.

MSIVs, MSIV Bypass Valves, and Non-Return Valves

The MSIVs, in conjunction with non-return valves, are located outside the containment and downstream of the MSSVs and ARVs. The valves function to prevent the uncontrolled blowdown of more than one steam generator and to minimize the RCS cooldown and containment pressure to within acceptable limits following a main steam line break. To accomplish this function, the design requirements specified that the MSIVs must be capable of closure within five seconds of receipt of a closure signal against steam break flow conditions in the forward direction.

Rapid closure of the MSIVs and non-return valves following postulated steam line breaks causes a significant differential pressure across the valve seats and a thrust load on the MSS piping and piping supports in the area of the MSIVs and non-return valves. The worst cases for differential pressure increase and thrust loads are controlled by the steam line break area (i.e., mass flow rate and moisture content), throat area of the steam generator flow restrictors, valve seat bore, and no-load operating pressure. Since the 1.4-percent power uprate does not impact these variables, the design loads and associated stresses resulting from rapid closure of the MSIVs and non-return valves will not change. Consequently, the 1.4-percent power uprate has no significant impact on the interface requirements for the MSIVs and non-return valves.

The MSIV bypass valves are used to warm up the main steam lines and equalize pressure across the MSIVs prior to opening the MSIVs. The MSIV bypass valves perform their function at no-load and low-power conditions where power uprate has no significant impact on main steam conditions (e.g., steam flow and steam pressure). Consequently, the 1.4-percent power uprate has no significant impact on the interface requirements for the MSIV bypass valves.

Steam Dump Control System

The steam dump system major components are discussed in Section 10.1.

The NSSS Reactor Control Systems and the associated equipment (e.g., pumps, valves, heaters, control rods) are designed to provide satisfactory operation (automatic in the range of 15- to 100-percent power) without reactor trip when subjected to the following load transients:

- Loading at 5 percent of full power per minute with automatic reactor control
- Unloading at 5 percent of full power per minute with automatic reactor control
- Instantaneous load transients of plus or minus 10 percent of full power (not exceeding full power) with automatic reactor control
- Load reductions of 50 percent of full power with automatic reactor control and steam dump

The SDS creates an artificial steam load by dumping steam from ahead of the turbine valves to the main condenser. The Westinghouse sizing criterion recommends that the steam dump system (valves and pipe) be capable of discharging 40 percent of the rated steam flow at full-load steam pressure to permit the NSSS to withstand an external load reduction of up to 50 percent of plant-rated electrical load without a reactor trip (Reference 6-1). To prevent a trip, this transient requires all NSSS control systems to be in automatic, including the Reactor Control Systems, which accommodate 10 percent of the load reduction. A steam dump capacity of 40 percent of rated steam flow at full-load steam pressure also prevents MSSV lifting following a reactor trip from full power.

<u>SDS Major Components</u>

Indian Point Unit 2 is equipped with 12 condenser steam dump valves and each valve is specified to have a flow capacity of 505,000 lbm/hr at a valve inlet pressure of 650 psia. For the current design parameters, which limit the minimum allowable full-load steam pressure to ≥650 psia (due to steam generator structural concerns), steam dump capacity was reported to be adequate for an external load reduction of up to 50 percent of plant-rated electrical load (Reference 6-3).

The capacity of the SDS (as a percentage of full-load steam flow) decreases as full-load stream pressure decreases and full-load steam flow increases. Nuclear Steam Supply System operation within the proposed range of design parameters for the 1.4-percent power uprate will result in a decrease in steam dump capability due to increased steam flow. The current minimum allowable full-load steam pressure (650 psia) (Table 2-1) based on steam generator structural concerns is not changed. An evaluation indicated the changed steam dump capacity would be reduced to 35.9 percent of rated steam flow (13.49 x 10^6 lb/hr), or 4.844 x 10^6 lb/hr at a full-load steam pressure of 650 psia. At full-load steam pressures higher than 650 psia, steam dump capacity would increase. For example, at a full-load steam pressure of 855 psia (Table 2-1) steam dump capacity would be 48.4 percent of rated flow (13.58 x 10^6 lb/hr), or 6.579 x 10^6 lb/hr.

The NSSS Reactor Control Systems' margin-to-trip analysis provides an evaluation of the adequacy of the SDS in conjunction with the control system setpoints at uprated conditions. Refer to Section 4.3.

To provide effective control of flow on large step-load reductions or plant trip, the steam dump valves are required to go from full-closed to full-open in 3 seconds at any pressure between 50 psi less than full-load pressure and steam generator design pressure. The dump valves are also required to modulate to control flow. Positioning response may be slower with a maximum full stroke time of 20 seconds. These requirements are not impacted by the 1.4-percent power uprate.

Condensate and Feedwater System

The C&FS must automatically maintain steam generator water levels during steady-state and transient operations. The NSSS design parameters will result in a required increase in feedwater volumetric flow (up to 1.8 percent) during full-power operation. The higher feedwater flow will have an impact on system pressure drop, which may increase by as much as 3.5 percent. Also, a comparison of the uprated design parameters with non-uprated design parameters indicates that steam generator full-power operating pressure may decrease by as much as 10 psi (768 psia to 758 psia).

The major components of the C&FS are the main feedwater control valves (FCVs) and the C&FS pumps. Each of these major components are discussed in the following sections.

Main Feedwater Isolation/Feedwater Control Valves

The main FCVs are located outside containment. The valves function in conjunction with backup trip signals to the feedwater pump discharge isolation valves, feedwater pumps, and other miscellaneous valves to provide redundant isolation of feedwater flow to the steam generators following a steam line break or a malfunction in the steam generator level control system. Isolation of feedwater flow is required to prevent containment overpressurization and excessive RCS cooldowns. To accomplish this function, the FCVs and the backup feedwater pump discharge isolation valves must be capable of fast closure, after receipt of a closure signal under all operating and accident conditions. This includes a maximum flow condition with all main feedwater pumps delivering to one steam generator.

The quick-closure requirements imposed on the FCVs and the backup feedwater pump discharge isolation valves causes dynamic pressure changes that may be of large magnitude and must be considered in the design of the valves and associated piping. The worst loads occur following a steam line break from no-load conditions with the conservative assumption that all feedwater pumps are in service providing maximum flow following the break. Since these conservative assumptions are not impacted by the uprate, the design loads and associated stresses resulting from rapid closure of these valves will not change.

Feedwater Control Valves, Condensate, and Feedwater System Pumps

The C&FS available head, in conjunction with the FCV characteristics, must provide sufficient margin for feedwater control to ensure adequate flow to the steam generators during steady-state and transient operation. A continuous, steady, feedwater flow should be maintained at all secondary-system loads. To assure stable feedwater control, with variable speed feedwater pumps, the pressure drop across the FCVs at rated flow (100-percent power) should be approximately equal to the dynamic losses from the feedwater pump discharge through the steam generator. These dynamic losses include the frictional resistance of feedwater piping, high-pressure feedwater heaters, feedwater flow meter, and steam generator. In addition, adequate margin should be available in the FCVs at full-load conditions to permit a C&FS delivery of 96 percent of rated flow with a 100 psi pressure increase above the full-load pressure with the FCVs fully open (Reference 6-1). The current feedwater pump speed control program is set to provide a FCV pressure drop of about 172 psi at full-load. This pressure drop results in a FCV lift of about 80 percent.

An evaluation of the C&FS for the range of design parameters approved for power uprate indicates the change in the lift of the FCVs at full-load (less than 3 percent) with the present feedwater pump speed control program is acceptable for both steady-state and transient operation.

To provide effective control of flow during normal operation, the FCVs are required to stroke open or closed in 20 seconds over the anticipated inlet pressure control range (approximately 0-1600 psig). Additionally, rapid closure of the FCVs is required after receipt of a trip close signal in order to mitigate certain transients and accidents. These requirements are still applicable at the 1.4-percent power uprate conditions (Reference 6-1).

Auxiliary Feedwater System

The AFWS supplies feedwater to the secondary side of the steam generators at times when the normal feedwater system is not available, thereby maintaining the steam generator heat sink. The system provides feedwater to the steam generators during normal unit startup, hot standby, and cooldown operations and also functions as an Engineered Safety Feature (ESF). In the latter function, the AFS is required to prevent core damage and system overpressurization during transients and accidents, such as a loss of normal feedwater or a secondary-system pipe break. The minimum flow requirements of the AFWS are dictated by accident analyses, and since the uprate does affect safety analyses performed at a nominal 100-percent power level, evaluations were performed to confirm that the AFWS performance is acceptable at the 1.4-percent power uprate conditions. These evaluations are described in Section 8.3 of this report and show acceptable results.

Auxiliary Feedwater Storage Requirements

The AFWS pumps are normally aligned to take suction from the condensate storage tank (CST). To fulfill the ESF design functions, sufficient feedwater must be available during transient or accident conditions to enable the plant to be placed in a safe shutdown condition.

The limiting transient with respect to CST inventory requirements is the LOOP transient. The IP2 licensing basis dictates that in the event of a LOOP, sufficient CST usable inventory must be available to bring the unit from full-power to hot-standby conditions, and maintain the plant at hot standby for 24 hours. In light of these design-bases requirements, the IP2 analysis of record concluded that the tank should be designed to accommodate a minimum usable inventory of 284,000 gallons. Accordingly, the Technical Specifications ensure a contained volume of 360,000 gallons.

The minimum required usable inventory (284,000 gallons) is based on reactor trip from 102 percent of maximum calculated power of 3216 MWt (or 3280.3). Since the 1.4-percent power uprate is based on improved calorimetric accuracy, no change in the analysis or the IP2 Technical Specifications is required with regard to auxiliary feedwater storage requirements at the 1.4-percent power uprate level.

Steam Generator Blowdown System

The SGBS is used to control the chemical composition of the steam generator secondary-side water within the specified limits. The blowdown system also controls the buildup of solids in the steam generator secondary side.

The blowdown flow rates required during plant operation are based on chemistry control and tube-sheet sweep requirements to control the buildup of solids. The blowdown flow rate required to control chemistry and the buildup of solids in the steam generators is tied to allowable condenser in-leakage, total dissolved solids in the plant service water, and the allowable primary-to-secondary leakage. Since these variables are not impacted by the power uprate, the blowdown required to control secondary chemistry and steam generator solids will not be impacted by the power uprate.

The inlet pressure to the SGBS varies with steam generator operating pressure. Therefore, as steam generator full-load operating pressure decreases, the inlet pressure to the SGBS control valves decreases and the valves must open to maintain the required blowdown flow rate into the system flash tank. The current design permits a maximum decrease in steam pressure from no load to full load of 370 psi (i.e., from 1020 psia to 650 psia). Based on the revised range of NSSS design parameters approved for the power uprate, the no-load steam pressure (1020 psia) remains the same and the current minimum allowable full-load steam pressure (650 psia) due to steam generator structural concerns is not changed. Therefore, the range of design parameters approved for the power uprate will not impact blowdown flow capability.

6.2.4 Conclusions

The following is a brief summary of the NSSS/BOP interface evaluation conclusions for the IP2 1.4-percent power uprate. Refer to the identified sections for a more detailed discussion.

Main Steam System

- The capacity of the installed MSSVs is adequate to meet the original sizing bases for the approved range of NSSS design parameters.
- The capacity of the installed ARVs is adequate to meet the original sizing for the approved range of NSSS design parameters.
- The 1.4-percent power uprate does not adversely impact the MSIVs and MSIV bypass valves.

Steam Dump System

- An evaluation of the SDS indicates that minimum system capacity is about equal to 35.9 percent of the uprate full-load steam flow at the current minimum allowable full-load steam pressure of 650 psia. At full-load steam pressures higher than 650 psia, steam dump capacity would increase. The control systems margin-to-trip analysis provides an evaluation of the adequacy of steam dump in conjunction the control system setpoints.

Condensate and Feedwater System

- An evaluation of the C&FS indicates the FCVs (in conjunction with the current feedwater pump speed control program) are acceptable for both steady-state and transient operation. This evaluation assumes feedwater pump speed control instrumentation channels are re-scaled to accommodate the increased steam flow at 100-percent power.

Auxiliary Feedwater System

- The minimum flow requirements of the AFWS are acceptable for the 1.4-percent power uprate.
- The CST minimum inventory required by the Technical Specification is adequate for the 1.4-percent power uprate.

Steam Generator Blowdown System

- The blowdown flow required to control secondary chemistry and steam generator solids is not impacted by the power uprate.
- The NSSS design parameters approved for the power uprate, coupled with the current minimum allowable full-load steam pressure, will not impact blowdown flow capability.

6.3 PLANT OPERABILITY

The incorporation of the 1.4-percent power uprate can affect plant operability in the following two ways:

- Pressure control component sizing: this includes the pressurizer heater, spray, and power operated relief valve (PORV) capacities. They have to continue to perform their intended functions successfully.

- Plant margin to trip: this includes the plant response to the normal design-basis plant operability transients, including 5-percent-per-minute loading and unloading, 10-percent step-load increase, large-load rejection, and turbine trip without reactor trip from 35-percent power or less. These transients should not result in the actuation of a reactor trip or Engineered Safety Features Actuation System (ESFAS) actuation and should not challenge the pressurizer or steam generator safety valves. These are the same transients that were reviewed as part of the Stretch Rating Program (Reference 6-2).

This section addresses the continued acceptability of the plant to meet its intended operability requirements. Each of the above functions are reviewed independently.

A comparison between the plant design parameters for the present 3083.4 MWt NSSS power level and for the 1.4-percent power uprate is shown in Table 5-1.

Where analyses were employed to demonstrate that operability requirements continue to be met, the LOFTRAN computer code was used (Reference 6-1). This computer code is Westinghouse configuration controlled and has been approved by the NRC.

6.3.1 NSSS Pressure Control Component Sizing

The various NSSS pressure control components are intended to maintain the pressurizer pressure at the nominal setpoint during steady-state operation, and to control the pressure excursions that occur during design-basis transients to an extent that a reactor trip, ESFAS actuation, or a pressurizer safety valve actuation would not occur. The intent of this assessment is to show that the installed capacity of the various pressure control components remains acceptable for the 1.4-percent power uprate conditions. The results obtained from the IP2 Stretch Rating Program (Reference 6-2) were largely used as a basis for the evaluation.

The following pressure control components were evaluated:

- Pressurizer heaters
- Pressurizer spray valves
- Pressurizer PORVs

Pressurizer Heaters

The pressurizer heaters are sized to be able to heat up the pressurizer liquid at a 200°F/hour rate during the initial plant heatup phase from cold shutdown. In addition, they are intended to assist the plant in controlling the pressurizer pressure decrease that would occur during design-basis transients that result in pressurizer outsurge events. These include the initial part of a 10-percent step-load increase transient, a 5 percent per minute plant unloading transient, or events resulting in a reactor trip. Generic analyses on Westinghouse plants have shown that the pressurizer heater capacity is not a strong influence on the minimum pressure noted during the above operational events, or during reactor trips. The minimum pressure is controlled by the outsurge that results during the transient. Analyses have been performed where the pressurizer heater capacity has been reduced by as much as 20 percent, and no major difference has been observed in the analysis results. The heatup time from cold shutdown to hot standby is not affected by the 1.4-percent power uprate. The heatup maneuver would be essentially the same as that which IP2 presently experiences. Therefore, the installed pressurizer heater capacity is acceptable for the 1.4-percent power uprate.

Pressurizer Spray

The design basis for the pressurizer spray capacity is to be able to handle a 10-percent step-load decrease transient without resulting in the pressure increasing to the pressurizer PORV setpoint. The limiting case is a 10-percent step-load decrease from 100-percent to 90-percent power.

The higher power rating would tend to increase the demand on the pressurizer spray. Therefore, it was decided to re-analyze the pressurizer spray sizing based on current-day analysis methodologies to ensure acceptability. The analysis was performed following the general guidelines presently in use. This included the following assumptions:

- The plant is initially at 100.6 percent of the 1.4-percent uprate power level. The standard analysis methodology is to assume a power uncertainty allowance of 2 percent. The 1.4-percent power uprate plus the 0.6-percent instrument uncertainty is bounded by this original 2-percent power uncertainty allowance.
- The plant is initially at nominal T_{avg} + 4°F uncertainty (standard uncertainty assumption).*
- The transient is a step-load reduction from the noted 100.6-percent turbine load to 89.4-percent load (i.e., 90-percent power plus minus the 0.6-percent instrument uncertainty). This bounds a step-load decrease from 100.6-percent power to a nominal 90-percent power level.
- The steam generator heat transfer coefficient is increased to the maximum credible value (0-percent fouling, 0-percent SGTP).
- Fuel reactivities are at conservative beginning-of-life (BOL) conditions.

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* This uncertainty bounds the calculated uncertainty.

- Credit is taken for automatic operation of all normally functioning NSSS control systems (reactor control, pressurizer pressure and level control, and feedwater control; steam dump is not actuated for a 10-percent step-load transient so it was not credited).

The limiting case is for the plant operating at the upper limit T_{avg} of 579.2°F. For this case, the peak pressurizer pressure was 2330 psia, below the pressurizer PORV setpoint of 2350 psia (2335 psig). Therefore, the installed pressurizer spray capacity is adequate for the 1.4-percent power uprate conditions.

Pressurizer PORVs

The design basis for the pressurizer PORV capacity is to be able to handle a 50-percent step-load decrease transient without resulting in the pressure increasing to the high pressurizer pressure reactor trip setpoint. The limiting case is a 50-percent step-load decrease from 100-percent to 50-percent power.

The higher power rating would tend to increase the demand on the pressurizer PORVs. Therefore, it was decided to re-analyze the pressurizer spray and PORV sizing based on current-day analysis methodologies to ensure acceptability. The analysis was performed following the general guidelines presently in use. This included the following assumptions:

- The plant is initially at 100.6 percent of the 1.4-percent uprate power level. The standard analysis methodology is to assume a power uncertainty allowance of 2 percent. The 1.4-percent power uprate plus the 0.6-percent instrument uncertainty is bounded by this original 2-percent power uncertainty allowance.

- The plant is initially at nominal T_{avg} + 4°F uncertainty (standard uncertainty assumption).*

- The transient is a step-load reduction from the noted 100.6-percent turbine load to 50-percent load.

- The steam generator heat transfer coefficient is increased to the maximum credible value (0-percent fouling, 0-percent SGTP).

- Fuel reactivities are at conservative BOL conditions.

- Credit is taken for automatic operation of all NSSS control systems (reactor control, pressurizer pressure and level control, feedwater control, and steam dump control).

An analysis done for the limiting Low T_{avg} condition of a T_{avg} of 549.4°F resulted in the installed pressurizer PORV capacity not being adequate to avoid a reactor trip on high pressurizer pressure. The problem is the steam dump setpoints. The present setpoints have a 5°F deadband with a 12°F proportional band. The temperature error (i.e., T_{avg} minus T_{ref}) at the transient start is only about 1°F (one-half of full-load T_{avg} of 549.4°F minus no-load T_{avg} of 547°F). Therefore, a large primary-side heatup must occur to

* This uncertainty bounds the calculated uncertainty.

generate a large enough temperature error to result in adequate steam dump relief. A revision to the steam dump setpoints was done as follows:

• Deadband	(TC-412J)	3°F	(vs. 5°F presently)
• High (T_{avg} - T_{ref}) valve trip open	(TC-412K)	9°F	(vs. 12°F presently)
• High-high (T_{avg} - T_{ref}) valve trip open	(TC-412R)	15°F	(vs. 17°F presently)

This revision resulted in the peak pressurizer pressure that was arrested by the actuation of the pressurizer PORVs actuating at their setpoint of 2350 psia (2335 psig). Therefore, the installed pressurizer spray and PORV capacity is adequate to maintain the pressurizer pressure below the high pressurizer pressure reactor trip setpoint of 2362 psig for a 50-percent step load decrease transient from the 1.4-percent power uprate conditions with the above-noted steam dump setpoint revision.

6.3.2 Plant Operability

The design-basis operability transients for IP2 were based on the following limiting transients analyzed as part of the Stretch Rating Program (Reference 6-2) and re-analyzed for the 1.4-percent power uprate:

- Large step-load decrease with steam dump; limiting case is from 100-percent to 50-percent power. This is the same transient described for the sizing of the pressurizer PORVs except that now it is analyzed from conditions more representative of actual plant operation rather than the more conservative and generic basis used for sizing of the pressure relief components. The same acceptance criteria are applied here. This transient should not result in actuation of the high pressurizer pressure trip setpoint (actuated at 2362 psig). In addition, this is the limiting operational transient for actuation of the overtemperature ΔT (OTΔT) or overpower ΔT (OPΔT) trip setpoints. The plant transient response acceptability should be that the trip setpoint is not reached.

- 10-percent step-load increase from 90-percent power. The criterion here is that the minimum steam line pressure does not drop down to the low steam line pressure ESFAS actuation setpoint of 525 psig.

- 10-percent step-load decrease from 100-percent power. The criterion here is that the peak pressurizer pressure is less than the PORV setpoint of 2350 psia (2235 psig).

- Turbine trip without reactor trip below 35-percent power. For this transient, the criterion is that the pressurizer PORVs will not be challenged even assuming no credit is taken for the rod control system (i.e., control system is defeated or in manual).

The analysis of the above large step-load decrease and 10-percent step-load increase transients would bracket the results for the less limiting 5-percent-per-minute unit loading and unloading transients. All of these transients should be accommodated without generating a reactor trip or ESFAS actuation.

Large Step-Load Decrease with Steam Dump

The design-basis load rejection is a step-load reduction in the turbine load from 100-percent to 50-percent power. To ensure consistency with the current-day analysis methods, Westinghouse re-analyzed this transient. The following are the key assumptions in this analysis:

- The plant is initially at 100 percent of the 1.4-percent power uprate level.
- The plant T_{avg} is at Low T_{avg} condition (nominal value of 549.4°F).
- A nominal steam generator heat transfer coefficient (25-percent SGTP condition) is used.
- The transient is a step-load reduction from the noted 100-percent turbine load to 50-percent load.
- Fuel reactivities are at conservative BOL conditions.
- Credit is taken for automatic operation of all NSSS control systems (reactor control, pressurizer pressure and level control, feedwater control, and steam dump control).

An analysis using the minimum acceptable full-power steam pressure of 650 psia and with the turbine load being reduced at a maximum rate of 200-percent-per-minute (generic maximum turbine unloading rate) still resulted in a reactor trip being generated on OTΔT. The IP2 is presently operating at a full-power T_{avg} of 559°F with a consideration being made to increase this value to 562°F. An analysis assuming a full-power T_{avg} of 559°F with the turbine load being reduced at a maximum rate of 200-percent-per-minute (generic maximum turbine unloading rate) still resulted in a reactor trip being generated on OTΔT. The problem is similar to that noted in Section 6.3.1 on the pressurizer spray and PORV sizing; the present steam dump setpoints were not allowing for a large enough steam dump demand. Another analysis was performed with the recommended steam dump setpoints shown in Section 6.3.1. For this case, the 50-percent load rejection was able to be handled without generating a reactor trip on any function. The limiting function was the OTΔT reactor trip, which noted a minimum margin to trip of 1.7 percent of nominal ΔT. Therefore, the 50-percent load rejection can be accommodated for the 1.4-percent power uprate without challenging any of the reactor trip or engineered safeguards setpoints assuming the turbine-generator is reduced in load at a maximum rate of 200-percent-per-minute, full-power T_{avg} is no less than 559°F, and the revised steam dumps setpoints shown in Section 6.3.1 are implemented.

10-Percent Step-Load Increase

The design-basis transient is a step-load increase in the turbine load from 90-percent to 100-percent power. To ensure consistency with the current-day analysis methods, Westinghouse re-analyzed this transient. The following are the key assumptions in this analysis:

- The plant is initially at 89.4 percent of the 1.4-percent power uprate level; this assumes an initial plant power level of 90-percent power minus 0.6-percent power uncertainty.

- The plant T_{avg} is at Low T_{avg} condition (full-power nominal value of 549.4°F; corresponding steam pressure of 564 psia).

- A nominal steam generator heat transfer coefficient (25-percent SGTP condition) is used.

- The transient is a step-load increase from the noted 89.4-percent turbine load to 100-percent load.

- Fuel reactivities are conservative BOL conditions.

- Analyses were done both assuming reactor/rod control is in automatic and that automatic control rod withdrawal is defeated (Reference 6-3). Credit is taken for automatic operation of other NSSS control systems (pressurizer pressure and level control and feedwater control; steam dump control is not actuated for load increase transients).

The limiting setpoint is the high steam line flow ESFAS/steam line isolation actuation function. This uses a coincident logic of a high steam line flow coincident with either a low steam line pressure or a low T_{avg} measurement. The 10-percent step-load increase transient would not actuate the high steam line flow portion of the ESFAS signal. However, partial actuation of the low T_{avg} (setpoint of 541°F) or the low steam line pressure setpoint (525 psig or 540 psia) might occur depending on the transient response.

The results showed that if the automatic control rod withdrawal is defeated, actuation of the low steam line pressure or low T_{avg} portion of the ESFAS high steam line flow function is possible. The minimum steam line pressure reached was 504 psia (in comparison to the setpoint value of 540 psia) and the minimum T_{avg} reached was 535.7°F (in comparison to the setpoint value of 541°F). This was from the conservative Low T_{avg} operating point (full-power T_{avg} of 549.4°F with a full-power steam line pressure of 564 psia). A re-analysis based on conditions for a full-power steam line pressure of no less than 650 psia resulted in a minimal margin to the setpoints. The minimum steam line pressure reached was 585 psia (in comparison to the setpoint value of 540 psia) and the minimum T_{avg} reached was 541.1°F (in comparison to the setpoint value of 541°F). The conservatisms in the analysis (largely beginning-of-core-life conditions plus load change accomplished in a step fashion) make this an acceptable result. Therefore, the 10-percent step-load increase transient can be accommodated even with automatic control rod withdrawal being defeated so long as the minimum full-power steam line pressure is no less than 650 psia. The 650 psia minimum steam pressure case is the limiting case. Margin is improved as the full-power T_{avg} and corresponding steam pressure are increased.

If automatic control rod withdrawal was credited, a noticeable improvement in the results is noted. The minimum T_{avg} was raised about 8°F and the minimum steam line pressure was raised about 45 psi for both of the above operating conditions (full-power T_{avg} of 549.4°F or a full-power steam pressure of 650 psia). Therefore, if automatic control rod withdrawal was implemented then, the 10-percent step-load increase transient could be accommodated even from the Low T_{avg} condition of 549.4°F.

10-Percent Step Load Decrease

The design-basis transient is a step-load decrease in the turbine load from 100-percent to 90-percent power. This transient is the same transient as described in Section 6.3.1 for the pressurizer spray sizing. As described in Section 6.3.1, the peak pressure reached was 2330 psia, which is less than the pressurizer

PORV setpoint of 2350 psia (2335 psig). No reactor trip or engineered safeguards setpoints were reached. Therefore the 10 percent step-load decrease can be accommodated for the 1.4-percent power uprate without challenging any of the reactor trip or engineered safeguards setpoints.

Turbine Trip Without Reactor Trip Below 35-Percent Power

The design basis for this transient is to not challenge the pressurizer PORVs even accounting for any credible single failure of any control system. Analysis of this transient was performed to support the turbine trip without reactor trip initiated from 35-percent power. The limiting case chosen for analysis was similar to the "Case B" analyzed for the Stretch Rating Program (Reference 6-2), which initiated the transient from the Low T_{avg} case with no credit for control rod operation (the assumed single failure). The following are the key assumptions in this analysis:

- The plant is initially at 35.6 percent of the 1.4-percent power uprate level (35-percent power plus 0.6-percent uncertainty).
- The plant T_{avg} is at Low T_{avg} condition (nominal value of 549.4°F).
- A nominal steam generator heat transfer coefficient (25-percent SGTP condition) is used.
- The transient is a step-load reduction to 0-percent turbine load.
- Fuel reactivities are at conservative BOL conditions.
- No credit is taken for reactor/rod control. Credit is taken for automatic operation of other NSSS control systems (pressurizer pressure and level control, feedwater control, and steam dump control).

An initial analysis for the above noted full-power T_{avg} of 549.4°F resulted in the pressurizer PORV setpoint being reached for the turbine trip transient. An analysis for operating conditions based on a full-load steam pressure no less than 650 psia resulted in a peak pressurizer pressure of 2338 psia. This is below the pressurizer PORV setpoint of 2350 psia (2335 psig).

Based on these limiting analyses, all of the normal plant operability transients can be accommodated for the 1.4-percent power uprate conditions without challenging any reactor trip or ESFAS setpoints. This is for the plant operating condition of a full-power steam pressure no less than 650 psia (minimum value acceptable based on the Model 44F steam generator structural analysis). This also requires changes in the steam dump load control system setpoints as described below.

Other Considerations

The preceding sections addressed the standard operational areas of consideration performed by Westinghouse when reviewing a plant capability to accept an uprate. Generally, a 1.4-percent power uprate is considered to have a minor effect on plant operability. This is similar to the differences normally seen in steam or feedwater flows when comparing 1 loop to another during normal 100-percent plant operation. The following are reviews of certain additional areas.

NSSS Control Systems Setpoints

The 1.4-percent power uprate requires the following revisions to the pressurizer level control system setpoints:

- For full-power T_{avg} between 549.4 and 553°F: level program is constant at 37 percent of span.
- For full-power T_{avg} between 553 and 579.2°F: level program is 37 percent of span at no-load (547°F) and increases linearly by 1.218 percent/°F for every °F increase in full-power T_{avg} above 553°F (example - for a full-power T_{avg} of 579.2°F, full-power pressurizer level is 68.9 percent of span).

The 1.4-percent power uprate requires the following revisions to the steam dump control system setpoints:

• Deadband	(TC-412J)	3°F	(vs. 5°F presently)
• High (T_{avg} - T_{ref}) valve trip open	(TC-412K)	9°F	(vs. 12°F presently)
• High-high (T_{avg} - T_{ref}) valve trip open	(TC-412R)	15°F	(vs. 17°F presently)

6.3.3 Cold Overpressure Mitigation System

Cold Overpressure Mitigation System (COMS) events, also known as Low-Temperature Overpressure Protection System (LTOPS) events, could potentially occur during cold shutdown operation (RCS temperature less than about 350°F). The 1.4-percent power uprate does not change any plant condition for which COMS/LTOPS is impacted. For these events, the plant is in a shutdown condition so the power uprate does not impact the plant response for these events. The intent of COMS/LTOPS is to prevent violation of the reactor vessel 10 CFR 50 Appendix G pressure-temperature limits. The Appendix G limit is not changing due to the 1.4-percent power uprate. Therefore, there is no impact on COMS/LTOPS due to the 1.4-percent power uprate.

6.3.4 Conclusions

Based on this review, the IP2 1.4-percent power uprate is not expected to result in unacceptable plant operations. The existing pressurizer pressure control component sizing is acceptable for the 1.4-percent power uprate conditions. The plant operability transients (5-percent-per-minute loading/unloading, 10-percent step-load increase, large-load rejection, and turbine trip without reactor trip below 35-percent power) can be accommodated and meet the existing design-basis requirements for the plant operation with a minimum full-power T_{avg} of 559°F. For the more restrictive plant operation at a full-power steam pressure no less than the 650 psia minimum needed to meet the steam generator primary-to-secondary pressure differential, all of the above transients can be successfully accommodated except for the large-load rejection transient. Certain of the steam dump controller setpoints require revisions as noted in the "NSSS Control System Setpoints" subsection above.

6.4 SECTION 6 REFERENCES

6-1. WCAP-7907-P-A, "LOFTRAN Code Description," April 1984.

6-2 WCAP-12187, "Consolidated Edison Company of New York, Inc., Indian Point Unit 2, NSSS Stretch Rating - 3083.4 MWt Engineering Report," March 1989.

6-3 "Consolidated Edison Company, Indian Point Unit 2, Design Basis Document for the Reactor Coolant System/Steam Generator Blowdown System," December 30, 2000 (with PCNs through July 5, 2001).

7 NSSS COMPONENTS

7.1 REACTOR VESSEL STRUCTURAL EVALUATION

The Indian Point Unit 2 (IP2) reactor vessel has been evaluated for impact due to the 1.4-percent power uprate. The 1.4-percent uprate has no effect on the results in the current IP2 reactor vessel report (Reference 7-1), since there is no change to any of the design inputs that were previously considered in the reactor vessel evaluations to support the installation of Model 44F steam generators. The normal operating vessel outlet temperature (T_{hot}) range (583.0°F to 611.7°F) (Table 2-1) and normal operating vessel inlet temperature (T_{cold}) range (515.8°F to 546.7°F) (Table 2-1) remain within the bounds of the previous reactor vessel structural evaluations documented in Reference 7-1, Appendix A, and performed in support of the installation of Model 44F steam generators. There are no changes to any of the primary-side design transients that were considered for the Stretch Rating Program in 1988 (Reference 7-1, Appendix C). The reactor vessel loss-of-coolant accident (LOCA) loads that were evaluated as continuing to apply for the implementation of Model 44F steam generators remain applicable to the 1.4-percent power uprate. The previous reactor pressure vessel system seismic analysis is not changed due to the 1.4-percent power uprate, since neither the seismic response spectra nor the mass inputs for the equipment are changed. Therefore, the loading condition is bounding for the faulted condition blowdown (LOCA) or to safe shutdown earthquake (SSE) seismic loads previously considered, and the reactor vessel structural analysis (Reference 7-1, Appendix B) is not impacted. As a result, there are no changes to the maximum stress intensities, the maximum ranges of stress intensity, or the maximum cumulative fatigue usage factors that were previously reported in the IP2 reactor vessel stress report (Reference 7-1). The IP2 reactor vessel continues to satisfy the applicable requirements of Section III (Nuclear Vessels) of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME B&PV) Code, 1965 Edition (Reference 7-2), in accordance with the reactor vessel design requirements.

Table 7-1 contains the calculated maximum ranges of primary-plus-secondary stress intensity and maximum cumulative fatigue usage factors (Reference 7-1) that remain applicable to the IP2 reactor vessel for the 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program.

Table 7-1 Maximum Range of Stress Intensity and Fatigue Summary [1]			
Limiting Location	**Calculated Maximum Range of Stress Intensity, ksi**	**Allowable Range of Stress Intensity, ksi**	**Limiting Fatigue Usage Factor, ΣU**
Main Closure Flange Region			
1. Closure Head Flange	45.37 ksi	$3\ S_m = 80.1$ ksi	0.0107 < 1.0
2. Vessel Flange	52.14 ksi	$3\ S_m = 80.1$ ksi	0.0229 < 1.0
3. Closure Studs	109.40 ksi	$3\ S_m = 110.40$ ksi	0.9078 < 1.0
CRDM Housings	77.70 [2] ksi	$3\ S_m = 69.9$ ksi	0.01 < 1.0
Outlet Nozzles and Supports	49.39 ksi	$3\ S_m = 50.1$ ksi	0.259 < 1.0
Inlet Nozzles and Supports	45.50 ksi	$3\ S_m = 80.1$ ksi	0.042 < 1.0
Vessel Wall Transition	37.90 ksi	$3\ S_m = 80.1$ ksi	0.0029 < 1.0
Bottom Head to Shell Juncture	34.10 ksi	$3\ S_m = 80.1$ ksi	0.003 < 1.0
Core Support Pads	55.26 ksi	$3\ S_m = 69.9$ ksi	0.874 < 1.0
Bottom Head Instrumentation Tubes	55.50 ksi	$3\ S_m = 69.9$ ksi	0.14 < 1.0

Notes:

1. ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition.
2. A simplified elastic-plastic analysis was performed to justify exceeding the $3\ S_m$ limit.

7.2 REACTOR VESSEL INTEGRITY-NEUTRON IRRADIATION

Reactor vessel integrity is affected by any changes in plant parameters that affect neutron fluence levels or temperature/pressure transients. The neutron fluence projections resulting from the IP2 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program have been evaluated to determine the potential effect on reactor vessel integrity. Typically, such an evaluation is performed by direct comparison of the neutron fluence projections from the analyses of record to the uprated neutron fluence projections. However, prior to the IP2 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program, Westinghouse revised the current reactor vessel integrity analyses of record for IP2 (Reference 7-3). These revisions were initiated based on the need to extend the pressure-temperature (P-T) limit curves and to document the bases for Pressure-Temperature Limit Report (PTLR). The updated reactor vessel integrity evaluations used neutron fluence projections that correspond to 3216 MWt, and thus bound the 1.4-percent power uprate. Indian Point Unit 2 has already submitted the revised analyses and associated P-T limit curves and received Nuclear Regulatory Commission (NRC) approval (Reference 7-4).

As such, the evaluations for the 1.4-percent power uprate documented below, will build on the approved analyses in Reference 7-3. More specifically, that includes the following evaluations:

- 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 the IP2 reactor vessel.

- Assessment of the P-T limit curves to confirm they are based on vessel fluence projections that bound the 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program (References 7-3 and 7-4).

- Review of the RT_{PTS} values to determine if the effects of the uprated fluence projections resulted in an increase in RT_{PTS} for the beltline materials in the IP2 reactor vessels at 32 effective full-power years (EFPY), which bounds the end of license (EOL) (Reference 7-3).

- Review of the upper shelf energy (USE) values at 32 EFPY, which bound the EOL, for all reactor vessel beltline materials in the IP2 reactor vessels to assess the impact of the uprated fluence (Reference 7-3).

The calculated fluences used in the 1.4-percent power uprate evaluation comply with Regulatory Guide (RG) 1.190 (Reference 7-5). As these calculations are performed on a plant-specific basis, consistent with the methodology in RG 1.190.

7.2.1 Surveillance Capsule Withdrawal Schedule

The revised fluence projections for the 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program have been used in the assessment of the current withdrawal schedule for IP2. 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°F. These ΔRT_{NDT} values would 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.

7.2.2 Applicability of Heatup and Cooldown Pressure-Temperature Limit Curves

The IP2 Technical Specifications contain 25 EFPY P-T limit curves (also documented in Reference 7-3). These P-T limit curves were based on fluence values that bound the 1.4-percent power uprate. Therefore, the existing heatup and cooldown curves for 25 EFPY are acceptable for the power uprate without any necessary changes or reduction in EFPY.

7.2.3 Emergency Response Guideline Limits

The current peak inside surface RT_{NDT} value at 32 EFPY (bounding EOL) was calculated to be 246°F for IP2 (Reference 7-3). The limiting material for IP2 is the intermediate to lower shell girth weld. This RT_{NDT} value places IP2 in Emergency Response Guideline (ERG) Category II.

7.2.4 Pressurized Thermal Shock

All beltline materials are expected to have RT_{PTS} values less than 270°F for plates, forgings, and longitudinal welds, and 300°F for circumferential welds. The pressurized thermal shock (PTS) calculations were performed for IP2 using the latest procedures required by the NRC (Reference 7-7). Based on the evaluation of PTS, all RT_{PTS} values will remain below the NRC screening criteria values using calculated fluence projections that bound the 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program through 32 EFPY (bounding EOL) for IP2 as shown in Table 7-2.

Table 7-2
RT_{PTS} Calculations for Indian Point Unit 2 Beltline Region Materials at 32 EFPY with Bounding (3216 MWt) Uprated Fluences

Material	Fluence (n/cm^2, E>1.0 MeV)	FF	CF (°F)	ΔRT_{PTS}[a] (°F)	Margin (°F)	$RT_{NDT(U)}$[b] (°F)	RT_{PTS}[c] (°F)
Intermediate Shell Plate B-2002-1	1.28×10^{19}	1.07	144	154.1	34	34	222
- Using Surveillance Capsule (S/C) Data	1.28×10^{19}	1.07	114	122.0	17	34	173
Intermediate Shell Plate B-2002-2	1.28×10^{19}	1.07	115.1	123.2	34	21	178
- Using S/C Data	1.28×10^{19}	1.07	118.2	126.5	34	21	182
Intermediate Shell Plate B-2002-3	1.28×10^{19}	1.07	176	188.3	34	21	243
- Using S/C Data	1.28×10^{19}	1.07	181.9	194.6	17	21	233
Lower Shell Plate B-2003-1	1.28×10^{19}	1.07	152	162.6	34	20	217
Lower Shell Plate B-2003-2	1.28×10^{19}	1.07	142	151.9	34	-20	166
Intermediate & Lower Shell Longitudinal Welds (Heat # W5214)	8.55×10^{18}	0.956	230.2	220.1	65.5	-56	230
- Using S/C Data	8.55×10^{18}	0.956	254.7	243.5	44.0	-56	232
Intermediate to Lower Shell Girth Weld (Heat # 34B009)	1.28×10^{19}	1.07	220.9	236.4	65.5	-56	246

Notes:

a. $\Delta RT_{PTS} = CF * FF$

b. Initial RT_{NDT} values are measured values

c. $RT_{PTS} = RT_{NDT(U)} + \Delta RT_{PTS} + \text{Margin (°F)}$

CF: Chemistry Factor

FF: Fluence Factor

7.2.5 Upper Shelf Energy

All beltline materials have a USE greater than 50 ft-lb through the EOL (32 EFPY) as required by the Code of Federal Regulations (CFR) 10 CFR 50, Appendix G (Reference 7-8). The 32 EFPY (bounding EOL) USE was predicted using the EOL 1/4 thickness (1/4T) fluence projections that bound the 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program. The predicted USE values for IP2 have been determined based on bounding fluence values as shown in Table 7-3 and documented in Reference 7-3.

7.2.6 Conclusions

The updated fluence projections that bound the 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program (Reference 7-3), while considering actual power distributions incorporated to date, have already been incorporated into the reactor vessel integrity analyses of record (P-T limit curves, ERG category, PTS, and USE). In addition, the uprate does not change the number of capsules required to be withdrawn from the IP2 reactor vessel. Therefore, there is no impact related to the 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program on reactor vessel integrity.

Table 7-3
Predicted 32 EFPY USE Calculations for all the Beltline Region Materials with Bounding (3216 MWt) Uprated Fluences

Material	Weight % of Cu	1/4T EOL Fluence (10^{19} n/cm^2)	Unirradiated USE [a] (ft-lb)	Projected USE Decrease (%)	Projected EOL USE (ft-lb)
Intermediate Shell Plate B-2002-1	0.19	0.763	70	20	56
Intermediate Shell Plate B-2002-2	0.17	0.763	73	21	58
Intermediate Shell Plate B-2002-3	0.25	0.763	74	32	50.3
Lower Shell Plate B-2003-1	0.20	0.763	71	27	52
Lower Shell Plate B-2003-2	0.19	0.763	88	27	64
Intermediate & Lower Shell Longitudinal Welds (Heat # W5214)	0.21	0.510	121	43	69
Intermediate to Lower Shell Girth Weld (Heat # 34B009)	0.19	0.763	82 [b]	32	56

Notes:

a. These values were obtained from original test reports. Values reported in the NRC Database RVID2 are identical with exception to Intermediate Shell Plates B-2002-1, 2. RVID2 reported the initial USE as 76 and 75. This evaluation conservatively used the lower values of 70 and 73.

b. Value was obtained from the average of three impacts tests (71, 84, 90) at 10°F performed for the original material certification.

7.3 REACTOR INTERNALS

The reactor internals support the fuel and control rod assemblies, absorb control rod assembly dynamic loads, and transmit these and other loads to the reactor vessel. The internals also direct flow through the fuel assemblies, provide adequate cooling to various internals structures, and support in-core instrumentation. The changes in the Reactor Coolant System (RCS) temperatures produce changes in the boundary conditions experienced by the reactor internals components. Also, increases in core power may increase nuclear heating rates in the lower core plate, upper core plate, and baffle-barrel former region. This section describes the analyses performed to demonstrate that the reactor internals can perform their intended design functions at the 1.4-percent power uprate conditions.

7.3.1 Thermal-Hydraulic Systems Evaluations

A key area in evaluation of core performance is the determination of the hydraulic behavior of the coolant flow and its effect within the Reactor Internals System. The core bypass flow is defined as the total amount of reactor coolant flow that bypasses the core region, and is not considered effective in the core heat transfer process. Consequently, the effect of increasing core bypass flow is a reduction in core power capability. The rod cluster control assembly (RCCA) scram time is affected by the flow and temperature conditions. The hydraulic lift forces are critical in the assessment of the structural integrity of the reactor internals and hold-down spring functionality. Baffle plate gap momentum flux/fuel stability is affected by pressure differences between the core and baffle former region.

The results of these evaluations are discussed below.

Core Bypass Flow Calculation

Bypass flow is the total amount of reactor coolant flow bypassing the core region. The principal core bypass flows are the barrel-baffle region, vessel head spray nozzles, vessel outlet nozzle gap, baffle plate core cavity gap, and the fuel assembly thimble tubes.

The design core bypass flow limit is 6.5 percent of the total reactor vessel flow. The effect of the 1.4-percent power uprate has an insignificant effect on the core bypass flow. Therefore, the total design core bypass flow value of 6.5 percent remains unchanged.

RCCA Drop Time

An evaluation was performed to demonstrate that the RCCA drop time is still within the current value of 2.4 seconds (required by the Technical Specifications) for the revised design conditions. The evaluation included the effects of VANTAGE+ fuel with intermediate flow mixers (IFMs) and thimble plugs removed along with the 1.4-percent power uprate. The result of the evaluation confirmed that the current Technical Specification limit of 2.4 seconds remain applicable to the 1.4-percent power uprate.

Hydraulic Lift Forces and Pressure Losses

The reactor internals hold-down spring is essentially a large belleville-type spring of rectangular cross-section. The purpose of this spring is to maintain a net clamping force between the reactor vessel head flange and the upper internals flange and the reactor vessel shell flange and the core barrel flange of the internals. An evaluation was performed to determine the hydraulic lift forces on the various reactor internal components to ensure that the reactor internals assembly would remain seated and stable for all conditions. The results indicate that the downward force remains essentially unchanged, indicating that the reactor internals would remain seated and stable for the 1.4-percent power uprate conditions.

Baffle Joint Momentum Flux and Fuel Rod Stability

Baffle jetting is a hydraulically induced instability or vibration of fuel rods caused by a high-velocity jet of water. This jet is created by high-pressure water being forced through gaps between the baffle plates that surround the core. The baffle jetting phenomenon could lead to fuel cladding damage.

A number of experimental tests have been performed to study the interaction between baffle joint jetting and the response of the fuel rod. These tests indicated that there are two vibration levels that can result in fuel rod damage. Lower levels of vibration amplitude can inflict damage in the form of vibration wear at the rod/grid interface. Large amplitude vibration (whirling), caused by fluid elastic instability, can result in fuel rod damage due to cladding fatigue failure, rod-to-rod contact, or even rod-to-baffle-plate wall contact.

To preclude fuel rod failures from flow-induced vibration, the crossflow emanating from baffle joint gaps must be limited to a specific momentum flux, V^2h; that is, the product of the gap width, h, and the square of the baffle joint jet velocity, V^2. This momentum flux varies from point to point along the baffle plate due to changes in the pressure differential across the plate and the local gap width variations. In addition, the modal response of the vibrating fuel rod must be considered. That is, a large value of local momentum flux impinging near a grid is much less effective in causing vibration than the same V^2h impinging near the mid span of a fuel rod.

Baffle joint momentum flux is dependent upon the pressure differential across the baffle plate, the baffle-to-baffle gap width, and the modal response of the fuel assembly. Any increase in baffle joint momentum flux would require an increase in at least 1 of these. The pressure differential across the baffle plate and the baffle gap width and fuel assembly modal response remains unchanged due to the 1.4-percent power uprate. Therefore, the baffle joint momentum flux would not change as a result of the 1.4-percent power uprate.

7.3.2 Mechanical Evaluations

The 1.4-percent power uprate conditions do not affect the current design bases for seismic and LOCA loads. Therefore, it was not necessary to re-evaluate the structural effects from the seismic operating basis earthquake (OBE) and SSE loads and the LOCA hydraulic and dynamic loads.

Regarding flow-induced vibration, the vessel/core inlet coolant temperature decreases 1.0°F for the High T_{avg} case. For the High T_{avg} case, the vessel outlet coolant temperature remains unchanged. This temperature change causes a change in water density that has a negligible impact on the vibratory response of the reactor internals. The design power capability parameters for the current design basis and the 1.4-percent power uprate remain essentially the same. Therefore, it can be concluded that there is no significant impact on the performance of the reactor internals with regard to flow-induced vibration.

7.3.3 Structural Evaluations

Evaluations were performed to demonstrate that the 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 temperatures (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.

Lower Core Plate Structural Analysis

The lower core plate is a perforated circular plate that supports and positions the fuel assemblies. The plate contains numerous holes to allow fluid to flow through the plate. The fluid flow is provided to each fuel assembly and the baffle-barrel region.

Due to the lower core plate's proximity to the core, it is subjected to the effects of heat generation. The heat generation rates in the lower core plate due to gamma heating can cause a significant temperature increase in this component. A structural evaluation was performed to demonstrate that the structural integrity of the lower core plate is not adversely affected by the revised design conditions. The cumulative fatigue usage factor of the lower core plate (Table 7-4), including the effects of the increase in the heat generation rates, is small, and the lower core plate is structurally adequate for the 1.4-percent power uprate conditions.

Baffle-Barrel Region Evaluations

The baffle-barrel regions consist of a core barrel into which baffle plates are installed. They are supported by bolting interconnecting former plates that attach the baffle and core barrel.

The baffle-to-former bolts restrain the motion of the baffle plates that surround the core. These bolts are subjected to primary loads consisting of deadweight, hydraulic pressure differentials, LOCA and seismic loads, as well as secondary loads consisting of preload, and thermal loads resulting from RCS temperatures and gamma heating rates. The baffle-to-former bolt thermal loads are induced by differences in the average metal temperature between the core barrel and baffle plate. In addition to providing structural restraint, the baffles also channel and direct coolant flow such that a coolable core geometry can be maintained.

Table 7-4 Margins of Safety and Fatigue Summary [1]			
Limiting Component	**Calculated Stress [2], psi**	**Allowable Stress, psi**	**Limiting Fatigue Usage Factor, ΣU**
Lower Core Plate	47,000.	48,600.	0.42
Baffle/Barrel Assembly	See Note 3	See Note 3	See Note 3
Upper Core Plate	23,700.	48,600.	0.123

Notes:

1. ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1986 Edition.
2. Primary Plus Secondary Stress Intensity (NG-3222.2)
3. No new cumulative usage factor (CUF) calculations were performed for the baffle-barrel region components since it has been demonstrated that the 1.4-percent power uprate conditions are bounded by the original design operating conditions.

The thermally induced displacements of the baffle-former bolts for the 1.4-percent power uprate relative to the original design conditions were calculated for a bounding range of conditions. The results demonstrated that the 1.4-percent power uprate conditions have smaller thermally induced bolt displacement than the original design conditions. The reason the thermally induced bolt displacements are smaller is that the peripheral assembly power distributions are less severe (lower) for the 1.4-percent uprate conditions that those of the original design conditions. Therefore, the baffle-barrel region thermal and structural analysis results are still bounding for the revised design conditions associated with the 1.4-percent power uprate.

Upper Core Plate Structural Analysis

The upper core plate positions the upper ends of the fuel assemblies and the lower ends of the control rod guide tubes. It serves as the transitioning member for the control rods for entry and retraction from the fuel assemblies. It also controls coolant flow in its exit from the fuel assemblies and serves as a boundary between the core and the exit plenum. The upper core plate is restrained from vertical movement by the upper support columns, which are attached to the upper support plate assembly. The lateral movement is restrained by four equally spaced core plate alignment pins.

The maximum stress contributor in the upper core plate is the membrane stress resulting from the average temperature difference between the center portion of the upper core plate and the rim. The increased stress from the increased gamma heating was determined as a function of the heat generation rate increment. The fluid temperature effect due to the 1.4-percent uprate is small. The results show that the structural integrity of the upper core plate is maintained for the 1.4-percent power uprate conditions. The cumulative fatigue usage factor of the upper core plate caused by the increase in the heat generation rates remains less than 1.0 (Table 7-4).

7.4 PIPING AND SUPPORTS

7.4.1 Nuclear Steam Supply System Piping

The potential effect of the 1.4-percent power uprate on the IP2 existing reactor coolant loop (RCL) and pressurizer surge line analyses was evaluated. The parameters associated with the 1.4-percent power uprate were reviewed for potential effect on the existing analyses for the RCL piping and the Class 1 auxiliary lines evaluation.

The 1.4-percent power uprate has no significant effect on the RCL analyses. Three basic sets of input parameters are used in the evaluation of the RCL and the pressurizer surge line:

- Nuclear Steam Supply System (NSSS) design parameters (Table 2-1)
- NSSS thermal design transients (Section 5)
- LOCA hydraulic forcing functions

The thermal expansion analysis performed for the RCL and pressurizer surge line envelopes the design parameters as identified in Table 2-1 and, therefore, bounds the design parameters developed for the 1.4-percent power uprate. The current results for the RCL and the pressurizer surge line remain bounding and applicable.

The potential effect on design transients due to the changes in full-power temperatures for the 1.4-percent power uprate is addressed in Section 5 of this report. Based on the small changes in operating temperatures, it was concluded that the current NSSS design transients for the primary RCS, pressurizer, and the secondary side remain applicable for the component evaluation for the 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program. The existing transients remain valid for the uprated condition. Therefore, there is no effect on the current design basis analyses for the RCL and the pressurizer surge line. Note that the code of record for the RCL piping is the USA Standard (USAS) Code B31.1-1955. However, the design-basis analysis is performed to the requirements of the 1973 edition (Reference 7-9). Per this Code, a detailed fatigue evaluation is not required for the RCL piping. The pressurizer surge line is evaluated to the ASME B&PV Section III, Subsection NB, 1986 Code, and includes the effects of thermal stratification as discussed in NRC Bulletin 88-11 and a detailed fatigue evaluation.

The RCL design basis was also reviewed for the impact on the LOCA hydraulic forces. The uprate has negligible effect on the hydraulic forcing functions. Therefore, there is no impact on the RCL LOCA analysis for the 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program. Additionally, the RCL design basis also considered postulated pipe ruptures at the main steam and feedwater nozzles to the steam generators. The evaluations performed for these breaks also bound the 1.4-percent power uprate conditions.

Based on the evaluations of the NSSS design parameters, NSSS thermal design transients, and the LOCA hydraulic forcing functions, the current design-basis analyses for the IP2 RCL and pressurizer surge line (Reference 7-10) remain applicable for the IP2 1.4-percent power uprate conditions. Additionally, there are no changes to any of the steam generator or RCL displacements, the RCL leak-before-break (LBB) input loads, the RCL branch nozzle qualifications, the Class 1 auxiliary line piping systems, the primary equipment nozzle qualifications, or the magnitude of the primary equipment support loads. The maximum primary and secondary stresses, including maximum fatigue usage factors as appropriate, also remain applicable for the 1.4-percent power uprate.

7.4.2 Reactor Coolant Loop Support System

The RCL supports are designed to support the reactor coolant equipment and piping for normal operating, seismic, and postulated accident conditions. The support structures were evaluated to the American Institute of Steel Construction (AISC) Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings, 1963 Edition. The evaluation was performed for the revised loading associated with the removal of 4 of the original 6 steam generator support snubbers per loop, and installation of replacement Model 44F steam generators. The 2 lower snubbers and 1 of the 2 snubbers on each side at the top of the support structure were not required.

The 1.4-percent power uprate does not significantly affect any of the loads applied to the equipment supports by the primary equipment and piping. Therefore, the design basis of the supports as reconciled for the IP2 Snubber Reduction Program remains applicable for the 1.4-percent power uprate.

The RCL supports were shown to meet the allowable stresses for all loading combinations for the IP2 Snubber Reduction Program.

The steam generator and reactor coolant pump (RCP) supports have been qualified for piping and component loads resulting from the Snubber Reduction Program. Since the 1.4-percent power uprate does not significantly change the loads exerted upon the support structures, the supports remain qualified for the 1.4-percent power uprate conditions.

7.4.3 Leak-Before-Break Analysis

The current LBB evaluation was performed for the primary-loop piping to provide the technical justification for eliminating dynamic effects of pipe rupture as the structural design basis for IP2. The evaluation was documented in WCAP-10977 Revision 2 (Reference 7-11) and WCAP-10977 Supplement 1 (Reference 7-12). In addition, Westinghouse performed an LBB evaluation in November 2000 for the Replacement Steam Generator and the Steam Generator Snubber Reduction Program.

To demonstrate the elimination of RCS primary loop pipe breaks, the following objectives must be achieved:

- Demonstrate that margin exists between the "critical" crack size and a postulated crack that yields a detectable leak rate
- Demonstrate that there is sufficient margin between the leakage through a postulated crack and the leak detection capability

- Demonstrate margin on the applied load
- Demonstrate that fatigue crack growth is negligible

These objectives were met as discussed in References 7-11 and 7-12.

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.

7.5 CONTROL ROD DRIVE MECHANISMS

The design temperature and pressure used for the IP2 control rod drive mechanisms (CRDMs) are 650°F and 2500 psia. The RCS design parameters given in Table 2-1 indicate that the operating temperature and pressure for the 1.4-percent power uprate are 611.7°F (vessel outlet) and 2250 psia, respectively. The zero-load temperature given in Table 2-1 is 547°F. Since the operating pressure and temperature has not changed from that considered in the original design and the Stretch Rating Program (Reference 7-13), there is no impact on the Reference 7-13 analysis and it remains bounded for the 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program.

Therefore, the current analysis of record (Reference 7-13) remains bounded for the proposed 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program.

7.6 REACTOR COOLANT PUMPS AND MOTORS

7.6.1 Reactor Coolant Pump

The RCPs are located between the steam generator outlet and reactor vessel inlet in the reactor coolant loops. The reactor vessel inlet (RCP outlet) temperature at 100-percent power can range between 515.8°F and 546.7°F for the 1.4-percent power uprate conditions, as shown in Table 2-1. This temperature is lower than the full-power temperature of 555°F defined in the RCP equipment specification and, therefore, represents a less limiting condition. This reactor vessel inlet (RCP outlet) temperature range is also bounded by the temperature range (515.8°F to 547.7°F) defined for the RCP in the 1988 IP2 Stretch Rating Program. The operating pressure for the 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program remains 2250 psia, as it was in the RCP equipment specification and for the 1988 Stretch Rating Program.

The NSSS design transients previously defined for the 1988 IP2 Stretch Rating Program remain unchanged for the 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program.

The operating temperature, operating pressure, and NSSS design transients for the 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program are bounded by the values for the 1988 IP2 Stretch Rating Program. Therefore, the RCP structural evaluation performed for the 1988 IP2 Stretch Rating Program remains applicable and bounding for the 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program.

7.6.2 Reactor Coolant Pump Motor

The limiting design parameter of the RCP motor is the horsepower loading at continuous hot and cold operation. Bounding loads on the IP2 RCP motors were developed based on the 1988 IP2 Stretch Rating Program best-estimate flows (BEFs) and operating temperatures. A steam generator outlet temperature range of 515.5°F to 547.4°F and a single-loop flow of 85,800 gpm were considered. The results show a hot-loop motor load of 6100 hp and a cold-loop motor load of 7600 hp. The IP2 RCP motors have an original nameplate rating of 6000 hp hot and a design limit of 7500 hp cold. Evaluation of the motor loading, which is in excess of the original nameplate rating, showed that operation of the motors at these conditions is acceptable with the resistance ring modifications that were made.

The above bounding motor evaluation applicable to IP2 was based on a flow of 85,800 gpm per loop. The 1.4-percent power uprate BEF is 85,900 gpm for the 25-percent steam generator tube plugging (SGTP) case. In addition, the low steam generator outlet temperature of 515.5°F associated with the above motor evaluation is the same as that defined for the 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program. The RCP operates in the portion of its flow-head curve where an increase in flow corresponds to a decrease in power. Since the 1.4-percent power uprate BEF and the steam generator outlet temperature are bounded by the above evaluated flow and temperature, these motor loads remain bounding and applicable for the 1.4-percent power uprate conditions.

Based upon this evaluation, the IP2 RCP motor evaluation performed for the 1988 Stretch Rating Program is bounding for the 1.4-percent power uprate, and the IP2 RCP motors are acceptable for operation at the 1.4-percent power uprate conditions.

7.7 STEAM GENERATORS

Evaluations of the thermal-hydraulic performance, structural integrity, and mechanical hardware have been performed to address operation at a 1.4-percent power uprate conditions.

7.7.1 Thermal-Hydraulic Evaluation

The thermal-hydraulic evaluations of the IP2 Model 44F steam generator focused on the changes to secondary-side operating characteristics at the 1.4-percent power uprate conditions. The 1.4-percent power uprate design operating conditions considered are presented in Table 2-1 of this report. The evaluations discussed in this section were performed to confirm the acceptability of the steam generator secondary-side parameters. Six cases were analyzed; at two power levels, 100 and 101.4 percent, two RCS primary average temperatures (T_{avg}), 549°F and 579°F, and two SGTP levels, 0 and 25-percent. The results of the thermal-hydraulic evaluations are summarized in Table 7-5. Based on these evaluations, the IP2 steam generators are qualified to operate at the 1.4-percent power uprate conditions with up to 25-percent SGTP.

Table 7-5 Thermal-Hydraulic Characteristics of IP2 Steam Generators						
	RCS Avg. Temp. (T_{avg})=549°F			RCS Avg. Temp. (T_{avg})=579°F		
Parameter	Pre-Uprate	Table 2-1 Case 1	Table 2-1 Case 2	Pre-Uprate	Table 2-1 Case 3	Table 2-1 Case 4
Power, %	100.0	101.4	101.4	100.0	101.4	101.4
Reactor Power, MWt	3071.4	3115.0	3115.0	3071.4	3115.0	3115.0
NSSS Power, MWt	3083.4	3127.0	3127.0	3083.4	3127.0	3127.0
Power per SG, MWt	770.85	781.75	781.75	770.85	781.75	781.75
SG Primary Flow per Loop, gpm	80,700	80,700	80,700	80,700	80,700	80,700
SG Primary Fluid Pressure, psia	2250	2250	2250	2250	2250	2250
SG Primary Outlet Temperature, °F	515.5	515.5	515.5	547.4	546.4	546.4
Feedwater Temperature, °F	430.0	431.8	431.8	430.0	431.8	431.8
Water Level Above Tubesheet, in.	459	459	459	459	459	459
Blowdown Flow Rate, lb/hr	33,150	33,150	33,150	33,150	33,150	33,150
Tube Plugging, % of Total Tubes	0	0	25	0	0	25
Fouling Factor, hr-ft^2- °F/Btu	0 00006	0.00006	0.00006	0.00006	0.00006	0.00006
Thermal Hydraulic Characteristics						
Steam Flow Rate per SG, 10^6 lb/hr	3.313	3.368	3.365	3.333	3.387	3.379
Steam Pressure at Nozzle, psia	625.8	623.9	563.2	842.9	833.1	759.6
Circulation Ratio	3.28	3.21	3.18	3.33	3.27	3.26
Separator Parameter	9.7118	10.0518	10.4659	8.6668	8.9984	9.3161
Moisture Carryover, %	0.0056	0.0071	0.0096	0.0027	0.0034	0.0042
Downcomer Fluid Velocity, ft/sec	10.74	10.71	10.50	11.24	11.22	11.10
Sec. Side Pressure Drop, psi	30.44	31.45	32.30	28.62	29.62	30.12
Sec. Side Liquid Mass, lb	76,587	76,106	74,328	80,818	80,284	79,144
Sec. Side Liquid Volume, ft^3	1533.5	1523.6	1474.8	1667.6	1654.4	1614.9
Sec. Side Vapor Mass, lb	4,444	4,446	4,079	5,782	5,737	5,276
Sec. Side Vapor Volume, ft^3	3193.9	3203.8	3252.6	3059.8	3073.0	3112.5
Total Secondary Fluid Mass, lb	81,031	80,552	78,407	86,600	86,021	84,420
Sec. Fluid Heat Content, 10^6 Btu	41.05	40.81	38.78	47.25	46.82	44.88
Average Heat Flux, Btu/hr-ft^2	60,511	61,367	81,822	60,511	61,367	81,823
Damping Factor, 1/hr	-482.5	-487.4	-522.1	-419.2	-422.8	-440.3

Table 7-5 Thermal-Hydraulic Characteristics of IP2 Steam Generators (cont.)						
Parameter	**RCS Avg. Temp. (T_{avg})=549°F**			**RCS Avg. Temp. (T_{avg})=579°F**		
	Pre-Uprate	**Table 2-1 Case 1**	**Table 2-1 Case 2**	**Pre-Uprate**	**Table 2-1 Case 3**	**Table 2-1 Case 4**
Prim. Fluid Heat Content, 10^6 Btu	23.67	23.68	19.45	24.23	24.22	19.89
SG Prim. Inlet (T_{hot}) Temperature, °F	582.2	583.1	583.1	611.8	611.7	611.7
SG Prim. Average Temperature, °F	548.8	549.3	549.3	579.6	579.0	579.0
Maximum (X/X_{DNB})	N/A	N/A	N/A	0.7180	0.7504	1.0084

Bundle Mixture Flow Rate

The steam flow rate increases proportionally with the 1.4-percent power uprate when operating with the same T_{avg} and feedwater temperature. With a 1.4-percent power uprate, the calculated steam flow rate per generator increases from 3.31 to 3.37 million lb/hr and the circulation ratio decreases from 3.28 to 3.21. Since the tube bundle mixture flow rate is the product of the circulation ratio and the steam flow rate, the resulting bundle flow rate is approximately 11 million lb/hr for all cases, or essentially the same at both 100- and 101.4-percent power.

The secondary fluid velocities in the U-bend region, at the uprate conditions, are approximately 2 percent without SGTP and 12 percent with 25-percent SGTP. The fluid velocities in the downcomer and at the wrapper opening are up to 2-percent lower at power uprate conditions with 25-percent SGTP. The 1.4-percent uprate and the small changes in T_{hot} and feedwater temperatures essentially have no effect on the secondary flow both in the downcomer and tube bundle.

Steam Pressure

Steam pressure is affected by both the available heat transfer area in the tube bundle and the average primary fluid temperature. Assuming the 1.4-percent power uprate conditions and a T_{avg} of 579°F, Westinghouse used the GENF code to perform a more rigorous thermal-hydraulic analysis. This analysis resulted in a calculated steam pressure decrease from 842.9 psia to 833.1 psia, which is in the acceptable range.

GENF is a one-dimensional steady-state thermal and hydraulic performance code developed by Westinghouse specifically for feed-ring steam generators. The code has been verified and is maintained under Westinghouse Configuration Control.

GENF calculates the overall primary-side heat balance based on the thermal power, primary flow rate, and the primary outlet temperature and operating pressure. On the secondary side, the code determines the secondary-side saturation pressure in the tube bundle using an iterative procedure. The steam outlet pressure is then calculated by subtracting all losses from the bundle region to the steam nozzle outlet.

The steam outlet pressure is used to determine steam flow rate via the secondary-side heat balance and feedwater inlet temperature.

An iterative calculation is performed to determine the circulation ratio and various secondary-side pressure drops. Finally, the fluid masses and volumes, and the stability damping factor are calculated. The stability-damping factor is a measure of stable operation of the steam generator.

Heat Flux

The average heat flux in the steam generator is directly proportional to the heat load and inversely proportional to the heat transfer area in service. For the 0-percent SGTP case, the average heat flux increases from 60,511 Btu/hr-ft^2 at 100-percent power to 61,367 Btu/hr-ft^2 at 101.4-percent power. With 25-percent SGTP and 101.4-percent power uprate conditions, the average heat flux increases to 81,822 Btu/hr-ft^2.

A measure of the margin for departure from nucleate boiling (DNB) transition in the bundle is a verification of the ratio of the local quality, to the estimated quality at DNB transition, or (X/X_{DNB}). The ATHOS analyses show that the maximum (X/X_{DNB}) increases from 0.718 at 100-percent power to 0.750 at 101.4-percent power with 0-percent SGTP, indicating a minimal impact due to the 1.4-percent power uprate. If the SGTP level is increased to 25 percent with the 1.4-percent power uprate, the maximum (X/X_{DNB}) ratio is 1.008 in a small area in the U-bend region of the bundle. This case was analyzed with the worst-case scenario where all the plugged tubes are concentrated around the bundle outer periphery. Normally, in an operating unit, the plugged tubes would be randomly distributed and the maximum (X/X_{DNB}) ratio would not approach 1.0 as calculated for this case. Therefore, tube wall dry out is not expected to occur. Hence, the 25-percent SGTP level would represent the limiting case for this 1.4-percent power uprate.

ATHOS is a three-dimensional computer program for computational fluid dynamics analysis of steam generators. The ATHOS code was developed under the sponsorship of the Electric Power Research Institute (EPRI). The ATHOS code consists of a geometry pre-processor, a thermal-hydraulic (ATHOS) solver, and a post-processor module. The geometry pre-processor simulates the detailed geometry. This geometry simulation includes the detailed tube layout, tube lane blocks, flow distribution baffle, tube support plates, anti-vibration bars (AVB), and opening of the primary separators. The ATHOS module utilizes the pre-processor data to calculate the primary- and secondary-side thermal-hydraulic parameters in the steam generator. The ATHOS code calculates both the heat flux and tube wall temperature, in addition to typical parameters such as liquid velocity, vapor velocity, and steam quality for a two-phase flow like that in the secondary side of a steam generator.

The ATHOS code for the analysis of steam generators has been verified and qualified by EPRI and Westinghouse. The Westinghouse developed post-processors process the large amount of output from the ATHOS calculation. Their capabilities include: (1) velocity vector plots, and (2) contour plots of thermal-hydraulic parameters, such as steam quality, velocity, heat flux and critical steam quality corresponding to DNB.

Moisture Carryover

The field tests for moisture carryover (MCO) of Model 44F steam generators have been performed at several plants with the same modular separator package as the IP2 steam generators. The operating parameters, which can have an effect on moisture performance, are steam flow (power), steam pressure, and water level. The MCO values for the IP2 1.4-percent power uprate conditions are calculated from the GENF results. The calculated MCO increases from 0.0056-percent of steam flow at 100-percent power to a maximum of 0.0096-percent at 101.4-percent power uprate and 25-percent SGTP conditions. The MCO will be well below the 0.25-percent limit at the 1.4-percent uprate condition. This demonstrates that the 1.4-percent power uprate will have a negligible effect on the moisture separator performance of the steam generators.

Hydrodynamic Stability

The hydrodynamic stability of a steam generator is characterized by its damping factor. A negative value of the damping factor indicates that any disturbance to the thermal-hydraulic parameters (e.g., flow rate or water level) will rapidly reduce in amplitude, and the steam generator will return to stable operation. The damping factor decreases from -482.5 hr^{-1} at nominal power to -487.4 hr^{-1} at the 1.4-percent uprate conditions. Therefore, the IP2 steam generators will continue to operate in a hydrodynamically stable manner at the 101.4-percent uprate power operating conditions.

Steam Generator Secondary-Side Fluid Inventory

Secondary-side fluid inventory consists of the mass of liquid and the vapor phases. The vapor mass is approximately 5 percent of the total inventory. With the proposed 1.4-percent power uprate, the secondary fluid mass decreases from 81,031 lbs to 80,552 lbs, a change of less than 1 percent. The minimum calculated inventory of 78,407 lbs would occur for the 101.4-percent power uprate case with 25-percent SGTP. The small changes in inventory are judged to have no effect on steam generator operation.

Steam Generator Secondary-Side Pressure Drop

The secondary-side pressure drop increases from 30.44 psi to 31.45 psi as result of the 1.4-percent power uprate. It further increases to 32.30 psi with 25-percent SGTP and 101.4-percent power uprate operating conditions. The small increase in pressure drop should have no significant effect on the feedwater system operation.

In conclusion, the thermal-hydraulic characteristics of the IP2 Model 44F steam generators are within acceptable ranges for the 1.4-percent power uprate conditions with an SGTP level up to 25 percent.

7.7.2 Structural Integrity Evaluation

The structural evaluation for the 1.4-percent power uprate conditions focused on the critical steam generator components. The critical components are those that are affected by changes in the pressure and temperature in the primary and secondary sides of the steam generator. The following discussions address the evaluation of the primary-side and secondary-side components.

Comparisons of the primary-side transients and RCS parameters were performed to determine the scale factors that would be applied to the baseline analyses maximum stress ranges and fatigue usage factors. The baseline analysis results for various components were then scaled to represent operation at the 1.4-percent power uprate conditions.

The primary-side critical components evaluated included the divider plate, tubesheet and shell junction, tube-to-tubesheet weld, and the tubes.

The secondary-side critical components considered were the feedwater nozzle, secondary-side manway studs, and the steam nozzle.

Input Parameters and Assumptions

The 1.4-percent power uprate structural evaluation was performed for 3127 NSSS power and 25-percent SGTP. The applicable NSSS design parameters used for the steam generator structural evaluation are shown in Table 2-1. The applicable design transients for the 1.4-percent uprated power condition are discussed in Section 5.1 of this report. A primary-to-secondary differential pressure (ΔP) evaluation was performed to evaluate the change in ΔP, and in turn generate scaling factors that were applied to the original stress reports results. The scaling factors were based on the steam temperature of 494.9°F, corresponding to a steam pressure of 650 psi.

For the 1.4-percent power uprate conditions, the vessel outlet temperature, T_{hot}, is estimated to be a minimum of 583°F (Table 2-1). Prior to the 1.4-percent power uprate, T_{hot} was 582.2°F, the temperature used for the current design-basis analysis. Since the increase in temperature is less than 1°F, the change in steam pressure is small, and the scale factor based on thermal stresses is very small.

For the feedwater nozzle, the temperature gradient ($T_{stm} - T_{fw}$) at the 1.4-percent power uprate conditions is less than that at the reference condition. The decrease in temperature gradient (ΔT) will minimize the effects of the thermal transients across the feedwater nozzle. The decreased gradient will result in reduced thermal stresses. Therefore, the thermal stresses can be considered unchanged as a result of the 1.4-percent power uprate.

Description of Steam Generator Component Structural Analyses and Evaluations

Comparisons of the primary-side transients and RCS parameters were performed to determine the scale factors that would be applied to the baseline analyses to calculate maximum stress ranges. The results were then used to calculate fatigue usage factors applicable to the 1.4-percent power uprate conditions.

For the primary-side components (particularly the divider plate, the tubesheet and shell junctions, the tube-to-tubesheet weld, and tubes), the applicable scale factors were the ratios of the primary-to-secondary-side ΔP for the baseline to that for the 1.4-percent power uprate conditions. For the baseline condition, the ΔP was based on a steam pressure of 768 psi. The maximum primary-to-secondary pressure differential occurs with the minimum steam pressure. For the 1.4-percent power uprate condition, the minimum steam pressure was determined to be 650 psi, which occurs for the Low T_{avg} operating condition, with 25-percent SGTP. The calculated scale factors were applied to the stresses and fatigue usage factors for all applicable transient conditions that were involved in the design-basis evaluation.

For the secondary-side components, such as the feedwater nozzle and secondary-side manway studs, the decrease in secondary-side pressure was the basis for determining the applicable scale factors. The scale factors were then applied to the lower bound stresses, which, in turn, conservatively increased the stress ranges involving transients that originate from, or lead to, full power. The increased stress ranges were addressed in the evaluation of the secondary-side components and factored into the calculation of fatigue usage.

Acceptance Criteria

The acceptance criteria for each component is consistent with the criteria used in the design-basis analysis referenced for that component, as reported in the original stress reports. The maximum range of primary-plus-secondary stresses was compared with the corresponding $3S_m$ limits of the ASME B&PV Code. For situations where these limits were exceeded, a simplified elastic-plastic analysis was performed per NB 3228.3 (Reference 7-14) consistent with the original design-basis analysis.

A cumulative fatigue usage factor of less than or equal to 1.0 demonstrates adequacy for a 40-year design life.

Results and Conclusions

The results of the evaluation show that all components analyzed meet ASME Code Section III limits (Reference 7-14). The results of the evaluation are summarized in Table 7-6.

7.7.3 Evaluation of Primary-to-Secondary-Side Pressure Differential

An analysis was performed to determine if the ASME B&PV Code (1965 Edition through Summer 1966 Addenda) (Reference 7-14) limits on the Model 44F replacement steam generator design primary-to-secondary ΔP are exceeded for any of the applicable transient conditions, for the 1.4-percent power uprate parameters. The design pressure limit for the primary-to-secondary pressure differential is 1700 psi as defined in the applicable design specification.

The normal/upset transient conditions are subject to the following design pressure requirements:

- Normal condition transients: Primary-to-secondary pressure gradient shall be less than the design limit of 1700 psi.

- Upset condition transients: If the pressure during an upset transient exceeds the design pressure limit, the stress limits corresponding to design conditions apply using an allowable stress intensity value of 110 percent of those defined for design conditions. In other words, as long as the upset condition pressure values are less than 110 percent of the design pressure values, no additional analysis is necessary. For the IP2 steam generators, 110 percent of the design pressure limit corresponds to 1870 psi.

The primary-to-secondary pressure differential evaluation was based on the transient parameters discussed in Section 5.1 and the corresponding full-power conditions that are defined in Table 2-1. The pressure differentials across the primary-to-secondary-side pressure boundary are calculated for these defined full-power conditions. Note that the evaluation was performed for the 25-percent SGTP condition, since increased levels of plugging result in greater primary-to-secondary-side pressure differentials.

The analysis determined that the maximum normal/upset operating condition primary-to-secondary-side differential pressures for High T_{avg} operation would be 1551 psi for normal operating condition transients, and 1642 psi for upset condition transients. For the Low T_{avg} operating conditions, the maximum pressure differentials are 1664 psi and 1600 psi for the normal and upset conditions, respectively. The results show that the maximum primary-to-secondary pressure gradients are less than the allowable values of 1700 psi and 1870 psi for normal and upset operating conditions, respectively. Therefore, the design pressure requirements of the American Society of Mechanical Engineers (ASME) Code continue to be satisfied.

7.7.4 Evaluations for Repair Hardware

The IP2 replacement steam generators were placed in service in 2000. During the fabrication on one of the steam generators, several Westinghouse shop-welded plugs were installed. These components were re-evaluated for the operating conditions and transients associated with 1.4-percent uprated power operation.

In anticipation of future needs, both "long" and "short" 7/8-inch ribbed mechanical plugs were qualified for installation in the Model 44F replacement steam generators for the 1.4-percent power uprate operating conditions. In addition, since there are circumstances that may require tube ends to be reamed, a 40-percent tube wall undercut was considered. The resulting reduced tube-mouth weld joint geometry is qualified for continued service. Also, if a future need arises that a steam generator tube may require stabilization, an evaluation to qualify a collar-cable tube stabilizer was performed.

Mechanical Plugs

The bounding condition for the Westinghouse mechanical plug (Alloy 690 plug shell material) is the one that results in the largest pressure differential between the primary and the secondary sides of the steam generator. Both the NSSS design parameter changes and the original NSSS design transients were used to determine the effect of the 1.4-percent power uprate on the mechanical plugs. The most critical set of parameters for the mechanical plug evaluation is the primary-side hydrostatic pressure test in which the ΔP across the plug is 3107 psi and is independent of a 1.4-percent power uprate.

<u>Description of Evaluation</u>

A structural evaluation was performed for both "long" and "short" Westinghouse 7/8-inch ribbed mechanical plugs for the 1.4-percent uprate condition. This evaluation was performed to the applicable requirements of ASME B&PV Code (Reference 7-14).

<u>Acceptance Criteria</u>

The Westinghouse mechanical tube plug was evaluated for the original NSSS design transients and for the changes to these transients due to the 1.4-percent power uprate. The primary stresses due to design, normal, abnormal, and test conditions must remain within the respective ASME B&PV Code allowable values (Reference 7-14). The maximum range of primary-to-secondary stresses is limited to $3S_m$. The

Table 7-6
IP2 1.4% Power Uprate Evaluation Summary
Primary-and-Secondary-Side Components

Component	Load Condition	Stress Category	Stress (ksi)/ Fatigue - Baseline	Stress (ksi)/ Fatigue - Uprate	Allow (ksi/ Fatigue	Comments
Primary-Side Components						
Divider Plate	Normal/ Upset	Pm+Pb+Q [1]	notes	notes	69.90	Plastic analysis performed
		Fatigue	0.664	0.786	1.00	
Tubesheet & Shell Junction	Normal/ Upset	Pm+Pb+Q	88.7 [1]	notes	80.10	
		Fatigue	0.356	0 482	1.00	
Tube to Tubesheet weld [2]	Normal/ Upset	Pm+Pb+Q [2]	notes	notes	69.90	Reference analysis used higher factors with elastic stresses
		Fatigue	0.072	0.086	1.00	
Tubes	Normal/ Upset	Pm+Pb+Q	59.77	59.77 [3]	79.8	
		Fatigue	0.142	0.191	1.00	
Secondary-Side Components [4]						
Main feed	Normal/ Upset	Pm+Pb+Q	76.02	76.11	80.10	
water nozzle		Fatigue	0.994	1.000	1.00	
Secondary	Normal/ Upset	Pm+Pb+Q	80.20	80.79	94.5 (2.7Sm)	
manway stud		Fatigue	0.665	0.710	1.00	
Steam Nozzle 3	Normal/ Upset					
Sec A-A		Pm+Pb+Q [5]	57.96	58.14	80.10	
Insert section 2 [4]		P_m+P_b+Q [1]	62.97 [1]	63.346 [1]	56.07	Simplified plastic analysis completed
Support ring (Sec A-A) [5]		P_m+P_b+Q [1]	46.527 [1]	46.64 [1]	36.50	Simplified plastic analysis completed
		Fatigue (A-A) inside	0.020	0.024	1.00	
		Nozzle insert	0.865	0.870	1.00	
		Support ring (Sec A-A) [5]	0.190	0.192	1.00	

Table 7-6 IP2 1.4% Uprate Evaluation Summary Primary And Secondary Side Components (cont.)
Notes: 1. Exceeds $3S_m$ simplified plastic analysis was done in the reference analysis for fatigue evaluation. 2. Conservative high fatigue strength reduction factors are used with elastic stresses in the fatigue evaluation since primary stresses exceed $3S_m$. 3. Values not affected by uprate. 4. Additional stress due to reduction of pressure is taken to calculate the increase in stress range for secondary-side components. 5. Steam Nozzle (Sec A-A) - Additional Pressure Stress: (10/1357)x24.87=0.183 ksi Steam Nozzle (Insert) - Additional Pressure Stress: (10/1357)x 50.27=0.464 ksi Steam Nozzle (Support Ring) - Additional Pressure Stress: (10/1357)x15.706=0.115 ksi

cumulative fatigue usage factor must be less than or equal to 1.0, or the ASME fatigue exemption rules must apply, for a 40-year fatigue life for the plug. In addition to the stress criteria, plug retention must be ensured.

Results

The critical loading parameter from the design of the plug shell is the primary pressure. The plug qualifying calculation was based on a primary pressure of 2485 psig. The maximum design primary-to-secondary ΔP of 1700 psi for plug retention was also considered.

All stress/allowable ratios are less than unity, indicating that all primary stress limits are satisfied for the plug shell wall between the top land and the plug end cap. The plug meets the Class 1 fatigue exemption requirements per N- 415.1 of the ASME Code (Reference 7-14).

Since this is a component that is installed into the steam generator after original fabrication is complete, and since this part is typically fabricated to the requirements of the 1989 ASME Code Edition (Reference 7-15), an evaluation was conducted based on the 1989 ASME Code requirements. It was determined that the mechanical plug is also acceptable for the 1.4-percent power uprate operating conditions based on the 1989 ASME Code edition.

Conclusions

Results of the analyses performed for the mechanical plug for IP2 steam generators show that both the long- and short-mechanical plug designs satisfy all applicable stress and retention acceptance criteria at the 1.4-percent power uprate conditions.

Shop-Weld Plugs

The Westinghouse shop-weld plugs are fabricated from ASME SB-166, Alloy 600 rod material. The minimum yield for this material is 35,000 psi.

Description of Evaluation

A structural evaluation was performed for the existing shop-weld tube plugs for the 1.4-percent power uprate operating conditions and the applicable design transients, including those revised for the 1.4-percent power uprate conditions (Section 5.1). The evaluation was performed to the applicable requirements of the ASME B&PV Code (Reference 7-14).

Acceptance Criteria

The primary stresses due to design, normal, abnormal, and test conditions must remain within the respective ASME Code allowable values (Reference 7-14). The maximum primary-to-secondary stresses are limited to $3S_m$. The cumulative fatigue usage factor must be less than or equal to 1.0, or the ASME fatigue exemption rules must apply, for a 40-year fatigue life for the plug.

Results

The evaluation of the weld plug first addressed the design condition. A vertical minimum weld thickness critical plane around the perimeter (circumference) of the weld plug was considered. The design pressure differential of 1700 psi between the primary and secondary was applied to the plug.

Test conditions for the primary hydrostatic and secondary hydrostatic tests were then evaluated. Values for primary stresses, primary stresses plus secondary stresses, and primary-to-secondary stress range intensities were calculated. All stress values were found to be acceptable.

The normal/abnormal conditions were then reviewed. It was found that the controlling transient for both the normal and abnormal conditions was the steady-state fluctuation transient. The differential pressure considered was 1664 psi. This was the controlling pressure condition for the 1.4-percent power uprate transient conditions. It was found that the stress limits are acceptable for the controlling ΔP.

The last step in the evaluation process considered fatigue. The approach was to investigate if the weld plug would be exempt from an explicit usage factor calculation based on the ASME requirements for fatigue exemption. The 6 required fatigue exemption conditions were found to be satisfied. Therefore, it was concluded that the welded plug meets the ASME Code cycle load fatigue limits for the 1.4-percent uprate.

Conclusions

All primary stresses are satisfied for the weld between the weld plug and the tube-sheet cladding. The overall maximum primary-plus-secondary stresses for the bounding transient case of steady-state fluctuation was found to be acceptable. The fatigue evaluation for the weld plug utilized the ASME fatigue exemption rules. It was found that the ASME fatigue exemption rules were met. Therefore, fatigue conditions are acceptable.

Tube Undercut Qualification

The field machining of steam generator tube mouth ends may be required to implement modifications and the repair of tubes (i.e., plugging, sleeving, and tube end reopening). It is sometimes necessary to remove a portion of the tube and weld material utilizing a machining process (drilling and reaming) when removal of a Westinghouse mechanical plug is required. The structural evaluation performed for the 1.4-percent power uprate conditions addressed the acceptability of up to a 0.020-inch undercut of the tube wall thickness (40 percent of the 0.050-inch tube wall). The evaluation was performed to the applicable requirements of ASME B&PV Code (Reference 7-14).

Description of Evaluation

Past structural evaluations for steam generator tube-end machining have been performed for various steam generator models. The approach for the IP2 tube-end evaluation was to utilize the results from a previous evaluation and adjust the stress values from this evaluation, as appropriate, for the original NSSS design transients and for changes to these transients due to the 1.4-percent power uprate. The adjustment value was conservatively based on the tube-sheet geometry and tube-hole pitch.

Acceptance Criteria

The primary stresses due to design must remain within the respective ASME B&PV Code allowable values (Reference 7-14). The maximum range of stress intensities is limited to $3S_m$. The cumulative fatigue usage factor must be less than or equal to 1.0, or the ASME fatigue exemption rules must apply, for a 40-year fatigue life for the tube undercut.

A similar approach, using stress factors, was utilized in the investigation of fatigue for the tube undercut machining.

Results

The results obtained found that all revised stresses for the 1.4-percent power uprate conditions are all within ASME B&PV Code allowable values.

It was found that fatigue usage values, when adjusted for the 1.4-percent power uprate conditions, remain acceptable.

Conclusions

The stress evaluation of the IP2 Model 44F steam generators determined that the stresses are all within ASME B&PV Code allowable values. Also, the fatigue usage factors were found to remain less than 1.0.

Collar-Cable-Stabilizer Qualification

The Westinghouse collar-cable stabilizer consists of a central coaxial cable made up of Type 302 stainless steel wire strands protected over its full length of the stabilizer by several Type 304 stainless steel tubular collars, which are swaged onto the cable. The swaged collars are about 8 inches long with a longitudinal space of about 1 inch between the adjacent collar segments. This arrangement provides flexibility and dynamic damping.

Description of Evaluation

The qualification method employed was to show that the wall of an assumed hypothetical fully severed host tube would wear out before the stabilizer collar wears away, should a random wear couple form between the severed host tube and the stabilizer collar. Under these conditions, the central coaxial cable of the stabilizer would remain intact and protected by the collar remnant for the life of the installation. The evaluation approach was based on the relative wear coefficients and cross-sectional areas of the tube and stabilizer, and is independent of the dynamic fluid forces causing potential random vibration of the assumed severed host tube.

Acceptance Criteria

The design intent of the Westinghouse cable stabilizer is that the local tube wall wears out totally before the tubular segment of the stabilizer wears out, thereby providing positive protection from wear of the stabilizer's central co-axial cable for the life of the installation. Also, the worn stabilizer remnant should prevent significant contact with the adjacent tubes.

Results

The qualification was based solely on geometric parameters and the relative wear coefficients between the stabilizer collars and the host tube materials. Should a potentially unstable dynamic condition occur and the tube starts to wear with the stabilizer collar, the tube wall essentially was found to wear through before the collar wears through (which protects the central co-axial cable for the life of the installation). Also, potentially deleterious contact with adjacent active tubes was found not to occur.

Conclusions

The evaluation of the straight-leg collar-cable stabilizer for the IP2 Model 44F steam generators determined that the 0.625-inch diameter stabilizer is acceptable for use in the 0.875-inch diameter, 0.050-inch nominal wall tubes for operation at the 1.4-percent power uprate conditions.

Structural Evaluation Conclusions

Results of the analyses performed on the IP2 Model 44F replacement steam generators show that all steam generator components continue to meet ASME B&PV Code Section III, "Rules for Construction of Nuclear Vessels," 1965 Edition, through Summer 1966 Addenda (Reference 7-14) limits for the 1.4-percent power uprate conditions. The primary-to-secondary pressure differential remains below the design value of 1700 psia for normal operating and 1870 psi for upset conditions. In addition, both the weld plugs and mechanical plugs remain qualified for use in the IP2 Model 44F steam generators.

7.7.5 Regulatory Guide 1.121 Analysis

The heat transfer area of steam generators in a pressurized water reactor (PWR) NSSS comprises over 50 percent of the total primary system pressure boundary. The steam generator tubing, therefore, represents a primary barrier against the release of radioactivity to the environment. For this reason, conservative design criteria have been established for the maintenance of tube structural integrity under the postulated design-basis accident condition loadings in accordance with Section III of the ASME Code.

Over a period of time, under the influence of the operating loads and environment in the steam generator, some tubes may become degraded in local areas. Partially degraded tubes are satisfactory for continued service as long as the defined stress and leakage limits are satisfied, and as long as the prescribed structural limit is adjusted to account for possible uncertainties in the eddy current inspection and an operational allowance for continued tube degradation until the next scheduled inspection.

The NRC Regulatory Guide 1.121 (Reference 7-16) describes an acceptable method for establishing the limiting safe condition of degradation in the tubes beyond which tubes found defective by the established in-service inspection shall be removed from service. The level of acceptable degradation is referred to as the "repair limit." For tube cracking due to fatigue and/or stress corrosion, a specification on the maximum allowable leak rate during normal operation must be established such that a reasonable likelihood that LBB would be achieved. If the leak rate exceeds the specification, the unit must be shut down and corrective actions taken to restore the integrity of the unit. The EPRI PWR Primary-to-Secondary Leak Guidelines (Reference 7-17) form the basis of the unit's operational leakage program.

Description of Evaluation

An analysis has been performed to define the "structural limits" for an assumed uniform thinning mode of degradation in both the axial and circumferential directions. The "structural limit" is defined as the percent of wall loss that may occur, before the tube becomes structurally unsound for continuing operation. After the structural limit is lost due to degradation the remaining tube wall thickness is referred to as T_{min}. The assumption of uniform thinning is generally regarded to result in a conservative structural limit for all flaw types occurring in the field. The allowable tube repair limit, in accordance with Regulatory Guide 1.121 (Reference 7-16), is obtained by incorporating into the resulting structural limit a growth allowance for continued operation until the next scheduled inspection and also an allowance for eddy current measurement uncertainty. Calculations have been performed to establish the structural limit for the tube straight-leg (free-span) region of the tube for degradation over an unlimited axial extent, and for degradation over limited axial extent at the tube support plate (TSP), flow distribution baffle (FDB), and AVB intersections.

Results and Conclusions

A summary of the tube structural limits as determined by this analysis for both the High T_{avg} and Low T_{avg} operating conditions is provided in Table 7-7. The corresponding repair limits are established by subtracting from the structural limits an allowance for eddy current measurement uncertainty and continued growth. The reduced t_{min} requirements established for the AVB intersections in Table 7-7 only apply to tube rows 14 and higher. The t_{min} requirements and structural limits corresponding to the FDB are to be used for AVB intersections in tube rows 1 to 13. A complete summary of the Regulatory Guide 1.121 analysis is contained in Westinghouse WCAP report WCAP-15909-P, Rev. 0.

7.7.6 Tube Vibration and Wear

The impact of the proposed 1.4-percent power uprate on the steam generator tubes was evaluated based on the current design-basis analysis and included the changes in the thermal-hydraulic characteristics of the secondary-side of the steam generator resulting from the 1.4-percent uprate. The effects of these changes on the fluid-elastic instability ratio and amplitudes of tube vibration due to turbulences have been addressed. In addition, the effects of the 1.4-percent uprate on potential future tube wear have also been considered.

Description of Analyses and Evaluations

The baseline tube vibration and wear analysis results for the IP2 Model 44F replacement steam generator demonstrate that the maximum fluid-elastic stability ratio for the expected tube support conditions was less than the allowable limit of 1.0. The original tube vibration analysis also determined that negligible tube responses occurred due to the vortex shedding mechanism. The amplitudes of vibration due to turbulence were also found to be reasonably small with maximum displacements found to be on the order of a few mils (.0067"). The maximum expected tube wear that could occur over the remaining period of operation was calculated to be .0013".

Table 7-7 Summary of Tube Structural Limits Regulatory Guide 1.121 Analysis			
Location / Wear Scar Length	**Parameter**	**High T_{avg}**	**Low T_{avg}**
Straight Leg	t_{min} (inch)	0.022	0.024
	Structural Limit (%) [1]	56.0	52.0
Anti-Vibration Bar [2]/0.5 in.	t_{min} (inch)	0.014	0.016
	Structural Limit (%) [1]	71.1	67.8
Flow Distribution Baffle/0.75 in.	t_{min} (inch)	0.018	0.020
	Structural Limit (%) [1]	64.6	61.0
Tube Support Plate/1.125 in.	t_{min} (inch)	0.021	0.022
	Structural Limit (%) [1]	59.0	55.2

Notes:

1. Structural Limit $= [(t_{nom} - t_{min}) / t_{nom}] \times 100\%$

   $t_{nom} = 0.050$ in

2. The tube structural limits and minimum thickness specified for the AVB applies only for tube rows 14 and higher. The structural limits and minimum thickness for the FDB locations are to be used for tube/AVB intersections or tube rows 1 to 13.

The results of the vibration and wear analysis were modified to account for anticipated changes in secondary-side operating conditions due to the uprate.

For the expected support conditions, it was found that straight-leg stability ratios were not significantly impacted. However the stability ratio for U-bend conditions increased from 0.7 to 0.77, and is still less than the allowable limit of 1.0. As a result, the analysis indicates that large amplitudes of vibration are not projected to occur due to the fluid-elastic mechanism while operating the steam generator in the 1.4-percent power uprate operating condition.

The maximum displacement values calculated for turbulence excitation in the original analysis were modified to account for the 1.4-percent-uprate-induced changes in the operating conditions. For the most limiting tube support condition, it was determined that the turbulence-induced displacement could increase from approximately .007" to approximately 0.010". Displacements of this magnitude are not sufficient to produce tube-to-tube contact. However, the potential for tube wear must be considered.

As in the original analysis, the vortex shedding mechanism was not found to be a significant contributor to tube vibration.

The potential for tube wear was addressed in the original analysis and addressed wear in both the straight-leg and U-bend portions of the steam generator. These calculations were then updated to reflect operation of the steam generators in the 1.4-percent uprate conditions. The calculations determined that the level of tube wear that could occur would increase from approximately .0013" to approximately 0.002" at the power uprate conditions. From these calculations, it can be concluded that although there may be an increase in the level of tube wear that would occur at the 1.4-percent uprate operating conditions, the increased level would not be significant. Any increase in the rate of tube wear would progress over many cycles and would be observable during normal eddy current inspections.

Effect of Changes in Primary-to-Secondary-Side Pressure Differential

Concerns have been raised in the review of other uprating evaluations regarding the effects of the change in primary-to-secondary-side pressure will have on tube wear. It should be noted that there is no direct correlation of flow-induced vibration with primary-to-secondary-side pressure differences. The steam generator tubes respond primarily to the conditions associated with the secondary side since the forcing functions associated with the secondary side of the steam generator dominate over any other effects. Any effects of primary-to-secondary-side pressure difference are inherently considered in the analysis in that the secondary-side conditions are defined by the total steam generator conditions such as steam pressure, flow rates, or recirculation, and includes the primary–to-secondary-side pressure difference.

High Cycle Fatigue Considerations

Note that in some model steam generators, particular consideration is given to the potential for high cycle fatigue of U-bend tubes. This phenomenon has been observed in tubes with carbon steel support plates where denting or a fixed tube support condition has been observed in the upper most plate. However, since the IP2 steam generator tube support plates are manufactured from stainless steel, there is no potential for the necessary boundary conditions (i.e., denting) to occur at the uppermost support plate. Hence, high cycle fatigue of U-bend tubes will not be an issue at IP2.

Conclusions

The analysis of the IP2 Model 44F replacement steam generators indicates that significant levels of tube vibration will not occur from either the fluid-elastic, vortex shedding, or turbulent mechanisms as a result of the 1.4 percent uprate. In addition, the projected level of tube wear as a result of vibration is expected to be small and not result in unacceptable wear.

7.7.7 Tube Integrity

Over a period of time, some tubes may become degraded locally under the influence of the operating loads and chemical environment in the steam generator. Degradation mechanisms observed in the first-generation steam generators (e.g., those using mill-annealed Alloy 600 tubing) include outside-diameter stress corrosion cracking (ODSCC), primary water system stress corrosion cracking (PWSCC), pitting, as well as tube wear at AVBs and TSPs due to tube vibration, and potentially at other locations, such as the FDB, due to maintenance operations. The potential for these degradation mechanisms affecting the IP2 steam generators due to the 1.4-percent power uprate is discussed below.

The IP2 steam generators are Model 44F steam generators that utilize Alloy 600 thermally treated (TT) tubes. Comparative studies (for example, Reference 7-18) of the performance of Alloy 600TT and Alloy 600 mill-annealed (MA) have shown Alloy 600TT to have superior resistance to corrosion compared to Alloy 600 MA. Plants utilizing Alloy 600TT have operated without evidence of PWSCC for over 15 EFPYs. ODSCC was reported in a plant with Alloy 600TT tubing in May 2002 after about 9.7 EFPYs operation. The cause for the ODSCC in this plant has not yet been confirmed. All other domestic steam generators with Alloy 600TT tubing have operated without evidence of cracking; the maximum operating time among these plants is over 15 EFPYs. These steam generators are operated at a higher temperature (approximately 618°F) than estimated for the IP2 uprated conditions. At the IP2 operating temperature of <612°F at the 1.4-percent power uprate conditions, a significantly longer operating period prior to initiation of corrosion degradation is expected.

The 1.4-percent power uprate results in a 0.5°F temperature increase, which has a negligible effect on the incidence of PWSCC in the IP2 steam generators. Similarly, based on the Arrhenius equation to compare corrosion initiation times from the pre- and post- 1.4-percent uprate temperatures, the change in the initiation time is negligible.

Pitting is principally a function of the chemical environment in which the tubes operate. The very small increase in temperature at the 1.4-percent power uprate conditions will not significantly affect the potential for the pitting of the tubes.

Section 7.7.5 summarizes the results of an analysis performed to define the limits on degradation allowable in the various regions of the steam generator tube (RG 1.121 analysis). The extent of allowable degradation is expressed in terms of structural limits. The results of an updated tube vibration and wear analysis to include the effects of 1.4-percent power uprate are summarized in Section 7.7.6. The RG 1.121 analysis is conservative because:

- The ASME Code minimum material properties are used.
- A bounding assumption of uniform thinning is assumed.

The actual average tensile properties and ultimate strength of the IP2 tubes are expected to be much higher than the code minimum value used in the analysis. For AVB wear, the assumption of uniform thinning is extremely conservative. Tests have shown that for AVB wear of approximately 0.4 inch in length, burst does not occur at wear depths greater than 90-percent of throughwall in Alloy 600TT tubing. For the condition tested, an analysis similar to the RG 1.121 analysis discussed in Section 7.7.5 predicted a structural limit of about 75-percent of throughwall using the same assumptions (code minimum allowables, uniform thinning). Therefore, testing has confirmed that the analysis is very conservative.

Typically, in the first-generation steam generators and in the Model F steam generators, a few tubes wear rapidly at the onset of operation. After 2 to 4 cycles of operation, high growth rates are no longer observed, and the growth rate continues to decline with addition operating time. For AVB wear, using background noise typical of the operating Model F (10-15 EFPY operation), the detectability of wear was shown to be about 8-percent of throughwall for Alloy 600 AVBs approximately 0.3-inch wide. At this depth, all eddy current analysts detected 100 percent of indications.

Shallow wear at the FDB has been observed only in steam generators that have experienced pressure-pulse cleaning. The wear is attributed to tube impact against the FDB during the cleaning operation. Reinspection of the shallow wear in these steam generators has established that the wear is not progressing, i.e., it has a zero growth rate. Detection of FDB and tube support plate wear is excellent, i.e., 100 percent of indications greater than approximately 8- to 10-percent of depth are detected.

Wear at TSPs not related to foreign objects has rarely been observed. Insufficient data are available to determine the growth rate. It is estimated that the growth rate is less than that for typical AVB wear since the AVB wear mechanism (fluid-elastic excitation of the tubes) is more energetic than the turbulence-induced tube motion at the TSPs. (Note that pre-heater turbulence induced wear is a non-applicable special case, since the pre-heater tubes were directly in the feedwater cross-flow field.) Therefore, a conservative estimate for TSP wear growth is approximately 4- to 6-percent of throughwall per 24-month operating cycle.

Straight-leg wear has been observed only as the result of a foreign object or due to maintenance tooling contacting tubes. Detection of free-span wear is excellent; generally better than AVB wear since foreign object wear tends to exhibit sharper discontinuities. Wear due to tool contact has no growth rate because the tools are not present during operation. Wear growth due to foreign object impingement cannot be predicted without a prior history of the object. Foreign object wear, when detected, is conservatively analyzed and proper disposition made relative to the applicable structural limit.

It is concluded that the current Technical Specification repair limit of 40-percent throughwall is adequate for the following reasons:

- The predicted structural limit is conservative due to the use of ASME Code minimum properties in the RG 1.121 analysis.
- Tests have demonstrated that the structural limit for AVB wear is significantly higher than the predicted limit.
- The 100-percent detection threshold for all forms of wear is very low ($\leq$ 8-percent of throughwall).
- Observed growth rates for operationally induced wear are small.
- Growth rates for maintenance-service-related wear are zero.

The inspection requirements in terms of the number of steam generators inspected and the sample size of tubes inspected depends on the progression (if any) of degradation. None of the potential degradation mechanisms are significantly affected by the 1.4-percent power uprate conditions. Therefore, the required frequency of inspection is also not affected significantly by the 1.4-percent power uprate.

7.8 PRESSURIZER

Evaluations of the pressurizer structural integrity have been performed to address operation at the 1.4-percent power uprate conditions.

7.8.1 Structural Analysis

The functions of the pressurizer are to absorb any expansion or contraction of the primary reactor coolant due to changes in temperature and/or pressure, and, in conjunction with the Pressure Control System components, to keep the RCS at the desired pressure. The first function is accomplished by maintaining the pressurizer approximately half full of water and half full of steam at normal conditions, connecting the pressurizer to the RCS at the hot leg of one of the reactor coolant loops, and allowing inflow to, or outflow from, the pressurizer as required. The second function is accomplished by maintaining the temperature in the pressurizer at the water saturation temperature (T_{sat}) corresponding to the desired pressure. The temperature of the water and steam in the pressurizer can be raised by operating electric heaters at the bottom of the pressurizer. The temperature of the water and the steam can be lowered by introducing relatively cool spray water into the steam space at the top of the pressurizer.

The components in the lower end of the pressurizer (e.g., surge nozzle, lower head/heater well, and support skirt) are affected by pressure and surges through the surge nozzle. The components in the upper end of the pressurizer (e.g., spray nozzle, safety and relief nozzle, upper head/upper shell, manway, and instrument nozzle) are affected by pressure, spray flow through the spray nozzle, and steam temperature differences.

Input Parameters and Assumptions

The reactor vessel outlet (T_{hot}) and the reactor vessel inlet (T_{cold}) temperatures from the NSSS design parameters in Table 2-1 define the normal operating temperatures for the surge and spray lines to the pressurizer. The reactor coolant pressure defines the pressurizer normal operating pressure (2250 psia) and saturated temperature (653°F). The minimum values of T_{hot} and T_{cold} from all cases included in Table 2-1 were used in this evaluation. The NSSS design transients discussed in Section 5.1 are also applicable to the pressurizer and were considered in this analysis.

Description of Analyses and Evaluation

The limiting operating conditions of the pressurizer occur when the RCS pressure is high and the RCS hot-leg (T_{hot}) and cold-leg (T_{cold}) temperatures are low. This maximizes the differential temperature, ΔT, that is experienced by the pressurizer. Due to flow out of and into the pressurizer during various transients, the surge nozzle alternately sees water at the pressurizer temperature (T_{sat}) and water from the RCS hot leg at T_{hot}. If the RCS pressure is high (which means, correspondingly, that T_{sat} is high) and T_{hot} is low, then the surge nozzle will see maximum thermal gradients (ΔT_{hot} = temperature difference between T_{hot} and the pressurizer (surge nozzle) temperature) and, thus, experiences the maximum thermal stress.

Likewise, the spray nozzle and upper shell temperature alternate between steam at T_{sat} and spray water, which for many transients is at T_{cold}. Thus, if RCS pressure is high (T_{sat} is high) and T_{cold} is low, then the spray nozzle and upper shell will experience the maximum thermal gradients [ΔT_{cold} = temperature difference between T_{cold} and the pressurizer (spray nozzle) temperature] and thermal stresses.

Only the surge and spray nozzle need to be evaluated. Since all other components experience lower stresses, all other components are not impacted by the 1.4-percent uprate. This surge and spray nozzle evaluation was based on the range of NSSS operating parameters to support an NSSS power level of 3127 MWt (Table 2-1).

The input parameters associated with the IP2 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program were reviewed and compared to the design inputs considered in the current pressurizer stress report. In cases where revised input parameters are not obviously bounded, pressurizer structural analyses and evaluations were performed. Using the existing design basis analyses as a starting point, scaling factors were applied to assess the impact of the changes in the parameters such as the system transients, temperatures, and pressures. New stresses and revised cumulative usage factors are calculated, as applicable, and compared to previous results.

For the 1.4-percent uprate power conditions, only the change in the ΔT_{cold} required an analysis of key upper shell components such as the spray nozzle, the safety and relief nozzle, and the upper shell itself. Since the change in the ΔT_{hot} was minimal and bounded by the original design basis, no analyses were necessary for the lower shell and its key components.

Acceptance Criteria

The acceptance criteria for each component are consistent with the criteria used in the design-basis analysis referenced for the component, as reported in the original stress report. The maximum range of primary-plus-secondary stresses was compared with the corresponding 3 S_m limits of the ASME Boiler and Pressure Vessel Code (Reference 7-14). A cumulative fatigue usage factor of less than or equal to 1.0 demonstrates design adequacy for a 40-year design life.

Conclusions

The analysis results demonstrate that the IP2 1.4-percent power uprate will have a minimal effect on the pressurizer components. Table 7-8 compares the fatigue usages calculated with those from the original design basis. The largest increase was for the spray nozzle where the fatigue usage increased from 0.848 to 0.994. The fatigue usage for the upper shell decreased significantly due to the removal of excessive conservatism in the original evaluation. For components not listed in Table 7-8, the current design basis analyses remains bounding.

It is concluded that the pressurizer components meet the stress/fatigue analysis requirement of the ASME Code, Section III (Reference 7-14) for plant operation at the 1.4-percent power uprate conditions.

Table 7-8
IP2 Fatigue Usage Components

Component	Revised Fatigue Usage	Previous Fatigue Usage
Spray Nozzle	0.994	0.848
Upper Shell	0.4158	0.9863
Safety and Relief Nozzle	0.2047	0.148

7.9 NSSS AUXILIARY EQUIPMENT

The NSSS auxiliary equipment includes the heat exchangers, pumps, valves, and tanks. An evaluation was performed to determine the potential effect that the 1.4-percent power uprate design conditions will have on these components. In Section 5.2, it was shown that these revised design conditions do not affect the auxiliary equipment design transients.

The revised design conditions have been evaluated with respect to the potential effects on the auxiliary heat exchangers, valves, pumps, and tanks. The results of this review show that the NSSS auxiliary equipment continue to meet the design pressure and temperature requirements, as well as the fatigue usage factors and allowable limits, for which the equipment is designed. Therefore, the IP2 NSSS auxiliary equipment is unaffected by the 1.4-percent power uprate.

7.10 FUEL EVALUATION

This section summarizes the evaluations performed to determine the potential effects of the 1.4-percent power uprate on the nuclear fuel at IP2. Fuel evaluations are performed for each specific IP2 operating cycle, and those evaluations vary based on the needs and specifications of each cycle according to the Westinghouse Reload Methodology in WCAP-9272-P-A (Reference 7-19). The evaluations herein address fuel-related analyses that are not cycle-specific, and that are directly affected by the 1.4-percent power uprate through changes in the related non-LOCA accident analyses limits that are used in various elements of the fuel and core design. The potential effects of the 1.4-percent power uprate on these analyses were evaluated in terms of the IP2 fuel/core nuclear design, the fuel rod design, the core thermal-hydraulic design, and the fuel structural integrity. This was done based on both the Westinghouse VANTAGE 5 and VANTAGE+ fuel types, although only VANTAGE+ fuel is currently being used at IP2 in Cycle 16.

The potential effects of the 1.4-percent power uprate on the analyses that are cycle-specific will be addressed consistent with the Westinghouse Reload Methodology (Reference 7-19) prior to implementation of the uprate.

7.10.1 Nuclear Design

Most of the non-LOCA-related safety analysis statepoints are unchanged for the 1.4-percent power uprate, and there is sufficient margin to applicable limits to accommodate a 1.4-percent power uprate. For statepoints that do change, an analysis was performed to show that the new statepoints meet the current design basis to support the implementation of the 1.4-percent power uprate at IP2 during Cycle 16.

All nuclear design analysis was performed using the standard Westinghouse core reload methodology described in Reference 7-19, and with the Westinghouse PHOENIX-P and ANC codes described in References 7-20 and 7-21, respectively.

Cycle-specific core design analyses are performed for each reload cycle to ensure that all core design and reload safety analysis parameters will be satisfied for the specific operating conditions associated with that cycle. These analyses will be repeated prior to implementation of the 1.4-percent power uprate of IP2 during Cycle 16 as well as prior to all subsequent cycles.

7.10.2 Fuel Rod Design

The current fuel rod design analyses for IP2 have been reviewed to assess the potential effect of the 1.4-percent power uprate. The design margin for rod internal pressure (gap re-opening and DNB propagation) and cladding stress have been re-evaluated based on the 1.4-percent power uprate conditions. Results indicate that these fuel rod design parameters will continue to meet the acceptance criteria at the 1.4-percent power uprate conditions. The remaining fuel rod design criteria are negligibly affected by an increase in power level, and sufficient margin currently exists to offset the result of a 1.4-percent power uprate.

The fuel rod design calculations were performed using the PAD 4.0 fuel performance models described in Reference 7-22.

Cycle-specific fuel rod design analyses are performed for each reload cycle to ensure that all fuel rod design criteria are satisfied for the specific operating conditions associated with that cycle (Reference 7-19). These analyses will be repeated prior to the implementation of the 1.4-percent power uprate of IP2 during Cycle 16, as well as prior to all subsequent cycles.

7.10.3 Core Thermal-Hydraulic Design

A core thermal-hydraulic evaluation was performed at the 1.4-percent power uprate nominal core power level of 3115 MWt. The evaluation was based on the 15x15 VANTAGE+ fuel design with the IFM grids that bounds the 15x15 VANTAGE 5 fuel design for future reloads. The key input parameters are summarized in Table 7-9.

The current design methodology for the IP2 reload safety evaluation remains unchanged for the 1.4-percent 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 current RTDP DNB ratio (DNBR) design limits with the revised power measurement uncertainty have been verified to meet the 95/95 DNB design basis. The DNBR safety analysis limits have been revised to account for increases in the nominal power and the best-estimate core bypass flow fraction. The DNBR limits and margin summary are provided in Table 7-10.

The Westinghouse version of the VIPRE-01 (VIPRE) code is used for DNBR calculations with the WRB-1 and the W-3 DNB correlations. The VIPRE code is equivalent to the THINC-IV (THINC) code and has been approved by the NRC for licensing applications to replace the THINC code. The use of VIPRE for the 1.4-percent power uprate analysis is in full compliance with the conditions specified in the NRC Safety Evaluation Report (SER) in WCAP-14565-P-A (Reference 7-23).

Based on the parameter values in Table 7-9, VIPRE DNBR calculations were performed for the 15x15 VANTAGE+ with IFM fuel to confirm the core thermal limits and to verify that the DNB design basis is met for DNB limiting events. The DNBR portion of the core limits and the axial offset limits were unchanged at the 1.4-percent power uprate conditions, in order to minimize the effect on the OTΔT and OPΔT protection setpoints. The limiting DNB transients, including loss of flow, locked rotor, dynamic dropped rod, static rod misalignment, rod withdrawal from subcritical, and steam line break events, all meet the revised DNBR Safety Analysis Limits in Table 7-10.

Table 7-9
Summary of Key Thermal-Hydraulic Input Parameters

Parameter	Cycle 16 (Current Power)	1.4% Power Uprate
Core Power, MWt	3071.4	3115
Minimum Measured Flow [1], gpm	330,000	330,000
Thermal Design Flow [2], gpm	322,800	322,800
Best-Estimate Core Bypass Flow [1]	5.0	5.9
Design Core Bypass Flow [2]	6.5	6.5
Vessel Average Temperature, °F	579.7	579.2
System Pressure, psia	2250	2250
Design $F_{\Delta H}$ Limit:		
VANTAGE+ Fuel	1.70	1.70
VANTAGE 5 Fuel	1.65	1.62
$F_{\Delta H}$ Part-Power Multiplier	0.3	0.3

Notes:

1. Used with Revised Thermal Design Procedure (RTDP).
2. Used with Standard Thermal Design Procedure (STDP).

Because of the significant amount of burnup, the 15x15 VANTAGE 5 fuel is less limiting than the VANTAGE+ with IFM fuel. The VANTAGE 5 fuel assemblies may be used for future reloads with a maximum hot channel enthalpy rise ($F_{\Delta H}$) value less than 1.62 (RTDP $F_{\Delta H}$ of 1.56) at the current DNBR limits.

In summary, a core thermal-hydraulic design evaluation has been performed for the current IP2 fuel designs in support of the 1.4-percent power uprate, based on an equivalent methodology used for the IP2 reload evaluations. The evaluation concludes that the current core operating limits and the DNB limiting events continue to meet the DNB design basis at the 1.4-percent power uprate nominal core power level of 3115 MWt.

Table 7-10 RTDP DNBR Limits and Margin Summary for 1.4% Core Power Uprate		
Parameter	**15x15 VANTAGE 5 Fuel**	**15x15 VANTAGE+ Fuel**
DNB Correlation	WRB-1	WRB-1
Safety Analysis Limit DNBR	1.58	1.58 [1]
Design Limit DNBR (Typical / Thimble Cells)	1.25 / 1.24	1.26 / 1.25
Margin Reserved (% DNBR)	15.5	14.9
DNBR Penalties (% DNBR):		
a) Rod Bow	0.25	0.25 [2]
b) Transition Core	<6.3	< 8.0
c) Barton Transmitter	1.0	1.0
d) Axial Offset (AO) Violation Penalty	5.5	N/A
Net Margin Available (% DNBR)	>2.45	> 5.65

Notes:

1. The Safety Analysis Limit DNBR of 1.58 is equivalent to 1.48 when reduced for penalties for the 1.4-percent power uprate and for core bypass flow increase due to VANTAGE+ fuel.
2. Applicable to the grid spans without IFM grids.

7.10.4 Fuel Structural Evaluation

The VANTAGE 5 and VANTAGE+ assembly designs were evaluated to determine the potential effect of the 1.4-percent power uprate on the fuel assembly structural integrity. The original core plate motions remain applicable for the 1.4-percent power uprate. The generic analysis (Westinghouse Owners Group (WOG) Control Rod Insertion Program) also remains bounding for Indian Point Unit 2 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program. Therefore, there is no effect on the fuel assembly seismic/LOCA structural evaluation. The 1.4-percent power uprate has an insignificant effect on the operating and transient loads, such that there is no adverse effect on the fuel assembly functional requirements. Therefore, the fuel assembly structural integrity is not affected, and the seismic and LOCA evaluations for the VANTAGE 5 and VANTAGE+ fuel assembly designs remain applicable. This evaluation was done specifically for the VANTAGE 5 and VANTAGE+ fuel assembly designs. However, other fuel designs can also be used in the future at IP2 if justified by cycle-specific evaluations.

7.11 REFERENCES

7-1 WCAP-15583, "Addendum to Analytical Report for the Indian Point Reactor Vessel Unit No. 2 (Steam Generator Replacement Evaluation)," October 2000.

7-2 ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, 1965 Edition, American Society of Mechanical Engineers, New York.

7-3 WCAP-15629, Revision 1 "Indian Point Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation," T. J. Laubham, December 2001.

7-4 NRC Amendment No. 224, "Entergy Nuclear Indian Point 2, LLC Entergy Nuclear Operations, Inc. Docket No. 50-247 Indian Point Nuclear Generating Unit No. 2 Amendment to Facility Operating License," J. T. Munday (Office of NRR), February 15, 2002.

7-5 Regulatory Guide 1.190 "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence."

7-6 ASTM E185-82, Annual Book of ASTM Standards, Section 12, Volume 12.02, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels."

7-7 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Federal Register, Volume 60, No. 243, December 19, 1995.

7-8 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements," Federal Register, Volume 60, No. 243, December 19, 1995.

7-9 USA Standard Code for Pressure Piping, Power Piping USAS B31.1 – 1973 (including Summer Addendum).

7-10 WCAP-12937, "Structural Evaluation of Indian Point Units 2 and 3 Pressurizer Surge Lines, Considering the Effects of Thermal Stratification," May 1991.

7-11 WCAP-10977, Revision 2, "Technical Bases for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Indian Point Unit 2," December 1986.

7-12 WCAP-10977, Supplement 1, "Additional Information in Support of the Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for Indian Point Unit 2," January 1989.

7-13 WCAP-12187, "Consolidated Edison Company of New York, Inc., Indian Point Unit 2, 3083.4 MWt Stretch Rating Engineering Report," March 1989.

7-14 ASME B&PV Code Section III, "Rules for Construction of Nuclear Vessels," American Society of Mechanical Engineers, New York, NY, 1965, plus Addenda through Summer 1966.

7-15 ASME B&PV Code Section III, "Rules for Construction of Nuclear Vessels," American Society of Mechanical Engineers, New York, NY, 1989 Edition.

7-16 USNRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes (for comment)," August 1976.

7-17 EPRI Report TR-104788-R2, "PWR Primary-To-Secondary Leak Guidelines – Revision 2," EPRI, Palo Alto, CA, 2000.

7-18 EPRI TR-108501, "Predicted Tube Degradation for Westinghouse Models D5 and F-Type Steam Generators"; September 1997.

7-19 WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," S. Davidson, et al., July 1985.

7-20 WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," T. Q. Nguyen, et al., June 1988.

7-21 WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code," Y. S. Liu, et al., September 1986.

7-22 WCAP-15063-P-A, Revision 1 (Proprietary) and WCAP-15064-NP-A, Revision 1 Proprietary), "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)," W. H. Slagle, (Editor), et al., July 2000.

7-23 WCAP-14565-P-A/WCAP-15306-NP-A, "VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis," Y. Sung, et al., October 1999.

8 UFSAR CHAPTER 14 ACCIDENT ANALYSES AND CALCULATIONS

Table 8-1 summarizes the Updated Final Safety Analysis Report (UFSAR) Chapter 14 accident analyses and calculations for Indian Point Unit 2 (IP2) and identifies whether each analysis is affected or unaffected (according to Nuclear Regulatory Commission (NRC) Regulatory Issue Summary (RIS) 2002-03) by the 1.4-percent power uprate. Details of how these analyses were evaluated or reanalyzed for the 1.4-percent power uprate follow in subsequent sections.

The current UFSAR Chapter 14 safety analyses that bound the 1.4-percent power uprate already assume a 2-percent uncertainty on power. These events are delineated in Table 8-1. As discussed in Section 8, these transients did not require explicit re-analyses for the 1.4-percent uprate because the power level assumed in the current analyses (current core power level of 3071.4 MWt plus 2-percent uncertainty) is equivalent to the 1.4-percent uprate power of 3114.4 MWt plus 0.6-percent uncertainty. The other Nuclear Steam Supply System (NSSS) design parameters that changed are the vessel average temperatures and the steam pressure. However, the vessel average temperatures changed by 0.5°F (and are bounded by the non-uprate T_{avg} values) and the steam pressure decreased by less than 10 psi. These condition changes were evaluated and determined to have an insignificant effect on the results of the safety analyses. The current safety-analysis basis was maintained for design parameters such as Reactor Coolant System (RCS) thermal design flow, T_{hot}, T_{cold} and steam generator tube plugging. Therefore, the 1.4-percent power uprate has no effect on the results of the current IP2 safety analyses that already assume a 2-percent uncertainty on power.

Table 8-1
Accident Analysis Design Basis Events

UFSAR Section	Event Description	Affected or Unaffected
Loss-of-Coolant-Accident (LOCA) Related Events		
14.3.4	LOCA Forces	Unaffected
14.3	Large-Break LOCA	Unaffected
14.3	Small-Break LOCA	Unaffected
NA	Post-LOCA Long-Term Core Cooling	Unaffected
NA	Hot-Leg Switchover Analysis	Unaffected
Non-LOCA Events		
Affected Events Re-analyzed for 1.4% Power Uprate		
14.1.2	Uncontrolled Rod Cluster Control Assembly (RCCA) Withdrawal at Power	Affected
14.1.8	Loss of External Electrical Load – Departure from Nucleate Boiling (DNB) Analysis	Affected

Table 8-1 Accident Analysis Design Basis Events (cont.)		
UFSAR Section	**Event Description**	**Affected or Unaffected**
14.1.10	Excessive Heat Removal Due to Feedwater System Malfunctions – Full Power Analysis	Affected
Affected Events Evaluated for the 1.4% Power Uprate Using Existing DNB Margin		
14.1.4	RCCA Drop	Affected
14.1.6	Loss of Reactor Coolant Flow – Partial and Complete Loss of Flow Analyses	Affected
14.1.6.5	Locked Rotor Accident – DNB Analysis	Affected
Events Bounded by Current 102% Power Assumption		
14.1.6.5	Locked Rotor Accident – Overpressure, Maximum Clad Temperature, and Maximum Zirconium-Water Reaction Analysis	Unaffected
14.1.8	Loss of External Electrical Load – Overpressure Analysis	Unaffected
14.1.9	Loss of Normal Feedwater (LONF)	Unaffected
14.1.12	Loss of All AC Power to the Station Auxiliaries	Unaffected
14.2.6	Rupture of a Control Rod Mechanism Housing (RCCA Ejection) – Full Power Analysis	Unaffected
Non-Limiting/Bounded Events		
14.1.1	Uncontrolled RCCA Withdrawal from a Subcritical or Low Power Startup Condition	Unaffected
14.1.5	Chemical and Volume Control System (CVCS) Malfunction	Unaffected
14.1.7	Startup of an Inactive Loop	Unaffected
14.1.10	Excessive Heat Removal Due to Feedwater System Malfunctions – Zero Power Analysis	Unaffected
14.1.11	Excessive Load Increase Incident	Unaffected
14.2.5	Rupture of a Steam Pipe – Zero Power Analysis	Unaffected
14.2.6	Rupture of a Control Rod Mechanism Housing (RCCA Ejection) – Zero Power Analysis	Unaffected
Radiological Dose Calculations		
14.2.1	Fuel Handling Accident Dose	Unaffected
14.2.2	Accidental Release of Waste Liquid Dose	Unaffected

Table 8-1 Accident Analysis Design Basis Events (cont.)		
UFSAR Section	Event Description	Affected or Unaffected
14.2.3	Accidental Release – Waste Gas	Unaffected
14.2.4	Steam Generator Tube Rupture	Unaffected
14.2.5.5	Rupture of a Steam Pipe Dose	Unaffected
14.3.6	LOCA Dose	Unaffected

8.1 LOCA HYDRAULIC FORCES

Loss-of-coolant accident hydraulic forces are used as input for faulted condition structural qualification of the RCS, including the reactor vessel, internals, fuel, loop piping, and supports. Loss-of-coolant accident forces are not directly affected by the reactor power, but LOCA forces are sensitive to RCS temperature and pressure, where decreases in temperature and increases in pressure lead to increases in calculated LOCA forces. As shown in Reference 8-1, LOCA forces are calculated at full power, minimum temperature, and minimum thermal design flow conditions to minimize the RCS cold-leg temperatures and, therefore, maximize the calculated LOCA forces. Because increases in reactor power can be accompanied by decreases in the minimum full-power cold-leg operating temperature, the power uprate conditions must be evaluated relative to the conditions assumed in the LOCA forces calculations.

The LOCA forces calculations currently applied to IP2 for qualification of reactor vessel, loop, and internals were developed using the NRC-reviewed and approved MULTIFLEX computer code (Reference 8-1). These LOCA forces were originally calculated for an IP2 Stretch Rating Program, which utilized accumulator and pressurizer surge line breaks and assumed a minimum cold-leg operating temperature of 515.8°F. The minimum cold-leg temperature for the IP2 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program is 515.8°F. The IP2 calculation included an RCS pressurizer pressure of 2250 psia, plus a 30 psi uncertainty for a total of 2280 psia. The IP2 full-power operating pressure is 2250 psia. Therefore, the 1.4-percent power uprate conditions for IP2 remain bounded by the existing LOCA forces.

In addition to these analyses, which are part of the existing design basis, there were two generic LOCA forces analyses developed with the intention of bounding IP2 such that the results may be applied to IP2 as needed. The first of these was an acceptable baffle-barrel-bolting analysis documented in Reference 8-2, performed in accordance with methodology reviewed and approved by the NRC in Reference 8-3. The second was a control rod insertion analysis for cold-leg break LOCA documented in Reference 8-4. This control rod insertion analysis was performed using the methodology identical to that applied to D. C. Cook in Reference 8-5, which was reviewed and approved by the NRC in Reference 8-6. In both cases, the generic analysis assumed a cold-leg temperature of 511.7°F and an RCS pressure of 2317 psia, which remain bounding for the IP2 1.4-Percent Measurement Uncertainty Recapture Power Uprate Program. Both the Reference 8-2 and 8-4 analyses confirm that 15x15 VANTAGE+ fuel remains qualified under bounding LOCA and seismic loads for IP2. Note that because these analyses used

511.7°F not 515.8°F, any restriction on minimum T_{cold} due to uncertainties could be reduced or eliminated with the implementation of these analyses for IP2.

8.2 LOCA AND LOCA-RELATED EVALUATIONS

8.2.1 Appendix K Small-Break LOCA

The current licensing-basis small-break loss-of-coolant accident (SBLOCA) analysis using the Code of Federal Regulations (CFR) 10 CFR Part 50, Appendix K methodology employs a nominal core power of 3071.4 MWt. The licensing-basis methodology includes a 2-percent calorimetric power measurement uncertainty (an assumed core power of 3132.8 MWt) in accordance with the original requirements of 10 CFR 50, Appendix K. Consistent with the recent change to Appendix K, Entergy proposes to reduce the power measurement uncertainty to 0.6 percent. The existing 2-percent uncertainty margin in the SBLOCA analysis is re-allocated with 1.4 percent applied to the increase in the licensed core power level (3114.4 MWt) and 0.6 percent retained to account for power measurement uncertainty. The total core power (including uncertainties) assumed in the analysis is 3132.8 MWt.

8.2.2 Best-Estimate Large-Break LOCA

WCAP-13837, Revision 1, "Best-Estimate Analysis of the Large-Break Loss-of-Coolant Accident for IP2 Nuclear Plant," describes the analysis performed for IP2 utilizing the best-estimate methodology for analyzing a large-break LOCA (LBLOCA). The analysis was performed assuming a core power of 3216 MWt. This core power (3216 MWt) bounds the 1.4-percent uprated power of 3114.4 MWt.

8.2.3 Post-LOCA Long-Term Core Cooling

The Westinghouse licensing position for satisfying the requirements of 10 CFR 50.46, Paragraph (b), Item (5), "Long-term cooling," concludes that the reactor will remain shut down by borated Emergency Core Cooling System (ECCS) water contained in the RCS/sump following a LOCA. Since credit for the control rods is not taken for an LBLOCA, the borated ECCS water provided by the refueling water storage tank (RWST) and accumulators must have a boron concentration that, when mixed with the other sources of water, will result in the reactor core remaining subcritical assuming all control rods out. The calculation is based upon the reactor steady-state conditions at the initiation of a LOCA and considers both borated and unborated fluid in the post-LOCA containment sump. The other sources of water considered in the calculation of the sump boron concentration are the RCS and ECCS/residual heat removal (RHR) piping. The water volumes and associated boric acid concentrations are not directly affected by the 1.4-percent power uprate. The core reload licensing process will confirm that there are no required changes to these volumes and boron concentrations. Therefore, there is no impact on the long-term core cooling (LTCC) analysis.

8.2.4 Hot-Leg Switchover

For a cold-leg break following a LOCA, some of the ECCS injection into the cold leg will circulate around the top of the full downcomer and out of the broken cold leg. Flow stagnation in the core and the boiling off of nearly pure water will increase the boron concentration of the remaining water. As the boron concentration increases, the boron could eventually precipitate and potentially inhibit core cooling.

Thus, at a designated time after a LOCA, the ECCS configuration is switched to hot-leg injection to flush the core with water and keep the boron concentration below the precipitation point. IP2 was licensed with a hot-leg switchover (HLSO) time supported by generic calculations that used representative assumptions for 4-loop plants. Since bounding values for maximum core power were used in establishing the IP2 HLSO time, the HLSO licensing basis remains unchanged and the IP2 HLSO time is not impacted.

8.3 NON-LOCA ANALYSIS

This section addresses the effects of the IP2 1.4-percent power uprate on the non-LOCA analyses presented in Chapter 14 of the IP2 UFSAR.

Non-LOCA design-basis events are documented in Sections 14.1 and 14.2 of the IP2 UFSAR. Three of the affected non-LOCA events were re-analyzed, and the three other affected events were evaluated to address the potential effects of the 1.4-percent power uprate. The remaining analyses were determined to be unaffected, as described herein.

Some non-LOCA analyses are affected by the 1.4-percent power uprate because the current analyses do not already explicitly account for a 2-percent power measurement uncertainty allowance. Therefore, the following 3 UFSAR events were analyzed to address the potential effects of the 1.4-percent power uprate:

- Uncontrolled RCCA Bank Withdrawal at Power (UFSAR Section 14.1.2)
- Loss of External Electrical Load (UFSAR Section 14.1.8) – DNB Analysis
- Excessive Heat Removal Due to Feedwater System Malfunctions (UFSAR Section 14.1.10) – Full Power Analysis

The following UFSAR non-LOCA events are also affected by the 1.4-percent 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. These events included those that use the Revised Thermal Design Procedure (RTDP) methodology (Reference 8-7):

- RCCA Drop (UFSAR Section 14.1.4)
- Locked Rotor Accident (UFSAR Section 14.1.6) – DNB Analysis
- Loss of Reactor Coolant Flow (UFSAR Section 14.1.6) – Partial and Complete Loss of Flow Analyses

The following UFSAR non-LOCA analyses are currently analyzed with an explicit 2-percent power measurement uncertainty allowance that already bounds operation at the 1.4-percent uprate power level with the reduced power measurement uncertainty of 0.6 percent:

- Locked Rotor Accident (UFSAR Section 14.1.6.5) – Overpressure, Maximum Clad Temperature, and Maximum Zirconium-Water Reaction Analysis

- Loss of External Electrical Load (UFSAR Section 14.1.8) – Overpressure Analysis
- Loss of Normal Feedwater (UFSAR Section 14.1.9)
- Loss of All AC Power to the Station Auxiliaries (UFSAR Section 14.1.12)
- Rupture of a Control Rod Mechanism Housing (RCCA Ejection) (UFSAR Section 14.2.6) – Full Power Analysis

The small changes in the plant initial operating conditions resulting from the 1.4-percent power uprate were evaluated, and it was determined that the current analyses of record for these events remain valid for the 1.4-percent power uprate conditions (i. e., these analyses are unaffected).

The following UFSAR non-LOCA events are also either bounded by the current respective analyses of record, or simply are not affected because they are performed starting at hot zero power (HZP) or a power less than 100 percent:

- Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition (UFSAR Section 14.1.1)
- CVCS Malfunction (UFSAR Section 14.1.5)
- Excessive Heat Removal Due to Feedwater System Malfunctions (UFSAR Section 14.1.10) – Zero Power Analysis
- Excessive Load Increase Incident (UFSAR Section 14.1.11)
- Rupture of a Steam Pipe (UFSAR Section 14.2.5) – Zero Power Analysis
- Rupture of a Control Rod Mechanism Housing (RCCA Ejection) (UFSAR Section 14.2.6) – Zero Power Analysis
- Anticipated Transients Without Scram (ATWS)

The following event is prohibited by the IP2 Technical Specifications and is unaffected:

- Startup of an Inactive Reactor Coolant Loop (UFSAR Section 14.1.7)

8.3.1 Design Operating Parameters and Initial Conditions

The design operating parameters that were used as a basis for the evaluations and analyses performed to support the 1.4-percent power uprate are given in Table 2-1.

For accident analyses that are performed to demonstrate that the DNB acceptance criteria are met, nominal values of initial conditions are assumed. In accordance with the RTDP methodology, uncertainty allowances on power, temperature, and pressure are considered in the convolution of uncertainties to statistically establish the DNB ratio (DNBR) limit.

For accident analyses that are not DNB limited, or in which RTDP is not utilized, the initial conditions assumed in the analysis include the maximum steady-state uncertainties applied in the direction that yields the more limiting analysis results.

The only uncertainty modified as a result of the 1.4-percent power uprate is the power measurement uncertainty, which now is ± 0.6 percent. All of the other uncertainties (i.e., average RCS temperature, pressurizer pressure, and RCS flow) were unaffected.

The effect of the revised power measurement uncertainty has been accounted for in the analyses and evaluations of the various non-LOCA accidents discussed herein. For analyses that utilize the RTDP method for the calculation of the minimum DNBR, the uncertainties are accounted for in the minimum DNBR safety analysis limit rather than being accounted for explicitly in the analyses.

8.3.2 Core Limits and Overtemperature and Overpower ΔT Setpoints

The overtemperature ΔT (OTΔT) and overpower ΔT (OPΔT) reactor trip function setpoints are assumed in a number of the non-LOCA safety analyses to ensure that the DNB design basis and the fuel centerline melting design basis are satisfied. The OTΔT and OPΔT reactor trip setpoints are generated assuming steady-state conditions, as described in WCAP-8745 (Reference 8-8). They are based on a number of inputs, which include the nominal core thermal power and the core thermal limits. The core thermal limits are the locus of core inlet temperature conditions for a range of powers and for a range of pressures, which ensure that the DNB design basis is satisfied.

The 1.4-percent increase in core thermal power results in a change to the core thermal limits, as a new higher nominal power must be accounted for. The core thermal limits are provided as a fraction of the nominal power level. Using the 1.4-percent power uprate nominal core thermal power and the revised set of core thermal limits, it was determined that the current OTΔT and OPΔT setpoints did not need to be modified to accommodate the 1.4-percent increase in core thermal power.

The effect of the change in the core thermal limits on the non-LOCA analyses was addressed as part of the evaluations and analyses described in the following subsections.

8.3.3 Affected Non-LOCA Events Re-analyzed for the 1.4-Percent Power Uprate

As shown in Table 8-1, 3 of the IP2 non-LOCA events have been analyzed in support of the 1.4-percent power uprate. Each of the analyses specifically models the increased power level. These events and analyses are described in the following 3 subsections of this report.

8.3.3.1 Uncontrolled RCCA Bank Withdrawal at Power (UFSAR Section 14.1.2)

An uncontrolled RCCA bank withdrawal at power that causes an increase in core heat flux may result from operator error or a malfunction in the rod control system. Immediately following the initiation of the accident, the steam generator heat removal rate lags behind the core power generation rate until the steam generator pressure reaches the setpoint of the steam generator relief or safety valves. This imbalance between heat removal and heat generation rate causes the reactor coolant temperature to rise. Unless terminated, the power mismatch and resultant coolant temperature rise could eventually result in DNB and/or fuel centerline melt. Therefore, to avoid damage to the core, the reactor protection system (RPS) is designed to automatically terminate any such transient before the DNBR falls below the safety analysis limit value or the fuel rod linear heat generation rate (kW/ft) limit is exceeded.

The automatic features of the RPS that prevent core damage in a RCCA bank withdrawal incident at power include the following:

- The power range high neutron flux instrumentation initiates a reactor trip on neutron flux if two-out-of-four channels exceed an overpower setpoint.

- A reactor trip is initiated if any two-out-of-four OTΔT channels exceed an OTΔT setpoint. This setpoint is automatically varied with the axial power distribution, coolant temperature, and pressure to protect against DNB.

- A reactor trip is initiated if any two-out-of-four OPΔT channels exceed an OPΔT setpoint. This setpoint is automatically varied with the coolant temperature, so that the allowable heat generation rate (kW/ft) is not exceeded.

- A high pressurizer pressure reactor trip, actuated from any two-out-of-three pressure channels, is set at a fixed point. This reactor trip on high pressurizer pressure occurs at a pressure that is less than the set pressure for the pressurizer safety valves.

- A high pressurizer water level reactor trip is initiated if any two-out-of-three level channels exceed a fixed setpoint.

The high neutron flux, OTΔT, and high pressurizer pressure reactor trip functions provide adequate protection over the entire range of possible reactivity insertion rates. The minimum value of DNBR is always larger than the safety analysis limit value, and the peak Main Steam System (MSS) pressure is maintained below 110 percent of the design pressure. The RCCA bank withdrawal at power analysis described in UFSAR Section 14.1.2 was evaluated for the 1.4-percent power uprate and remains bounding for peak RCS pressure.

Table 8-2 provides details concerning the key input assumptions, methodology, safety analyses limits, and calculated results for this IP2 analysis.

Since all applicable acceptance criteria continue to be met for the 1.4-percent power uprate, this event will not adversely affect the IP2 core, the RCS, or the MSS.

Table 8-2 Transient: Uncontrolled Control Rod Assembly Withdrawal at Power	
Related UFSAR Section(s)	14.1.2
Key Inputs	• Initiating event: RCCA bank withdrawal • A spectrum of reactivity insertion rates ranging from 1 pcm/sec to 110 pcm/sec were examined at 10%, 60%, and 100% of nominal power in order to demonstrate that the applicable acceptance criteria, namely the minimum DNBR safety analysis limit, are satisfied over a wide range of conditions. • Both maximum and minimum reactivity feedback conditions were examined. • A conservatively high OTΔT reactor protection setpoint was assumed [K1 (constant term in OTΔT setpoint equation) = 1.40. • A conservatively high neutron flux reactor protection setpoint of 118% of uprated rated thermal power was assumed.
Methodology	The applied methodology is consistent with the current licensing basis analysis presented in the UFSAR supporting the Stretch Rating / Optimized Fuel Assembly (OFA). As the LOFTRAN code was utilized in the analysis, the Westinghouse LOFTRAN methodology described in WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Non-Proprietary), "LOFTRAN Code Description," T. W. T. Burnett, et al., April 1984 was applied. The Westinghouse reload safety evaluation methodology described in WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," F. M. Bordelon, et al., July 1985 was also applied.
Safety-Analysis Limits	The DNBR safety-analysis limit (SAL) is 1.58 for the 1.4% power uprate program, corresponding to the WRB-1 DNBR correlation. In comparison, the DNBR SAL for the Stretch Rating / OFA program was 1.52, corresponding to the WRB-1 DNBR correlation. The peak primary and secondary pressure limits are 110% of design pressure, or 2748.5 psia and 1208.5 psia, respectively. There is a 118% limit for peak core average heat flux to preclude fuel centerline melt.

Table 8-2 Transient: Uncontrolled Control Rod Assembly Withdrawal at Power (cont.)	
Calculated Results	• For the 1.4% power uprate, the minimum DNBR calculated using LOFTRAN is 1.7222 and corresponds to a case initiated from 100% power assuming minimum reactivity feedback conditions and a reactivity insertion rate of 2.6 pcm/sec. • The peak secondary pressure calculated for the 1.4% power uprate is 1168.1 psia. • The peak core average heat flux calculated for the 1.4% power uprate is 116.9%. • The peak primary pressure calculated for the 1.4% power uprate is 2728.5 psia.

8.3.3.2 Loss of External Electrical Load-DNB Analysis (UFSAR Section 14.1.8)

The loss of external electrical load and/or turbine trip event is defined as a complete loss of steam load or a turbine trip from full power without a direct reactor trip. This event is analyzed as a turbine trip from full power as this bounds both the loss of external electrical load and turbine trip events. The turbine trip is more severe than the total loss of external electrical load event because it results in a more rapid reduction in steam flow.

With respect to pressure effects, the turbine trip event is more limiting than any other partial or complete loss-of-load event, since it results in the most rapid reduction in steam flow. This causes the most limiting increase in pressure and temperature in the RCS and the MSS, due to the very rapid decrease in secondary steam flow.

The analysis conservatively assumes that the reactor trip is actuated by the RPS and not by the turbine trip signal. This assumption is made because the UFSAR analysis is done to show that the safety-grade RPS signals are capable of providing a reactor trip in sufficient time following the event initiation, to satisfy the acceptance criteria for the event, and to conservatively bound the other events listed above.

For the event analyzed, the reactor may be tripped by any of the following RPS trip signals:

- OPΔT
- OTΔT
- Pressurizer high pressure
- Low-Low steam generator water level

In the event that the steam dump valves fail to open following a large loss of load, the sudden reduction in steam flow results in an increase in pressure and temperature in the steam generator secondary side. As a result, the heat transfer rate in the steam generator is reduced, causing the reactor coolant temperature to rise. This causes coolant expansion, a pressurizer insurge, and a rise in RCS pressure. Throughout the event, power is available for the continued operation of plant components, such as the reactor coolant pumps.

Unless the transient RCS response to the turbine trip event is terminated by manual or automatic action, the resultant reactor coolant temperature rise could eventually result in DNB and/or the resultant pressure increases could challenge the integrity of the reactor coolant pressure boundary, or the MSS pressure boundary. To avoid the potential damage that might otherwise result from this event, the RPS is designed to automatically terminate any such transient before the DNBR falls below the safety-analysis limit value, and before the RCS and/or MSS pressures exceed the values at which the integrity of the pressure boundaries would be jeopardized.

The most significant potential effects associated with the turbine trip are overpressurization of the RCS and MSS, and possible fuel cladding damage resulting from the increase in RCS temperature.

The transient responses for a turbine trip from full-power conditions are presented in the IP2 UFSAR as two cases: first, for the scenario with pressurizer pressure control, and second, where pressurizer pressure control is assumed to not be available. Both cases assume minimum reactivity feedback conditions.

Only the case in which pressurizer pressure control is assumed to be available (the case in which the DNB design basis is examined) was considered in support of the 1.4-percent power uprate. The case in the licensing-basis analysis in which pressurizer pressure control is assumed to be unavailable (the case in which RCS and MSS overpressure criteria are examined) is not impacted by an increase in the nominal full power because the power level assumed in the current analysis for this case (with 2-percent uncertainty) is equivalent to that based upon the uprated power of 3127 MWt, combined with the lower uncertainty of 0.6 percent.

As stated earlier in this section, the turbine trip analysis bounds the total loss of electrical load event for IP2 because it results in a more rapid reduction in steam flow. Therefore, the analysis documented in this section bounds both a complete loss of steam load, and a turbine trip from full power without a direct reactor trip.

The results of this analysis (the case analyzed with pressurizer pressure control available) demonstrate that the fuel design limits continue to be maintained by the RPS, since the DNBR is maintained above the safety-analysis limit value.

Table 8-3 provides details concerning the key input assumptions, methodology, safety-analyses limits, and calculated results for the loss of electrical load analysis.

Since all applicable acceptance criteria continue to be met for the 1.4-percent power uprate, this event will not adversely affect the IP2 core, RCS, or MSS.

Table 8-3 Transient: Loss of External Electrical Load	
Related UFSAR Section(s)	14.1.8
Key Inputs	• Initiating event: Turbine Trip • A conservatively high OTΔT reactor protection setpoint was assumed [K1 (constant term in OTΔT setpoint equation) = 1.40 (1.4% uprate and OFA fuel transition/stretch rating)]. • The pressurizer sprays and power-operated relief valves are assumed to be available. • Least-negative moderator temperature coefficient (0.0 pcm/°F). • Least-negative Doppler power defect.
Methodology	The applied methodology is consistent with the current licensing basis analysis presented in the UFSAR supporting the Stretch Rating / OFA. As the LOFTRAN code was utilized in the analysis, the Westinghouse LOFTRAN methodology described in WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Non-Proprietary), "LOFTRAN Code Description," T. W. T. Burnett, et al., April 1984 was applied. The Westinghouse reload safety evaluation methodology described in WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," F. M. Bordelon, et al., July 1985 was also applied.
Safety-Analysis Limits	The minimum DNBR SAL is 1.58 for the 1.4% power uprate program, corresponding to the WRB-1 DNBR correlation. In comparison, the DNBR SAL for the Stretch Rating/OFA RTSR Program was 1.52, corresponding to the WRB-1 DNBR correlation.
Calculated Results	• For the 1.4% power uprate, the minimum DNBR calculated using LOFTRAN is 2.04 [1].

Note:

1. The DNBR value corresponding to nominal conditions is used in the calculation of DNBR values during transient conditions. For the loss-of-load (LOL)/turbine trip analysis associated with the OFA fuel transition/stretch rating, the nominal DNBR value that was used in calculating the transient DNBR values was that corresponding to STANDARD fuel. This value was used because nominal DNBR values for STANDARD fuel are lower than nominal DNBR values calculated for OFA fuel at the same conditions and using the lower DNBR value corresponding to the STANDARD fuel bounded the transition cores. Also, the LOL/turbine trip event was evaluated for the VANTAGE + fuel transition and, thus, the results associated with the OFA fuel transition/stretch rating remained applicable for the VANTAGE + fuel transition. However, since the transition to VANTAGE+ fuel is now complete at IP2, the nominal DNBR value that was used in calculating the transient DNBR values for the 1.4% power uprate corresponds to VANTAGE + fuel and is higher than that used in the OFA fuel transition/stretch rating calculations (i.e., STANDARD fuel DNBR value). Therefore, the minimum DNBR values that were calculated for the LOL/turbine trip transient for the 1.4% power uprate improved when compared to those values calculated for this transient for the OFA fuel transition/stretch rating and the VANTAGE + fuel transition.

8.3.3.3 Excessive Heat Removal Due to Feedwater System Malfunctions (UFSAR Section 14.1.10)

The analysis for this event results from an increase in primary-to-secondary heat transfer caused by an increase in feedwater flow that can result in the primary-system temperature and pressure decreasing significantly. The negative moderator and fuel temperature reactivity coefficients and the actions initiated by the reactor rod control system can cause core reactivity to rise as the primary-system temperature decreases. In the absence of a RPS reactor trip or other protective action, this increase in core power coupled with the decrease in primary-system pressure can challenge the core thermal limits.

An increase in feedwater flow can be caused by a failure in the feedwater control system or an operator error that leads to the simultaneous full opening of a feedwater control valve. At power, this excess flow causes a greater load demand on the primary system due to increased subcooling in the steam generator. With the plant at zero-power conditions, the addition of relatively cold feedwater may cause a decrease in primary-system temperature and, thus, a reactivity insertion due to the effects of the negative moderator temperature coefficient.

Transients initiated by increases in feedwater flow are attenuated by the thermal capacity of the primary and secondary systems. If the increase in reactor power is large enough, the primary RPS trip functions (e.g., high neutron flux, OTΔT, or OPΔT) will prevent any power increase that can lead to a DNBR less than the safety-analysis limit value. The RPS trip functions may not actuate if the increase in power is not large enough.

The analysis presented herein is for the excessive heat removal due to feedwater system malfunctions event of UFSAR Section 14.1.10. The feedwater system malfunction that causes a reduction in feedwater temperature continues to be bounded by the excessive increase in secondary steam flow event.

The maximum feedwater flow to one steam generator due to a control system malfunction that causes the feedwater control valves to fail in the full-open position is assumed. Cases with and without automatic rod control initiated at hot full-power conditions were considered in support of the 1.4-percent power uprate. The licensing-basis analysis also addresses cases that are initiated at HZP conditions, but these are not impacted by an increase in the nominal full-power rating. Thus, the conclusions of the feedwater malfunction analysis at HZP conditions continue to remain valid for the 1.4-percent uprate.

The results of the analysis show that the minimum DNBR calculated for an excessive feedwater addition at power is above the safety-analysis limit value. Therefore, the DNB design basis is met. With regard to the RCS and MSS overpressure criteria, this event is bounded by the turbine trip analysis documented in Section 8.3.2.2.

Table 8-4 provides analysis results for the transient excessive heat removal due to feedwater system malfunctions.

As all applicable acceptance criteria have been satisfied, the failure of any of the feedwater control valves will not challenge the RCS and MSS pressure boundaries, nor will the integrity of the fuel cladding be compromised due to DNB.

Table 8-4 Transient: Excessive Heat Removal Due to Feedwater System Malfunctions	
Related UFSAR Section(s)	14.1.10
Key Inputs	• Initiating event: accidental opening of one feedwater control valve with the reactor at full power. This results in a feedwater flow increase to 120% of nominal flow to one steam generator. • Two cases were examined: one with the rod control system in automatic mode and one with the rod control system in manual mode. • Initial steam generator water level was set to the value that corresponds to the nominal level [52% narrow range span (NRS)]. • The High-High steam generator water level turbine trip setpoint was conservatively maximized at 100% NRS. • Most-negative moderator and Doppler temperature coefficients. • Least-negative Doppler power defect.
Methodology	The applied methodology is consistent with the current licensing basis analysis presented in the UFSAR supporting the stretch rating/OFA RTSR. As the LOFTRAN code was utilized in the analysis, the Westinghouse LOFTRAN methodology described in WCAP-7907-P-A (Proprietary) and WCAP-7907-A (Non-Proprietary), "LOFTRAN Code Description," T. W. T. Burnett, et al., April 1984 was applied. The Westinghouse reload safety evaluation methodology described in WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," F. M. Bordelon, et al., July 1985 was also applied.
Safety-Analysis Limits	The minimum DNBR SAL is 1.58 for the 1.4% power uprate program, corresponding to the WRB-1 DNBR correlation. In comparison, the DNBR SAL for the Stretch Rating/OFA RTSR Program was 1.52, corresponding to the WRB-1 DNBR correlation.
Calculated Results	The minimum DNBR values calculated using LOFTRAN for the two cases are listed as follows: • Automatic rod control - 2.306 (1.4% uprate) [1] - 2.101 (Stretch rating/OFA RTSR) • Manual rod control - 2.306 (1.4% uprate) [1] - 2.101 (Stretch rating/OFA RTSR)

Table 8-4 Transient: Excessive Heat Removal Due to Feedwater System Malfunctions (cont.)
Note: 1. The DNBR value corresponding to nominal conditions is used in the calculation of DNBR values during transient conditions. For the loss-of-load (LOL)/turbine trip analysis associated with the OFA fuel transition/stretch rating, the nominal DNBR value that was used in calculating the transient DNBR values was that corresponding to STANDARD fuel. This value was used because nominal DNBR values for STANDARD fuel are lower than nominal DNBR values calculated for OFA fuel at the same conditions and using the lower DNBR value corresponding to the STANDARD fuel bounded the transition cores. Also, the LOL/turbine trip event was evaluated for the VANTAGE + fuel transition and, thus, the results associated with the OFA fuel transition/stretch rating remained applicable for the VANTAGE + fuel transition. However, since the transition to VANTAGE+ fuel is now complete at IP2, the nominal DNBR value that was used in calculating the transient DNBR values for the 1.4% power uprate corresponds to VANTAGE + fuel and is higher than that used in the OFA fuel transition/stretch rating calculations (i.e., STANDARD fuel DNBR value). Therefore, the minimum DNBR values that were calculated for the LOL/turbine trip transient for the 1.4% power uprate improved when compared to those values calculated for this transient for the OFA fuel transition/stretch rating and the VANTAGE + fuel transition.

8.3.4 Affected Non-LOCA Events Evaluated for the 1.4-Percent Power Uprate Using Existing DNB Margin

8.3.4.1 Rod Cluster Control Assembly Drop (UFSAR Section 14.1.4)

An RCCA misalignment includes the following events:

- One or more dropped RCCAs within the same group
- A dropped RCCA bank
- A statically misaligned RCCA

The dropped RCCA transients (including the dropped RCCA bank) were previously analyzed using the methodology described in WCAP-11394 (Reference 8-9), with the application of turbine runback features for the plant, and are examined to demonstrate that the DNB design basis is met.

The methodology described in WCAP-11394 is applicable for both generic (non-turbine runback) statepoint generation and statepoint generation specific for turbine runback plants. Sensitivity studies on the effect of a power increase on the generic statepoints were previously performed for a 4-loop plant. The studies quantified the effect of an approximate 5-percent increase in power on the 4-loop generic statepoints, and found that the statepoints were still applicable for use at the 1.4-percent power uprate conditions. Though the 4-loop plant statepoint analysis used in the power increase sensitivity study is not a turbine runback plant, the evaluation concluded that the impact of a power uprate on the turbine runback statepoints will have the same non-adverse impact. Therefore, since the IP2 1.4-percent power uprate is much smaller than the uprate used in the sensitivity studies (approximately 5 percent), both the generic statepoints (non-turbine runback) and turbine runback statepoints continue to be applicable to IP2.

Although the statepoints are unaffected, the increase in nominal core heat flux must be addressed with respect to the calculated DNBR. An evaluation of the DNB design basis using the generic statepoints and increased nominal core heat flux confirmed that the DNB design basis continues to be met. Therefore, all applicable acceptance criteria continue to be met for the 1.4-percent power uprate.

8.3.4.2 Loss of Reactor Coolant Flow (UFSAR Section 14.1.6)

The loss of reactor coolant flow events include the partial and complete loss of forced reactor coolant flow events and the reactor coolant pump (RCP) shaft seizure (locked rotor) and RCP shaft break events. The following subsections describe the details of the evaluations completed for these events.

8.3.4.2.1 Loss of Reactor Coolant Flow-Partial and Complete Loss of Forced Reactor Coolant Flow

The partial and complete loss of forced reactor coolant flow events may result from mechanical or electrical failure(s) in the RCPs. These faults may occur from an undervoltage condition in the electrical supply to the RCPs or from a reduction in motor supply frequency to the RCPs due to a frequency disturbance on the power grid. These analyses demonstrate that the minimum DNBR remains above the safety-analysis limit value. The limiting results are obtained at full-power conditions and occur very quickly following initiation of the event.

Since the 1.4-percent power uprate could potentially affect the minimum DNBR, an evaluation was completed for this event. The evaluation concluded that the existing statepoints for the limiting complete loss of forced reactor coolant flow event remain valid, with the exception of the nominal core heat flux. The nominal core heat flux increases due to the 1.4-percent power uprate. Therefore, the higher nominal core heat flux must be applied to the power statepoints, which are fractions of the nominal value.

Revised statepoints that included the increased nominal heat flux were evaluated with respect to DNBR. The analysis showed that the DNB design basis remains satisfied. Therefore, all applicable acceptance criteria continue to be met for the 1.4-percent power uprate.

8.3.4.2.2 Locked Rotor Accident – DNB Analysis

A single RCP shaft seizure (locked rotor) event is based on the sudden seizure of an RCP impeller, or failure of an RCP shaft. A reactor trip via the low RCS flow protection function terminates this event very quickly. Since the 1.4-percent power uprate could potentially affect the minimum DNBR, an evaluation was completed to confirm that no rods violate the DNBR limit. The evaluation concluded that the existing statepoints for this event remain valid with the exception of the nominal core heat flux, which increases due to the uprate. Therefore, the higher nominal core heat flux must be applied to the power statepoints, which are fractions of the nominal value.

Revised statepoints that included the increased nominal heat flux were evaluated with respect to the rods-in-DNB limit. The analysis showed that the DNB design basis remains satisfied.

Therefore, all applicable acceptance criteria continue to be met for the 1.4-percent power uprate.

8.3.4.2.3 Results and Conclusions

Section 8.3.4.2 indicates that loss of flow and RCP shaft seizure events were evaluated with respect to DNBR. The analysis showed that the DNBR design basis remains satisfied. The following discussion provides additional details of these evaluation/analysis including the calculated minimum DNBR for these events.

The loss of reactor coolant flow analyses yield statepoints for core heat flux, core mass flow rate, pressurizer pressure, and core inlet temperature. The core heat flux and mass flowrate statepoints are presented as fraction of nominal (FON) values. A sensitivity study performed by Westinghouse showed that for a 1.4-percent power uprate, there is a negligible effect on loss of reactor coolant flow statepoint values, including the core heat flux and core mass flow FON values. Only the actual nominal core heat flux would increase as a result of the 1.4-percent uprate. Therefore, the limiting loss of reactor coolant flow events were re-analyzed using the same statepoints, but with an increased nominal core heat flux.

The minimum DNBR values that were calculated for the complete loss of flow and RCP shaft seizure analyses, based on the WRB-1 DNB correlation, were 1.762 and 1.536, respectively. Compared to the minimum DNBR values calculated for the same statepoints with a current nominal core heat flux (based on a core power of 3071.4 MWt), the minimum DNBR values became slightly worse. However, the decrease in the minimum DNBR values was accommodated by existing margin to the DNBR safety-analysis limit, and the results are still maintained above the DNBR safety-analysis limit.

8.3.5 Non-LOCA Events Bounded by Current 102-Percent Power Assumption

8.3.5.1 Locked Rotor Accident - Overpressure, Maximum Clad Temperature, and Maximum Zirconium-Water Reaction Analysis (UFSAR Section 14.1.6.5)

The case completed to confirm that the RCS pressure, cladding temperature, and Zirconium-water reaction criteria are met was not re-analyzed, since it currently models a 2-percent power uncertainty. As such, the RCS pressure, cladding temperature, and Zirconium-water reaction criteria continue to be met for the locked rotor event.

8.3.5.2 Loss of External Electrical Load – Overpressurization Analysis (UFSAR Section 14.1.8)

Only the case in which pressurizer pressure control is assumed to be available (the case in which the DNB design basis is examined) was addressed to support the 1.4-percent power uprate. The case in the licensing-basis analysis in which pressurizer pressure control is assumed to be unavailable (the case in which the RCS and MSS overpressure criteria are examined) 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-percent uncertainty) is equivalent to that based upon the uprated power of 3115 MWt, combined with the lower uncertainty of 0.6 percent and remains bounding for the 1.4-percent power uprate. Therefore, the results of the case analyzed without pressurizer pressure control available is unaffected and the turbine trip presents no hazard to the integrity of the RCS or MSS pressure boundary.

8.3.5.3 Loss of Normal Feedwater and Loss of all AC Power to the Station Auxiliaries (UFSAR Sections 14.1.9 and 14.1.12)

Both the current loss of normal feedwater (LONF) and loss of all AC (LOAC) power analyses of record already model a 2-percent power uncertainty allowance. Therefore, these analyses are unaffected by the 1.4-percent power uprate.

Additional analyses, not related to the uprate, have recently been performed to support an increase in RCS T_{avg} to 562°F, an updated pressurizer level uncertainty of 6 percent, and a conservative steam generator tube plugging (SGTP) assumption of 0 percent. These analyses continue to support the 1.4-percent power increase conditions, and the results of these analyses continue to demonstrate that the acceptance criteria are met.

8.3.5.4 Rupture of a Control Rod Mechanism Housing – Full Power Analysis (UFSAR Section 14.2.6)

The rupture of a control rod drive mechanism (CRDM) housing (RCCA ejection) event is the result of the assumed mechanical failure of a CRDM pressure housing such that the RCS would eject an RCCA and drive shaft to the fully withdrawn position. The transient responses for the hypothetical RCCA ejection event are analyzed at beginning of life (BOL) and end of life (EOL), for both full [hot full-power (HFP)] and zero (HZP) operation, in order to bound the entire fuel cycle and expected operating conditions. The current analyses of record were performed to show that the fuel and cladding limits are not exceeded. Since this analysis is not performed to evaluate the minimum DNBR, the RTDP method is not utilized. (The limiting fuel rod is conservatively assumed to undergo DNB very early in the transient, thus maximizing the fuel temperature response.)

The current HFP analysis is performed at 102 percent of licensed core power. As such, the increase in core power, combined with the reduction in the power uncertainty, is bounded by the current assumption in the analysis.

The rupture of a CRDM housing event also models the power range neutron flux setpoints, which have not been changed for the 1.4-percent power uprate conditions. However, since these setpoints are fractions of the nominal full-power level, it was necessary to confirm that the acceptance criteria continue to be met.

The effect of the power increase on the reactor trip time was also considered. The high neutron flux trip setpoints modeled in these analyses are 35 and 118 percent for the HZP and HFP cases, respectively. Since the trip setpoints are not changing for this program, the power level at which the unit would now trip during these events will be slightly higher than that which is modeled in the analyses. However, the initial power increase that results from the rod ejection is terminated by reactivity feedback, not rod insertion. The power increases to a peak and is decreasing at a rapid rate by the time that the rods begin to drop. The specific value of the trip setpoint is secondary to other critical parameters, such as the ejected rod worth and Doppler defect, which significantly affect the results of the analysis. The power level increases at a very rapid rate in this analysis, such that the delay in reaching 35 percent (or 118 percent) of the uprated power versus 35 percent (or 118 percent) of the current power would be on the order of milliseconds. The time at which the rods would begin to drop into the core would be virtually unchanged. Since the rod drop time in essentially unaffected, and the initial nuclear power transient is defined by reactivity feedback, the total duration of energy addition would be almost identical. Therefore, the subsequent fuel rod heat flux increase resulting from the energy addition would also be insignificantly different. Therefore, this analysis of record is unaffected and bounds the 1.4-percent power uprate.

8.3.6 Non-Limiting/Bounded Non-LOCA Events

8.3.6.1 Uncontrolled Rod Control Cluster Assembly Withdrawal from a Subcritical or Low Startup Power Condition (UFSAR Section 14.1.1)

This event is defined as an uncontrolled addition of reactivity to the reactor core caused by the withdrawal of one or more RCCA banks, resulting in a rapid power excursion. This transient is promptly terminated by the power range neutron flux low setpoint reactor trip. Due to the inherent thermal lag in the fuel pellet, heat transfer to the RCS is relatively slow. The purpose of the analysis is to demonstrate that the minimum DNBR remains above the safety analysis limit value.

Since the Rod Withdrawal from Subcritical (RWFS) event occurs from a subcritical core condition with the RCS at no-load temperature conditions, this event is unaffected by an increase in the reactor full-power level.

Since the power range neutron flux low setpoint of 35 percent is not changing for the 1.4-percent power uprate, the power level at which the unit would now trip during this event will be slightly higher than that which is modeled in the analysis. However, the initial power increase that results from the rod withdrawal is terminated by reactivity feedback, not rod insertion. The power increases to its peak and is rapidly decreasing by the time the rods begin to drop. The power level increases at a very rapid rate in this analysis, such that the delay in reaching 35 percent of the uprated power versus 35 percent of the current power would be on the order of milliseconds. The time at which the rods would begin to drop into the core would be virtually unchanged. Since the rod drop time is essentially unaffected and the initial nuclear power transient is defined by reactivity feedback, the total duration of energy addition would be almost identical. Therefore, the subsequent fuel rod heat flux increase resulting from the energy addition would also be insignificantly different.

The existing statepoints for the RWFS event remain valid, with the exception of the nominal core heat flux. The nominal core heat flux increases due to the 1.4-percent power uprate. Therefore, the higher nominal core heat flux must be applied to the power statepoints, which are fractions of the initial value.

Revised statepoints that included the increased nominal heat flux were evaluated with respect to DNBR. The analysis showed that the DNB design basis is satisfied. Therefore, this analysis of record is unaffected and bounds the 1.4-percent power uprate.

8.3.6.2 Chemical and Volume Control System Malfunction (UFSAR Section 14.1.5)

The CVCS malfunction (resulting in a boron dilution) analysis is performed to demonstrate that the operator has sufficient time (15 minutes for Modes 1 and 2, 30 minutes for Mode 6) to terminate the RCS dilution before a complete loss of shutdown margin occurs. The critical parameters in the determination of the time available to terminate the dilution include the overall RCS active volume, the RCS fluid density, the dilution flow rate, and the initial and critical boron concentrations. The analysis does not explicitly model or consider the initial power level.

A sensitivity study performed by Westinghouse showed that a 1.4-percent power uprate would increase the reactor trip (due to an OTΔT condition) time by less than 1 second. The assumed reactor trip time is based on an uncontrolled RCCA withdrawal at power analysis in which the reactivity insertion rate is equivalent to that expected for the Mode 1 boron dilution scenario. It was determined that this is applicable to IP2.

An evaluation of the Mode 1 analysis was performed that showed that the 1.4-percent power uprate does not significantly affect the automatic reactor trip time used in the analysis. Since the reactor trip time assumed in the analysis is still valid, the results of the Mode 1 analysis also remain valid. With respect to the Modes 2 and 6 analyses, the increase in full power does not affect the results of these analyses, since the reactor is not at power (or full power for Mode 2). Therefore, this analysis of record is unaffected and bounds the 1.4-percent power uprate.

8.3.6.3 Startup of an Inactive Reactor Coolant Loop (UFSAR Section 14.1.7)

The analysis of record for this event conservatively assumes initial conditions representative of this event, with 3 loops in operation. However, the IP2 Technical Specifications require that all 4 RCS loops are operable and in operation while the reactor is in Modes 1 and 2. This precludes operation at the initial conditions assumed in the current licensing basis analysis. Therefore, the conclusions documented in the UFSAR for this event are not applicable to IP2 operation or operation with the 1.4-percent power uprate.

8.3.6.4 Excessive Heat Removal Due to Feedwater System Malfunction – Zero Power Analysis (UFSAR Section 14.1.10)

Cases with and without automatic rod control initiated at hot full-power conditions were considered in support of the 1.4-percent power uprate. The licensing-basis analysis also addresses cases that are initiated at the HZP conditions, but these are not affected by an increase in the nominal full-power rating. Thus, the conclusions of the feedwater malfunction analysis at the HZP conditions are unaffected and continue to remain bounding for the 1.4-percent power uprate.

8.3.6.5 Excessive Load Increase Incident (UFSAR Section 14.1.11)

This transient is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. Cases are evaluated at BOL and EOL conditions with and without rod control to demonstrate that the DNB design basis is met. The transient response to this accident is relatively mild such that the reactor stabilizes at a new equilibrium condition corresponding to conditions well above that which would challenge the DNBR limit, without generating a reactor trip.

This transient was evaluated by comparing plant conditions, conservatively bounding deviations in core power, average coolant temperature, and RCS pressure to conditions corresponding to those required to exceed the core thermal limits. The evaluation concluded that there is sufficient margin to the core thermal operating limits in each case considered. Thus, since the core thermal limits are not challenged, the minimum DNBR remains above the limit value for all cases. Therefore, this analysis is unaffected by the 1.4-percent power uprate, and the conclusions presented in Section 14.1.11 of the UFSAR remain valid.

8.3.6.6 Rupture of a Steam Pipe (UFSAR Section 14.2.5)

Steam system piping failures, such as ruptures, would result in steam being discharged from the steam generators. This escaping steam would cause an increase in steam flow, which would result in an increase in the heat extraction rate and a consequential reduction in primary-system temperature and pressure. Due to the negative moderator temperature coefficient and fuel temperature reactivity feedback at the end-of-cycle conditions, the core reactivity would increase, as the primary-coolant temperature decreases. If no automatic or manual actions are taken, the core power will eventually rise to a level that corresponds to the increased steam flow rate.

The main steam line rupture event is analyzed at zero-power conditions. This event is analyzed using non-statistical DNB methods, assuming a double-ended guillotine rupture of the main steam line on one steam generator. Uncontrolled steam releases could also result from the inadvertent opening of a steam generator relief valve, steam generator safety valve, or steam dump valve. The steam line rupture event is analyzed to demonstrate that any return to power resulting from the uncontrolled steam release does not result in a violation of the DNB design basis.

Based on the fact that the zero-power steam line rupture event is analyzed using non-statistical DNB methods, and that it is analyzed from a shutdown condition, the analysis results are not affected by the 1.4-percent power uprate. Therefore, the licensing-basis zero-power steam line rupture analysis presented in Section 14.2.5 of the IP2 UFSAR remains valid and bounds the 1.4-percent power uprate. Additionally, the results of the licensing-basis inadvertent opening of a steam generator relief or safety valve event presented in Section 14.2.5 of the IP2 UFSAR remain bounded by the results of the zero-power, double-ended rupture of a main steam line.

8.3.6.7 Rupture of a Control Rod Mechanism Housing (RCCA Ejection) – Zero Power Analysis (UFSAR Section 14.2.6)

The current HZP analysis of record is unaffected since it is performed at 0-percent power. A change in the full-power value does not change the results.

8.3.6.8 Anticipated Transients Without Scram

For Westinghouse designed pressurized water reactors (PWRs), the licensing requirements related to ATWS are those specified in the Final ATWS Rule, 10 CFR 50.62 (b). The requirement set forth in 10 CFR 50.62 (b) is that all Westinghouse designed PWRs must install ATWS Mitigation System Actuation Circuitry (AMSAC). In compliance with 10 CFR 50.62 (b), AMSAC has been installed and implemented at IP2.

As documented in SECY-83-293 (Reference 8-10), the analytical bases for the Final ATWS rule are the generic ATWS analyses for Westinghouse PWRs generated by Westinghouse in 1979. These generic

ATWS analyses were formally transmitted to the NRC via letter NS-TMA-2182 (Reference 8-11) and were performed based on the guidelines provided in NUREG-0460 (Reference 8-12).

In the generic ATWS analyses documented in NS-TMA-2182, ATWS analyses were performed for the various American Nuclear Society (ANS) Condition II events (i.e., anticipated transients) considering various Westinghouse PWR configurations applicable at that time. These analyses included 2-, 3-, and 4-loop PWRs with various steam generator models. For IP2, the generic ATWS analyses applicable at that time are those for a 4-loop PWR with Model 44 steam generators and a core power of 3025 MWt. These conditions are summarized in Table 3-1-d of NS-TMA-2182. For this plant configuration, the peak RCS pressure reported in NS-TMA-2182 for the limiting loss-of-load ATWS event is 2979 psia.

The IP2 is currently licensed to an NSSS power of 3083.4 MWt (core power of 3071.4 MWt) and operates with replacement Westinghouse Model 44F steam generators. Since the replacement steam generators have similar operating conditions to the original Model 44 design and do not impact results of the safety analyses, the generic ATWS analyses documented in NS-TMA-2182 continue to appropriately reflect the current plant configuration for IP2.

The generic ATWS analyses documented in NS-TMA-2182 also support the analytical basis for the NRC approved generic AMSAC designs generated for the Westinghouse Owners Group (WOG) as documented in WCAP-10858P-A, Revision 1 (Reference 8-13). For the purpose of these AMSAC designs, the generic ATWS analyses for the 4-loop PWR configuration with Model 51 steam generators were used to conservatively represent all of the various Westinghouse PWR configurations contained in NS-TMA-2182. For IP2, WCAP-10858P-A AMSAC Logic 2, AMSAC Actuation on Low Main Feedwater Flow, has been employed.

For the subject power uprating, the power increase will result in a maximum NSSS power of 3127 MWt. This NSSS power is 3.4-percent higher than the power level of 3025 MWt considered in the generic ATWS analyses for 4-loop PWRs with Model 44 steam generators. Operating at a higher power level will result in a higher peak RCS pressure condition following the pressure limiting ATWS events. As documented in NS-TMA-2182, the 4-loop PWR model with Model 51 steam generators showed a peak RCS pressure penalty (increase) of 44 psia with a 2-percent power increase in the limiting loss-of-load ATWS event. Based on this sensitivity, an NSSS power 3.4-percent higher than the generic analysis will be a 75 psia pressure penalty in the peak RCS pressure calculated in the limiting loss-of-load ATWS event.

As prescribed by NUREG-0460, the 1979 generic ATWS analyses for Westinghouse PWRs documented in NS-TMA-2182 assumed a full-power moderator temperature coefficient (MTC) of -8pcm/°F. A sensitivity analysis including the use of an MTC of -7pcm/°F was also provided as prescribed by NUREG-0460. In 1979, the MTC values of -8pcm/°F and -7pcm/°F represented MTCs that Westinghouse PWRs would be more negative than for 95 percent and 99 percent of the cycle, respectively. The base case of 95 percent represents a 95-percent confidence limit on favorable MTC for the fuel cycle. For IP2, the Technical Specification requirement on MTC is limited to < 0pcm/°F at all power levels. The current MTC Technical Specification for IP2 remains the same as that which was applicable for most Westinghouse PWRs in 1979. Therefore, the reactivity feedback for IP2 remains sufficiently negative to be comparable to the generic Westinghouse ATWS analyses presented in NS-TMA-2182.

Relative to the other conditions important to the ATWS analyses, the pressurizer power-operated relief valve (PORV) relief capacity, safety valve relief capacity, and auxiliary feedwater (AFW) capacity is unaffected by the proposed 1.4-percent power uprate as documented in Section 6 of this report. The design capacity of each IP2 pressurizer PORV (179,000 lbm/hr) and pressurizer safety relief valve (408,000 lbm/hr) are consistent with the relief capacities assumed in the 1979 generic ATWS analysis for this plant configuration.

For IP2, the design capacities of the AFW pumps are as follows:

- Motor-driven AFW pump - 400 gpm
- Turbine-driven AFW pump - 800 gpm

The IP2 AFW system has 2 motor-driven AFW pumps (each pump aligned to 2 steam generators) and 1 turbine-driven AFW pump (aligned to all 4 steam generators). Therefore, the total design capacity of the IP2 AFW System is 1600 gpm. This is 160 gpm less, or approximately 91 percent of the total AFW System capacity of 1760 gpm, assumed in the 1979 generic ATWS analyses for the Westinghouse 4-loop plant configuration with Model 44 steam generators (as documented in NS-TMA-2182). As reported for the generic 4-loop PWR with Model 51 steam generators, reducing the AFW by 10 percent (i.e., 176 gpm) increases peak RCS pressure in the limiting loss-of-load ATWS event by 12 psia.

Therefore the current conditions of IP2 with a 1.4-percent power uprate results in a 3.4-percent higher reactor power and a conservatively assumed 10-percent lower AFW flow rate than that assumed in the generic limiting loss of load ATWS event applicable for IP2. The higher reactor power and lower AFW flow result in a combined overall peak RCS pressure penalty (increase) of 87 psia (75 psia + 12 psia) relative to the peak RCS pressure of 2979 psia reported in the generic ATWS analysis. This results in a net peak RCS pressure of 3066 psia (i.e., 2979 psia + 87 psia), or a margin to the ATWS peak RCS pressure limit of 3200 psia of 134 psia (i.e., 3200 – 3066 psia).

Based on the above, it is concluded that operation of IP2 at an uprated NSSS power of 3127 MWt remains within the bounds of the generic Westinghouse ATWS analysis documented in NS-TMA-2182 and, therefore, would remain in compliance with the Final ATWS Rule, 10 CFR 50.62 (b).

8.4 CONTAINMENT INTEGRITY

8.4.1 Long-Term Steam Line Break Mass and Energy Releases Inside Containment Evaluation

The licensing-basis safety analysis of record related to the steam line break mass and energy releases was previously performed at a 1.4-percent power uprate on IP2. The NSSS design parameters for IP2, as shown in Table 2-1, remain unchanged or bounded by the current safety analysis values.

The critical parameters for the long-term steam line break event include the following conditions on the primary and secondary sides: NSSS power level, reactivity feedback characteristics including the minimum shutdown margin, the initial value for the steam generator water mass, main feedwater flow, AFW flow, and the time at which feed line isolation occurs. The input assumptions related to these critical parameters dictate the quantity and rate of the mass and energy releases.

The 1.4-percent power uprate analysis of record included a 0.6-percent power calorimetric power uncertainty. Therefore, the current licensing-basis long-term steam line break mass and energy release analysis of record and the steam line break containment integrity analysis of record are unaffected by the 1.4-percent power uprate.

Although not related to the 1.4-percent uprate, an additional analysis not crediting closure of isolation valves associated with the feedwater regulating valves (BFD-5) has recently been performed. The analysis results demonstrate continued compliance with the containment pressure limit of 47 psig and the results do not change as a result of the 1.4-percent uprate conditions.

8.4.2 LOCA Mass and Energy Releases and Containment Integrity (UFSAR Section 14.3.5)

The licensing-basis safety analyses related to the LOCA mass and energy releases and containment integrity were evaluated to determine the effect of the 1.4-percent power uprate. These analyses demonstrate the ability of the containment safeguards systems to mitigate the consequences of a hypothetical LBLOCA. The most limiting LOCA long-term mass and energy release calculation of record was performed using the NRC-approved methodology documented in WCAP-10325-P-A. The containment response analysis of record has been performed using the COCO computer code as described in IP2 UFSAR Section 14.3.5.

The analyses of record presently assume a core thermal power of 3071.4 MWt, plus an additional 2-percent power measurement uncertainty allowance. Since the resulting power level used in the current analysis of record is the same as that which results from the 1.4-percent power uprate with an ~ 0.6-percent power measurement uncertainty allowance, the current analyses of record are unaffected and bound the 1.4-percent power uprate.

The following IP2 LOCA containment integrity analysis criteria were considered in the evaluation:

- Peak pressure ≤ 43 psig (43 psig is the current analysis of record calculated result, the design limit is 47 psig)
- Peak steam temperature ≤ 263°F (263°F is the current analysis of record calculated result)
- Containment pressure at 24 hours < 50 percent of the peak pressure
- Containment liner temperature ≤ 247°F
- Sump maximum water temperature ≤ 265°F (265°F is the current analysis of record result, the design limit is 275°F)
- Maximum air density for the reactor containment fan cooler loads ≤ current analysis of record

8.4.3 Short-Term LOCA Mass and Energy Release Analysis (UFSAR Section 14.3.5.4.2)

Short-term LOCA mass and energy release calculations are performed to support the reactor cavity and loop subcompartment pressurization analyses. These analyses are performed to ensure that the walls in the immediate proximity of the break location can maintain their structural integrity during the short pressure pulse (generally less than three seconds) that accompanies a LOCA within the region.

IP2 has been approved for leak-before-break methods, eliminating the postulated primary system large pipe break from the subcompartment design basis. The analysis inputs that may potentially change with the 1.4-percent power uprate are the initial RCS fluid temperatures. Since the critical portion of this event lasts for less than three seconds, the single effect of reactor power is not significant.

The critical flow correlation used in the mass and energy releases for this analysis will provide an increase in the mass and energy release for a slightly lower fluid temperature. Based on a review of the 1.4-percent power uprate conditions, the limiting RCS conditions for pressure and temperature bound the proposed 1.4-percent uprate. Therefore, the current licensing basis for the short-term LOCA subcompartment pressurization analysis is unaffected by the 1.4-percent power uprate.

8.5 STEAM LINE BREAK OUTSIDE CONTAINMENT

As defined by IP2, high-energy lines are pipes in which, during normal plant operation, the fluid temperature exceeds 200°F or the pressure exceeds 275 psig.

The piping outside of containment at IP2 that qualifies for high-energy line break (HELB) analyses are:

- The shield wall area, which consists of the auxiliary feedwater pump room
- The steam and feed line penetration area
- The blower room, the pipe bridge between the shield wall area, the Turbine Building and the Primary Auxiliary Building including the pipe penetration area
- The safety injection pump room
- The RHR pump room
- The sample line tunnel
- The cable tunnel
- General areas of the Primary Auxiliary Building

This breakdown of piping systems by area is consistent with the method of analysis for HELB used at IP2.

Since changes to operating temperatures, pressures, and flow rates for applicable high- and moderate-energy piping systems are sufficiently small (≤3 percent), the existing design basis for pipe break, jet impingement, and pipe whip considerations remains valid for 10 CFR 50 Appendix K power uprate parameters.

The proposed 1.4-percent Appendix K power uprate does not create any new high-energy lines, and the existing HELB analysis for breaks outside containment will bound the uprate conditions. Therefore, no additional HELB evaluations on systems outside containment are required in support of the proposed power uprate.

8.6 POST-LOCA CONTAINMENT HYDROGEN GENERATION

A review was conducted of the current analysis for post-LOCA hydrogen generation inside containment. The current analysis was performed at a reactor power level of 3216 MWt. The 1.4-percent power uprate to 3114.4 MWt is, therefore, bounded by the existing analysis.

8.7 STEAM GENERATOR TUBE RUPTURE THERMAL-HYDRAULIC ANALYSES

The thermal-hydraulic analysis for the steam generator tube rupture (SGTR) event is performed to calculate the primary-to-secondary break flow and steam releases to the environment, which are then used as input to the radiological consequences analysis for the SGTR. The analysis assumes that the operator identifies the accident type and terminates break flow to the ruptured steam generator within 30 minutes of accident initiation. The thermal-hydraulic analysis considers a core power of 3074.1 MWt, with an increase of at least 5 percent (the actual value varies dependent on the release path and timing) applied to enable future evaluations without requiring re-analysis. Since break flow is not sensitive to power, and the steam releases are considered to be proportional to power, the analysis margin will bound the potential effects of the 1.4-percent power uprate. Therefore, the current analysis of record for the SGTR event is unaffected, and bounds the 1.4-percent power uprate.

As documented in the IP2 UFSAR, sufficient indications and controls are provided at the control board to enable the operator to complete the functions necessary to terminate break flow within 45 minutes for the design-basis event even without offsite power. This evaluation included a 2-percent power uncertainty allowance. Therefore, the evaluation is unaffected by the 1.4-percent uprate.

8.8 ACCIDENT ANALYSES RADIOLOGICAL CONSEQUENCES

An evaluation was performed to determine the potential effects of the 1.4-percent power uprate that will increase IP2 core thermal power from 3071.4 MWt to 3115 MWt plus a calorimetric uncertainty of 0.6 percent.

The IP2 radiological dose analyses are:

- Locked Rotor Accident (UFSAR Section 14.1.6.5)
- Fuel Handling Accident (UFSAR Section 14.2.1)
- Waste Liquid Tank Failure (UFSAR Section 14.2.2)

- Waste Gas System Failure (UFSAR Section 14.2.3)
- Steam Generator Tube Rupture Accident (UFSAR Section 14.2.4)
- Main Steam Line Break (UFSAR Section 14.2.5)
- Rod Cluster Control Assembly Ejection (UFSAR Section 14.2.6)
- Loss-of-Coolant Accident (UFSAR Section 14.3.6)

The source terms used in these analyses were calculated at a core power of 3216 MWt and, therefore, bound operation at a core power level of 3115 MWt with a 0.6-percent power measurement uncertainty. The steam release rates to the environment were also evaluated and were determined to bound those for the 1.4-percent power uprate.

Therefore, the current IP2 radiological analyses of record are unaffected and bound operation with the 1.4-percent power uprate.

8.9 REFERENCES

8-1 WCAP-8708-P-A / WCAP-8709-NP-A, "MULTIFLEX A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics," K. Takeuchi, et al., September 1977.

8-2 WCAP-15270 (Westinghouse Proprietary Class 2), "Determination of Acceptable Baffle-Barrel-Bolting for Four Loop Westinghouse Downflow and Converted Upflow Domestic Plants," R. E. Schwirian, et al, November 2001.

8-3 WCAP-15029-P-A / WCAP-15030-NP-A, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions," R. E. Schwirian, J. A. Barsic, W. D. Rabenstein, D. R. Bhandari, A. W. Harkness, January 1999.

8-4 WCAP-15704 (Westinghouse Proprietary Class 2), "Control Rod Insertion Following a Large Cold Leg LOCA Generic Analysis for 3 Loop and 4 Loop Plants," D. J. Fink, et al, October 2001.

8-5 WCAP-15245 (Proprietary) / WCAP-15246 (Non-Proprietary), "Control Rod Insertion Following a Cold Leg LBLOCA, D. C. Cook Units 1 and 2," J. A. Barsic, D. C. Garner, Y. C. Lee, K. B. Neubert, C. Yu, February 28, 1999.

8-6 Letter from John F. Stang (US NRC) to Robert P. Powers (Indiana Michigan Power Company), "Issuance of Amendments - Donald C. Cook Nuclear Plant, Units 1 and 2 (TAC Nos. MA6473 and MA6474)," December 23, 1999.

8-7 WCAP-11397-A (Proprietary) and WCAP-11398 (Non-Proprietary), "Revised Thermal Design Procedure," A. J. Friedland and S. Ray, April 1989.

8-8 WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.

8-9 WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," R. L. Haessler, et al., January 1990.

8-10 SECY-83-293, "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," W. J. Dircks, July 19, 1983.

8-11 NS-TMA-2182, Letter from T. M. Anderson (Westinghouse) to Dr. S. H. Hanauer (NRC) dated December 30, 1979, "ATWS Submittal."

8-12. NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," December 1978.

8-13 WCAP-10858-P-A, Rev. 1, "AMSAC Generic Design Package," M. R. Adler, July 1987.

9 ELECTRICAL POWER

The station electrical power systems consist of the Electrical Distribution Systems (EDSs), power block equipment, emergency diesel generators (EDGs), and the Direct Current (DC) System. The equipment was evaluated to determine the potential effects of a 1.4-percent power uprate on their ability to function within current design parameters. The electrical systems and equipment were found to be acceptable for the 1.4-percent power uprate.

The iso-phase bus will limit the amount of MVARs exported with the generator at a lagging power factor and the main transformers (MTs) will limit the amount of MVARs imported with the generator at a leading power factor.

9.1 ELECTRICAL DISTRIBUTION SYSTEM

The station EDS is comprised of the connections to the unit auxiliary transformer (UAT) and the station auxiliary transformer (SAT), the medium voltage buses and low voltage buses and transformers, and interconnections and the DC Distribution System.

9.1.1 DC Systems

The Indian Point Unit 2 (IP2) site electrical power supplies include the Class 1E Battery System. This system consists of four independent separated buses. One available battery charger and one battery energize each bus.

The DC System is not affected by the 1.4-percent power uprate since no loads were added to the system.

9.1.2 AC Systems

Voltage Profile

The electrical changes resulting from the power uprate occur for the balance-of-plant (BOP) equipment at the medium voltage level. A review of the loading in the Load Flow Calculation shows that the load increases are within the capability of the medium voltage buses. The changes in bus loading will be reflected in revised bus voltages. Since the magnitude of the changes is so minimal and the On Load Tap Changer regulates voltage at 6.9 kV level, the changes will have minimal impact on the existing station voltage profile.

The degraded voltage relays monitor voltage at the low voltage bus level on buses 2A, 3A, 5A and 6A. There were no load increases on these low voltage buses. The voltage change from the load increase on the medium voltage buses feeding buses 2A, 3A, 5A, and 6A is minimal under normal flow and transient flow (load-rejection) conditions. Consequently, impact at the low voltage level is also minimal. Therefore, the Station Auxiliary Electrical Distribution Systems will remain within acceptable limits.

During normal plant operation, medium voltage buses 5 and 6 receive power from the 138 kV system by bus main breakers and the 138/6.9 kV SAT, while buses 1, 2, 3, and 4 receive power from the main generator by bus main breakers and the UATs. On a generator trip, other than a generator over-frequency trip, a "dead-fast" transfer scheme ties buses 1 and 2 to bus 5, and bus 3 and 4 to bus 6, by bus tie breakers. A review of the Load Flow Calculation shows that the Post Bus Transfer 6.9 kV bus voltage levels have sufficient margin to accept the minimal load increase resulting from the 1.4-percent power uprate.

Station Service Fault Analysis

An evaluation of the changes to the Medium Voltage System indicates the available fault current at the 6900 V switchgear will not increase for the 1.4-percent power uprate condition. The medium voltage motors affected by the 1.4-percent power uprate will remain the same size, and no additional motor loads were added. The 1.4-percent power uprate does not result in any changes to the system impedance network because no equipment replacements or additions are required.

9.1.3 Non-Segregated Phase Bus Ducts

There are two voltage levels of non-segregated bus duct, one for the medium voltage switchgear and one for the low voltage switchgear.

The medium voltage Non-segregated Phase Bus Duct connects the UAT to 6900 V switchgear. The segments that run from the UAT to Switchgear Bus 4 and the SAT to Switchgear Bus 6 have a continuous rating of 4000 Amps. The remaining segments are rated 2000 Amps or 1200 Amps per phase.

The low voltage Non-segregated Phase Bus Ducts connect the EDGs (EDG 21, EDG 22, and EDG 23) to respective 480 V switchgear buses 5, 2A/3A, and 6A. The capacity of the non-segregated bus that connects the UAT and the SAT to the medium voltage bus exceeds the transformer capacity.

The other buses are sized to the same rating as the bus ratings they feed. A review of the Load Flow Calculation showed that the bus currents are less then the rating and remain less with the additional load added by the 1.4-percent power uprate. Therefore, the medium voltage non-segregated buses are acceptable.

No additional load was added to the low voltage buses. Therefore, the low voltage non-segregated buses are acceptable.

9.1.4 Station Service Transformers

Four Station Service Transformers (SSTs) (2, 3, 5 and 6) supply power to the 480 V switchgear buses 2A, 3A, 5A, and 6A and each have a nameplate rating of 6,900 to 480/277 V, 2000/2666 kVA AA/FA (self-cooled/forced-air cooled) @ 150°C rise, three-phase, 60 HZ.

The increase in loading resulting from the power uprate occurs at the medium voltage level. There is no effect on loading at the low voltage level. As a result, the load on the SSTs is unchanged.

9.2 POWER BLOCK EQUIPMENT

The power block equipment consists of the main generator, the iso-phase bus duct, the MTs, the UATs and SATs, the switchyard, and the lines to the switchyard. Grid stability is also addressed with the power block equipment.

9.2.1 Main Generator

The nameplate rating of the main generator is 1439.2 MVA (based on 75 psig hydrogen pressure), 0.91 power factor, 22 kV, three-phase, 60 HZ at 1800 RPM. The generator is supported by a number of systems, including cooling, lubrication, and sealing systems.

The output of the main generator at the current power level of 3071.4 MWt and 559°F T_{avg}, is 1022 MWe, taken from the current station heat balance. For the 1.4-percent power uprate, at 3114.4 MWt and 562°F T_{avg}, the main generator is anticipated to operate at an output as high as 1042.4 MWe, based on the baseline heat balance for 101.4 percent of the current core thermal power. The generator runs at an H_2 pressure of 60-65 psig. Assuming an H_2 pressure of 60 psig, the generator's capability curves show that at 1042.4 Mwe the generator is capable of exporting approximately 630 MVAR (lagging power factor of 0.856). At the same real power output and an H_2 pressure of 60 psig, the generator is capable of importing approximately 510 MVARs (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 an H_2 pressure of 60 psig 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 volt-ampere reactive (VAR) production.

Main Generator Protection

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. They are not affected by machine operation at the 1.4-percent power uprate conditions. For example, overlapping differential schemes provide machine protection for both internal (generator differential and unit differential schemes) and external (unit differential scheme) phase fault conditions. The schemes are not affected by load changes within the rated operating range of the machine. Ground-fault protection schemes provided by ground over-voltage relays, are designed and set based upon the system grounding design, and are independent of main generator output. Loss of excitation and negative sequence protection schemes that are included among the remaining main generator protection schemes are similarly unaffected by unit operation at the 1.4-percent power uprate conditions because the machine will be operated within its rated capability. A voltage balance relay protects against a blown PT fuse.

9.2.2 Isolated Phase Bus Duct

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 amount of MVARs exported by the generator will be limited to maintain the loading within the rating of the most limiting section of iso-phase bus at the 1.4-percent power uprate level.

The tap to the UAT is rated at 1.5 kV. This is a self-cooled rating. The tap is capable of carrying the full-load rating of the transformer, 1.13 kA with the generator at 95 percent, 1.188 kA.

9.2.3 Transformers

Main Transformers

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. Main transformers 21 and 22 each have an impedance of 16.08 percent at the 55°C rating, therefore, the load will divide evenly between the transformers.

The total capacity of the MT bank is 1084 MVA at the 55°C rating and 1214 MVA at the 65°C rating. With the generator operating at the 1.4-percent power uprate level, lagging pf, and allowing for the load of the UAT, and MT losses, the required capacity of the MTs at the 1.4-percent power uprate level exceeds the 55°C rating, but 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-percent power uprate level, lagging pf and an H_2 pressure of 60 psig.

With the generator operating at a leading pf, and allowing for the load of the UAT, and MT losses, the MVARs imported by the generator will be limited to keep the loading on the MTs within the 65°C rating at the 1.4-percent power uprate level.

Main 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 percent load tap changer. The UAT supplies power to BOP systems under normal operating conditions.

The BOP systems are those systems most affected by power uprate. 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 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. Since there is no net increase in house loads resulting from power uprate, the UAT is acceptable.

Station Auxiliary Transformer

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 percent load tap changer. The SAT provides power to BOP systems under abnormal operating conditions.

The Balance of Plant Systems are those systems most affected by power uprate. 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 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. Since there is no net increase in house loads resulting from power uprate, the SAT is acceptable.

Transformer Protection

A review of one-line diagrams and protective relay settings indicates that MT, UAT, and SAT protection essentially consists of high-speed phase-fault protection and ground-fault protection.

The main power transformer is protected by a differential relay scheme. Backup protection is provided by the unit differential relay and the main transformer neutral time over-current relay.

The UAT is protected by a differential relay scheme for internal phase and ground fault. A neutral time over-current relay provides ground fault protection to the low voltage winding. Backup protection is provided by instantaneous and time over-current relays on each phase.

The SAT is protected by a differential relay scheme. Backup protection is provided by single-phase instantaneous and time over-current relays and a neutral time over-current relay protection scheme.

9.2.4 Switchyard

The current to the switchyard is bounded by the capability of the main transformers. The transformers are dual rated, 55°C/65°C and the MVA ratings are 542/607. The rating of the overhead lines from the MT is 2.13 kA, which will carry the full load of the transformers. Therefore, the overhead lines are acceptable at the 1.4-percent power uprate level.

The 345 kV breakers and switches installed in the 345 kV switchyard are rated as 3000 Amps. This exceeds the capability of the main transformer, therefore, the breakers and switches are acceptable at the 1.4-percent power uprate level.

Overhead Lead Protection

Main transformer overhead leads to the 345 kV switchyard are protected by primary and backup pilot wire relay schemes that provides internal-phase and ground-fault protection. The SAT transformer leads from the 138 kV switchyard are protected by primary and backup pilot wire relay schemes that provide internal-phase and ground-fault protection. Since the existing transformers will continue in service, the existing electrical protection schemes are unaffected when the units operate at the 1.4-percent power uprate conditions.

9.2.5 Grid Stability

A grid stability study was performed in January 2001. This study showed the unit is stable. It is expected that this study will remain valid for the 1.4-percent power uprate.

9.3 EMERGENCY DIESEL GENERATORS

The onsite standby power supply consists of three independent EDGs.

The emergency bus loading was evaluated to determine any load increases that would affect it as a result of the 1.4-percent power uprate. The load changes only occur to medium voltage motors. Therefore, the EDGs are not affected by uprate conditions.

A review of the electrical loading associated with each EDG concluded that the loads are unaffected by operation at 1.4-percent power uprate conditions. Since no new loads or EDG changes have been identified, the existing EDG electrical protection schemes are similarly unaffected.

9.4 MISCELLANEOUS ELECTRICAL EQUIPMENT

The electrical equipment that supports the mechanical systems is typically motors, cables, and circuit breakers.

System evaluations have determined that some medium voltage motors powered from the non-safety-related 6,900 V switchgear have revised operating points. The condensate pumps, rated 3000 HP each, and the heater drains pumps, rated at 1000 HP each, experience a BHP increase. The existing motor drives will operate at a BHP less than the nameplate rating of the motor during full-load conditions at power uprate, so no motor replacements will be required at power uprate. Normal design practice is to size the motor feeder cables to the nameplate rating of the motor and to also allow for motor service factor. The BHP of the condensate and heater drains pumps remain below nameplate at power uprate, therefore, the cables should be acceptable.

A review of one-line diagrams and protective relay settings indicates that several different schemes are used to provide medium voltage (6900 V) motor and motor feeder protection. The essence of each scheme is to provide electrical protection against the damaging effects of sustained overload, locked rotor, and phase- and ground-fault conditions. For example, instantaneous over-current and time over-current relays provide phase and ground-fault protection and motor overload protection, respectively. Some schemes also incorporate thermal overload relays.

Design of the applied motor protective relay schemes is based upon motor application, ratings, and design parameters and feeder ratings. Since the motors affected by the 1.4-percent power uprate will be operated within their respective rated capabilities and because none of the affected motor drives will be replaced, operation at the 1.4-percent uprate conditions will not affect the existing medium voltage motor.

10 BALANCE OF PLANT

The Indian Point Unit 2 (IP2) Balance-of-Plant (BOP) Systems were reviewed for potential effects due to the 1.4-percent power uprate to 3114.4 MWt reactor core power. The BOP systems that could potentially be affected due to the 1.4-percent power uprate are the:

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

10.1 MAIN STEAM AND STEAM DUMP SYSTEM

The current and expected uprate steam flow parameters for average reactor coolant temperatures (T_{avg}) of 559°F and 562°F are shown in Table 10-1.

Table 10-1
Current and Expected Uprate Steam Flow Parameters

	Current, T_{avg} = 559°F	1.4% Uprate, T_{avg} = 559°F	Current, T_{avg} = 562°F	1.4% Uprate, T_{avg} = 562°F
Steam Flow, lbs/hr	13.260×10^6	13.479×10^6	13.184×10^6	13.391×10^6
Steam Pressure, psia	708	698	756	752
Steam Temperature, °F	504.4	502.8	511.7	511.1

These pressure and temperature changes are bounded by the design pressure and temperature values. The major components of the Main Steam System (MSS) were evaluated for the increase in system flow.

The major components of the MSS include the main steam safety valves (MSSVs), atmospheric relief valves (ARVs), main steam isolation valves (MSIVs), high-pressure steam dump valves, and turbine overspeed and low-pressure steam dump valves. These components were evaluated (as discussed below) for the proposed uprate conditions and they are adequate to support the proposed uprate.

The zero-power load conditions do not change and remain bounding for the main steam design pressure and temperature.

Steam Generator Main Steam Safety Valves

The setpoints of the MSSVs are determined based on the design pressure of the steam generators (1085 psig) and the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code. Since the design pressure of the steam generators has not changed with the 1.4-percent power uprate, there is no need to revise the setpoints of the safety valves.

Indian Point Unit 2 has 20 safety valves with a total capacity of 15.108 x 10^6 lb/hr, which provide about 113 percent of the uprate full-load heat balance steam flow of 13.39 x 10^6 lb/hr. Therefore, based on the range of Nuclear Steam Supply System (NSSS) design parameters for the uprate, the capacity of the installed MSSVs meets the Westinghouse sizing criterion of being able to pass 100 percent of the maximum calculated steam flow. Refer to Section 6.2.3 of this report for additional information on the MSSVs.

The original design requirements for the MSSVs (as well as the ARVs and steam dump valves) included a maximum flow limit per valve of 890,000 lb/hr at 1085 psig. Since the actual capacity of any single MSSV, ARV, or steam dump valve is less than the maximum flow limit per valve, the maximum capacity criterion is satisfied.

Steam Generator Power-Operated Atmospheric Relief Valves

The ARVs, which are located upstream of the MSIVs and downstream of the MSSVs, are automatically controlled by steam line pressure during plant operations. The ARV set pressure for these operations is between zero-load steam pressure and the setpoint of the lowest-set MSSVs. Since neither of these pressures change for the proposed range of NSSS operating parameters, there is no need to change the ARV setpoint.

The primary function of the ARVs is to provide a means for decay heat removal and plant cooldown by discharging steam to the atmosphere when either the condenser, the condenser circulating water pumps, or steam dump to the condenser is not available. In the event of a tube rupture event in conjunction with loss of offsite power, the ARVs are used to cool down the Reactor Coolant System (RCS) to a temperature that permits equalization of the primary and secondary pressures at a pressure below the lowest-set MSSV. The RCS cooldown and depressurization are required to preclude steam generator overfill and to terminate activity release to the atmosphere.

The steam generator ARVs are sized to have a capacity equal to 10 percent of rated steam flow at no-load pressure. The total relieving capacity of these valves is 1,369,000 lbs/hr at a valve inlet pressure of

1005 psig. This capacity is approximately 10.2 percent of the uprate steam flow. Therefore, the current valve capacity is adequate. Refer to Section 6.2.3 of this report for additional information on the ARVs.

Main Steam Isolation Valves, Non-Return/Check Valves, and Main Steam Isolation Bypass Valves

The MSIVs, in conjunction with check valves, are located outside the containment and downstream of the MSSVs and ARVs. The valves function to prevent the uncontrolled blowdown of more than one steam generator and to minimize the RCS cooldown and containment pressure to within acceptable limits following a main steam line break. To accomplish this function, the design requirements specified that the MSIVs must be capable of closure within five seconds of receipt of a closure signal against steam break flow conditions in the forward direction (IP2 Improved Technical Specifications).

Rapid closure of the MSIVs and check valves following a postulated steam line break would cause a significant differential pressure across the valve seats and a thrust load on the MSS piping and piping supports in the area of the MSIVs and check valves. The worst cases for differential pressure increase and thrust loads are controlled by the steam line break area (i.e., mass flow rate and moisture content), throat area of the steam generator flow restrictors, valve seat bore, and no-load operating pressure. Since these variables and no-load operating pressure are not affected by the 1.4-percent power uprate, the design loads and associated stresses resulting from rapid closure of the MSIVs and check valves will not change. Consequently, the 1.4-percent power uprate has no significant effect on the interface requirements for the MSIVs or check valves.

The MSIV bypass valves are used to warm up the main steam lines and equalize pressure across the MSIVs prior to opening the MSIVs. The MSIV bypass valves perform their function at no-load and low-power conditions where power uprate has no significant effect on main steam conditions (e.g., steam flow and steam pressure). Consequently, the 1.4-percent power uprate has no significant effect on the interface requirements for the MSIV bypass valves.

Steam Dump System

The High-Pressure Steam Dump System (SDS) creates an artificial steam load by dumping steam from ahead of the turbine valves to the main condenser. A steam dump capacity of 40 percent of rated steam flow at full-load steam pressure is required for a large-load rejection transient, as well as preventing an MSSV lifting following a reactor trip from full power.

Indian Point Unit 2 is equipped with 12 condenser steam dump valves, and each valve is specified to have a flow capacity of 5.05×10^5 lb/hr at a valve inlet pressure of 650 psia, or 6.06×10^6 lb/hr total capacity. The total capacity provides a valve capability of about 46 percent of the current steam flow (13.18×10^6 lb/hr). The steam dump valves will pass approximately 45 percent of the uprate steam flow (13.39×10^6 lb/hr). Refer to Section 6.2.3 of this report for additional information on the SDS.

The design requirements for valve stroke times are still applicable for the NSSS design parameters for the 1.4-percent power uprate.

The NSSS controls systems analysis provides an evaluation of the adequacy of the Steam Dump Control System at the 1.4-percent power uprate NSSS design parameters.

The Low-Pressure Steam Dump System is designed to bypass steam from the high-pressure turbine exhaust lines directly to the condenser. The system is provided to minimize turbine speedup immediately following a turbine trip or generator breaker opening. Upon any generator breaker opening, turbine trip, or overspeed trip with the isolation valves open and dump valves closed, the dump valves would be activated. This would divert approximately 25 percent of the steam available to overspeed the turbine to the condensers, thus reducing the potential maximum turbine speed. The uprate conditions remain bounded by the design of the Overspeed System.

10.2 CONDENSATE AND MAIN FEEDWATER SYSTEMS

The Condensate and Main Feedwater Systems automatically maintain steam generator water levels during steady-state and transient operations. The Main Feedwater System also automatically isolates the Condensate and Main Feedwater Systems from the steam generators, when required, in order to mitigate the consequences of an accident. The revised 1.4-percent power uprate operating conditions will affect both the feedwater volumetric flow and system pressure drop. However, in all cases, the results of the evaluations conclude that the respective system designs remain adequate for uprate operation.

The Condensate and Main Feedwater Systems include the condensate pumps, heater drain pumps, and main feedwater pumps (MFPs). These pumps, in conjunction with the feedwater regulating valves (FRVs) and low-flow feedwater regulating bypass valves, serve to regulate the main feedwater flow to the steam generators to maintain steam generator water level during steady-state and transient operation. The MFPs are variable speed and are adjusted to maintain main feedwater flow and pressure during operation. For normal, full-load conditions, there are three condensate pumps running, two heater drain pumps running, and two steam-powered MFPs running. The FRVs are arranged in parallel with the low-flow feedwater regulating bypass valves. The system is designed to provide adequate main feedwater flow during a 50 percent of full-power load rejection transient. The result of the 1.4-percent power uprate will be to increase the amount of feedwater supplied to the steam generators at full load by approximately 1.7 percent.

Condensate and Main Feedwater System pressure and temperatures are evaluated for uprate conditions. This review is based on design and current calculations. The results indicate that the system pressure and temperatures continue to be bounded by design parameters.

The MFP suction pressure was evaluated because reductions in this pressure could affect MFP net positive section head available (NPSHa) and possible instrument settings. Operational data was used to establish the current MFP suction pressure for this mode of operation, and existing system calculations were scaled to uprate flows to estimate the increase in system pressure loss and the decrease in pump head. Current data indicated the MFP suction pressure is approximately 400 psia. This review indicated that, at uprate conditions, the MFP suction pressure would decrease approximately 5 psi to approximately 395 psia for the 1.4-percent power uprate conditions. This pressure is still above the minimum required for adequate NPSH. Adequate MFP NPSHa is also maintained for a large-load rejection transient at the uprate conditions.

The uprate flows are within the capability of the MFPs. The MFP speed is adjustable to maintain a preset pressure differential between main steam pressure and MFP discharge pressure. Based on current pump speed, there is sufficient margin available to accommodate any Appendix K uprate-related increase in pump speed. The FRV position will increase slightly to accommodate the increased flow. The FRV position is expected to increase from approximately 80-percent open to approximately 83-percent open. This is acceptable for plant operation.

The feedwater heater shell and tube side conditions are discussed in Section 10.5 of this report.

An evaluation of the Condensate and Main Feedwater Systems has not identified any limitations in the existing design that would preclude the 1.4-percent power uprate. Component parameters are bounded by original equipment design, or by the original design considerations for off-normal operation.

10.3 CONDENSER/CIRCULATING WATER

An evaluation of the Main Condenser and Circulating Water System has not identified any limitations in the existing design that would preclude the 1.4-percent power uprate. Component parameters are bounded by original design equipment ratings, or by the original design considerations for off-normal operation. The small increase in heat load will result in an insignificant change in condenser operating back-pressure. The potential effect of minor temperature increases in the Circulating Water System discharge temperature is addressed in Section 12.3, Environmental Impact Consideration.

10.4 EXTRACTION STEAM SYSTEM

An evaluation of the Extraction Steam System has not identified any limitations in the existing design that would preclude the 1.4-percent power uprate. Minor changes are expected in system flow rates, pressures, and temperatures. All uprate pressures and temperatures are bounded by the piping design ratings. Most extraction lines experience an increase in flow rate, and subsequent increase in flow velocity. The changes in flow conditions may affect erosion concerns for these lines. The IP2 Flow Accelerated Corrosion Program will be updated to evaluate these conditions to ensure satisfactory pipe conditions are maintained. The pressure drops in the extraction lines for the turbine to feedwater heaters remain acceptable.

10.5 FEEDWATER HEATERS AND DRAINS

The feedwater heaters were reviewed based on design information, current operating conditions, and expected uprate conditions. All heater tube-side pressures, temperatures, and flow velocities are acceptable at uprate conditions. The uprate will, generally, increase (by approximately 2 percent) the extraction steam flow to the heaters; for some heaters there is little change or a decrease in flow expected. This results in some heaters being predicted to exceed the heater design conditions on the shell side of the heaters for uprate operations. The higher than design extraction flow increases the cascaded drain flows such that these flows exceed the design drain flow. The increase in drain flow is expected to be approximately 2 percent. Additionally, based on operating information, some heaters are currently exceeding design conditions. Based on plant monitoring of the current heater conditions, little degradation (tube plugging, wall and nozzle thinning) has been shown to the heaters in service. Continued monitoring and review of heater conditions will be performed to ensure satisfactory heater conditions in support of the power uprate.

All heater drain piping remains within design pressure and temperature ratings. All heater drain and heater drain tank control valves (normal and alternate paths) are adequate for the uprate operating conditions. The feedwater heater relief valves and vents were also evaluated and found to be acceptable for 1.4-percent power uprate operation.

For the reheater drains, the required control valve flow under 1.4-percent power uprate conditions is similar to the current flow. Based on the current operating positions, no normal control valve capacity limitations have been identified. Component parameters are bounded by original design equipment ratings, or by the original design considerations for off-normal operation.

10.6 COOLING WATER SYSTEMS

10.6.1 Service Water System

Following a loss-of-offsite-power and/or loss-of-coolant-accident, the Service Water System provides cooling water directly to the following essential loads:

- Containment recirculation fan cooling coils
- Containment recirculation fan motor cooling coils
- Instrument air closed cooling water heat exchangers
- Diesel generator lube oil coolers and jacket water coolers
- Radiation monitors R-46 and R-53 for water samples taken downstream from the containment recirculation fan cooling coils and fan motor cooling coils
- Sample cooling for radiation monitors R-46, R-49, and R-53
- Service water pump strainer blowdown
- Chlorine and zebra mussel monitors

The increased decay heat and turbine auxiliaries cooling loads will have a small impact on the cooling water temperature increase. However, they do not impact the required cooling water flow rates. The evaluation of the Service Water System has not identified any limitations in the existing design that would preclude the 1.4-percent power uprate. Component parameters are bounded by original design equipment ratings, or by the original design considerations for off-normal operation.

10.6.2 Component Cooling Water System

An evaluation of the Component Cooling Water System has not identified any limitations in the existing design that would preclude the 1.4-percent power uprate. Because the accident analyses are performed assuming a core thermal power level of at least 102 percent of the rated value, the Component Cooling Water System performance remains bounding for operation under post-accident conditions. Component parameters are bounded by original design equipment ratings, or by the original design considerations for off-normal operation.

10.7 HEATING, VENTILATION, AND AIR CONDITIONING SYSTEMS

10.7.1 Central Control Room HVAC Systems

The Central Control Room (CCR) HVAC System is designed to provide 3 main functions. To remove/add heat from/to supply air as required in order to maintain design temperature and relative humidity conditions inside the CCR during all modes of plant operation, isolate the CCR to prevent infiltration of toxic gases and smoke during high radiation and/or safety injection conditions, and maintain a slight positive pressure in the control room during normal, high radiation, and safety injection modes of operation. The CCR is served by a combination of the original Unit 1 and the Unit 2 HVAC system.

An evaluation of the CCR HVAC System has not identified any limitations in the existing design that would preclude the 1.4-percent power uprate. Component parameters are bounded by original design equipment ratings, or by the original design considerations for off-normal operation.

10.7.2 Auxiliary Feedwater/Electrical Vent System

The Auxiliary Feedwater/Electrical Vent System (AEVS) is comprised of the following subsystems:

- Auxiliary Feedwater Building Ventilation System
- CCR Electrical Tunnel Exhaust System
- Electrical Switchgear Room (480V) Ventilation System
- Battery Room Exhaust Ventilation System

The AEVS at IP2 is designed to provide the following:

- Ventilation to Auxiliary Feedwater Building, CCR electrical tunnel, electrical switchgear room, and battery rooms to maintain the space temperatures at or below the maximum design limits by removing internally generated heat from these areas. In the cold season, the AEVS provides these spaces with sufficient heat to maintain the space temperatures at or above the minimum design temperature.

- Ventilation to battery rooms to prevent hydrogen concentration from reaching explosive levels.

An evaluation of the AEVS has not identified any limitations in the existing design that would preclude the 1.4-percent power uprate. Operation at the 1.4-percent power uprate conditions will not affect the operation of any of the AEVS. Component parameters are bounded by original design equipment ratings, or by the original design considerations for off-normal operation.

10.8 INSTRUMENTATION AND CONTROLS

With the exception of modifications required to reduce the uncertainty associated with main feedwater flow measurements, no BOP plant modifications to the Instrumentation and Control System design are required or recommended to support the 1.4-percent power uprate. The small increase (~1.6 percent) in BOP flows are within the plant instrumentation ranges or setpoints and no changes have been identified due to the power uprate. The existing Instrument and Control System design will not be affected when the plant operates at the 1.4-percent power uprate conditions.

10.9 PIPING AND SUPPORT EVALUATION

Balance-of-plant piping systems and their components were evaluated for increases in operating temperatures, pressures, and flow rates that would result from the implementation of the 1.4-percent power uprate.

The piping system evaluations performed concluded that all piping systems and related components remain acceptable and will continue to satisfy design-basis requirements when considering the temperature, pressure, and flow rate effects resulting from power uprate conditions.

The changes in operating parameters (i.e., temperature, pressure, and flow rate) were determined to be insignificant and were concluded to have a negligible effect on the existing piping system qualifications. No specific pipe stress re-analyses were required to document the acceptability of the 1.4-percent power uprate conditions.

The High and Moderate Energy Line Break Program ensures that systems or components that are required for safe shutdown or are important to safety are not susceptible to the consequences of high and/or moderate energy piping failures. Since changes to operating temperatures, pressures, and flow rates for applicable high and moderate energy piping systems were determined to be sufficiently small, the existing design basis for pipe break, jet impingement, and pipe whip considerations remains acceptable for the 1.4-percent power uprate conditions. That is, the 1.4-percent power uprate does not result in any new or revised pipe break locations, and the existing design basis for pipe break, jet impingement, and pipe whip considerations remains valid for power uprate conditions.

10.10 SPENT FUEL POOL COOLING SYSTEM

The design basis of the Spent Fuel Pool (SFP) Cooling System (SFPCS) includes the capability to maintain the SFP below 140°F following the discharge of 72 fuel assemblies, and below 180°F following the discharge of a full core. These criteria are based, in part, on the calculated time it would take the SFP to boil in the event of a loss of SFP cooling.

An evaluation of the SFPCS has not identified any limitations in the existing design that would preclude the 1.4-percent power uprate. The required post-accident functions are analyzed for at least 102 percent of the current rated core thermal power. These analyses bound the 1.4-percent power uprate conditions.

10.11 MAIN TURBINE

The main turbines (high-pressure and low-pressure units) were supplied by Siemens-Westinghouse. These units have been evaluated and found acceptable for service at the 1.4-percent power uprate conditions.

11 OTHER RADIOLOGICAL CONSEQUENCES

The potential radiological effects of the 1.4-percent power uprate are evaluated for the following:

- Normal operation shielding and personnel exposure
- Normal operation annual radwaste effluent releases
- Radiological environmental qualification (EQ)
- Post-loss-of-coolant-accident (LOCA) access to vital areas including Technical Support Center habitability

Radiological evaluations for accident-related issues are assessed at a core power level of 3133.1 MWt (100.6 percent of 3114.4 MWt) to include a margin of 0.6 percent for power level instrument inaccuracy. Installation of improved core power measurement accuracy enables the allowance for instrument error to be reduced from the traditional 2 percent as recommended in Regulatory Guide 1.49, Revision 1, to 0.6 percent.

Except as noted, radiological evaluations for normal operation related issues are assessed, for the 1.4-percent power uprate, at a core power level of 3114.4 MWt. The normal operation radwaste effluent assessment is based on an assumed core power level of 3133.1 MWt to reflect current regulatory guidance relative to power levels to be used for the Code of Federal Regulations (CFR) 10 CFR 50 Appendix I assessments.

11.1 NORMAL OPERATION ANALYSES

11.1.1 Radiation Source Terms

The current licensed core power level for Indian Point Unit 2 (IP2) is 3071.4 MWt. The 1.4-percent power uprate will increase the isotopic inventory in the core by approximately the percentage of the uprate, i.e. 1.4 percent. The radiation source terms in the reactor coolant (which reflects leakage of core activity from defective fuels, and escape coefficients of the isotopes and its precursors), and all subsequent process streams, will also increase by approximately the percentage of the uprate.

11.1.2 Normal Operation Shielding and Personnel Exposure

The 1.4-percent increase in expected radiation levels will not affect radiation zoning or shielding requirements in the various areas of the plant. As noted in the Updated Final Safety Analysis Report (UFSAR) Section 11.2.2 and Tables 11.2-2 through 11.2-6, IP2 shielding design is based on a core power level of 3216 MWt and a design-basis Reactor Coolant System (RCS) with 1-percent fuel defects. Therefore, this encompasses plant operation at the 1.4-percent power uprate.

Individual worker exposures will be maintained within acceptable limits by the site As Low As Reasonably Achievable (ALARA) Program that controls access to radiation areas. In addition, procedural controls may be used to compensate for increased radiation levels.

11.1.3 Normal Operation Annual Radwaste Effluent Releases

Gaseous and Liquid Releases

The 1.4-percent power uprate will increase the release rate of radioactive isotopes into the reactor coolant and, by primary-to-secondary leakage, to the secondary steam. Due to leakage or process operations, fractions of these fluids are transported to the Liquid and Gaseous Radwaste Systems where they are processed prior to discharge. The effects on these systems are the following:

- Liquid radioactive waste: As the activity levels in the reactor coolant or secondary fluids increase, the activity levels of liquid radwaste inputs /effluents are proportionately increased. Although some wastes, such as from the RCS feed-and-bleed operations, may increase due to a power uprate, most, if not all, of the water generated by these operations would be recycled within the plant. This thereby minimizes the effect of additional waste generation on plant effluent analyses.

- Gaseous radioactive waste: Relative to gaseous radioactivity released into the RCS, the rate of activity entering the Gaseous Radwaste System will increase roughly in proportion to the rate of power increase as the Chemical and Volume Control System (CVCS) letdown will remain constant. As the gaseous radwaste transport and treatment processes are not directly affected by the activity input rate, system performance will not be affected by the 1.4-percent power uprate and the release rate of activity in the Gaseous Radwaste System effluents would increase in proportion to the percentage of uprate.

11.1.4 10 CFR 50 Appendix I Evaluation

The 10 CFR 50 Appendix I evaluation submitted to the Nuclear Regulatory Commission (NRC) in support of the IP2 license ("An Evaluation to Demonstrate Compliance of the Indian Point Reactors with the Design Objectives of 10 CFR Part 50, Appendix I," February 1977), was based on a power level of 3216 MWt and thus encompasses plant operation at a core power level of 3133.1 MWt (i.e., 3114.4 MWt plus a 0.6 percent power level uncertainty). Note that the actual release concentrations and offsite doses are controlled by the IP2 Offsite Dose Calculation Manual, which assures compliance with the IP2 Technical Specifications and 10 CFR 50 Appendix I.

Solid Waste

Per regulatory guidance for a "new" facility, the estimated volume and activity of solid waste is linearly related to the core power level. However, for an existing facility that is undergoing a power uprate, the volume of solid waste would not be expected to increase proportionally. This is because the 1.4-percent power uprate neither appreciably affects installed equipment performance, nor does it require drastic changes in system operation. Only minor, if any, changes in waste generation volume are expected. However, it is expected that the activity levels for most of the solid waste would increase proportionately to the increase in long half-life coolant activity.

Thus, while the total long-lived activity contained in the waste is expected to be bounded by the percentage of the uprate, the increase in the overall volume of waste generation resulting from the 1.4-percent power uprate is expected to be minor.

11.1.5 Normal Operation Analyses - Summary

Based on the discussions provided above, a core power uprate to 3114.4 MWt will not cause radiological exposure in excess of the dose criteria (for restricted and unrestricted access) provided in the current 10 CFR 20. From an operations perspective, radiation levels in most areas of the plant are expected to increase no more than the percentage increase in core power level. Individual worker exposures will be maintained within acceptable limits by the site ALARA Program, which controls access to radiation areas. Gaseous and liquid effluent releases are also expected to increase by no more than the percentage increase in power level. Offsite release concentrations and doses will be maintained within the limits of the current 10 CFR 20 and 10 CFR 50 Appendix I by the site Radwaste Effluent Control Program.

11.2 RADIOLOGICAL ENVIRONMENTAL QUALIFICATION

In accordance with 10 CFR 50.49, safety-related electrical equipment must be qualified to survive the radiation environment at their specific location during normal operation and during an accident.

For purposes of equipment qualification, IP2 is divided into various environmental zones. The radiological environmental conditions noted for these zones are the maximum conditions expected to occur and are representative of the whole zone. Normal operation values represent 40 years of operation. Post-accident radiation exposure levels are determined for a 1-year period following a LOCA. With the exception of equipment that fall under Division of Operating Reactors (DOR) Guidelines, a 10-percent margin is applied to the accident contribution in accordance with the requirements of Institute of Electrical and Electronic Engineers (IEEE)-323.

The 1.4-percent power uprate will increase the activity level in the core by the percentage of the core uprate. The radiation source terms in equipment/structures containing post accident fluids, and the corresponding post-LOCA dose rates/integrated doses in the plant, will also increase by the percentage of uprate.

The normal operation contribution to the EQ dose at IP2 is based on survey data and will, therefore, also increase by the percentage of core uprate.

The post-accident and normal operation environmental dose estimates noted in the IP2 Electrical Equipment Qualification Program Manual encompasses operation at the uprate conditions. Environmental levels for in-containment locations address 3 different power levels, i.e.; 2758 MWt, 3071.4 MWt, and 3216 MWt, whereas environmental levels noted for areas outside containment are based on a power level of 3216 MWt.

A comparison of the "specification" versus "qualification" doses associated with the safety-related components in the IP2 EQ Program indicates that there is sufficient available margin to accommodate a 1.4-percent power uprate, thus demonstrating continued compliance with the requirements of 10 CFR 50.49 and the margin requirements of IEEE-323.

11.3 POST-LOCA ACCESS TO VITAL AREAS

The original design review conducted by IP2 to demonstrate compliance with the requirements of NUREG 0578, Item 2.1.6.b and NUREG 0737, II.B.2, was based on a power level of 2758 MWt. The review identified approximately 30 target areas that may require occupancy during post-LOCA recovery operations. Operator doses while performing vital functions post-LOCA at these areas were estimated resulting in several plant modifications that included implementation of additional shielding, functional changes from manual to automatic actuation, and changes to emergency operating procedures. The NRC approval of the shielding review and implementation of the required modifications was documented in the Safety Evaluation Report (SER) issued in 1983.

Core power uprate will increase the activity level in the core by the percentage of the uprate. The radiation source terms in equipment/structures containing post-accident fluids, and the corresponding post-LOCA dose rates in the plant/operator doses, will also increase by the percentage of uprate.

As part of the power uprate assessment, the IP2 NUREG 0737 design review was reviewed to evaluate the impact of a core power level of 3133.1 MWt (i.e., 3114.4 MWt, plus a 0.6-percent power level uncertainty). The updated operator dose estimates demonstrate continued compliance with the 5 rem operator exposure dose limits of NUREG 0737, II.B.2.

A core power uprate to 3133.1 MWt will not invalidate post-LOCA habitability of the Technical Support Center.

12 MISCELLANEOUS EVALUATIONS

12.1 PLANT OPERATIONS

12.1.1 Procedures

Significant changes to plant procedures will not be required for the 1.4-percent power uprate. Procedural limitations on power operation due to balance-of-plant (BOP) equipment unavailability will be revised as necessary to account for the increase in core power to 3114.4 MWt. Changes associated with the 1.4-percent power uprate will be treated in a manner consistent with any other plant modification.

Procedures required for the operation and maintenance of the Caldon Leading Edge Flow Meter (LEFM) Check System will be implemented.

Specific operator actions to be taken when the Caldon LEFM Check System is inoperable are discussed in Section 3.3 and will be addressed in procedural guidance.

12.1.2 Effect on Operator Actions and Training

Engineered Safety Features System design and setpoints, and procedural requirements, already bound the proposed uprate. The responses of the reactor operators to any event will be unaffected by a change in rated thermal power (RTP).

There will be minimal effect on alarms, controls, and displays for a 1.4-percent power uprate. The Caldon LEFM Check System has indication on the plant computer displays in the control room to alert operators to conditions that impair its availability or accuracy. No other alarm effects 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 1.4-percent power uprate is implemented, the Nuclear Instrumentation System will be adjusted to indicate the new 100-percent RTP in accordance with improved Technical Specification requirements and plant administrative controls. Since the 1.4-percent power uprate is predicated on the availability of the Caldon LEFM Check 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 1.4-percent power uprate will be reflected in the plant simulator.

12.1.3 Plant Integrated Computer System

Process parameter scaling changes will be made, as required, to the Plant Integrated Computer System (PICS). There are no other effects to the PICS from the 1.4-percent power uprate.

12.2 PLANT PROGRAMS

12.2.1 10 CFR 50, Appendix R

The emergency lighting and reactor coolant pump (RCP) oil collection sections are not affected by the 1.4-percent power uprate. The safe shutdown capability is discussed below.

For a postulated fire with a loss-of-offsite power, Indian Point Unit 2 (IP2) uses one of three internal combustion gas turbines as an AC power source to operate the applicable systems for safe shutdown of the plant. An Appendix R gas turbine provides an additional permanently installed alternative alternating current (AC) power supply as part of an enhanced alternate safe shutdown capability. The Appendix R gas turbine loads are not affected by the 1.4-percent power uprate.

In accordance with the IP2 Appendix R Fire Protection Report, the Residual Heat Removal System (RHRS) must also be capable of achieving Reactor Coolant System (RCS) cold shutdown (down to 200°F) in less than 72 hours after reactor shutdown. This was addressed for the 1.4-percent power uprate in Section 6.1.3.

The safe shutdown capability, with regards to Appendix R requirements, is not affected by the uprate. The 72-hour cooldown requirement is maintained for uprate conditions and there are no physical changes associated with the 1.4-percent power uprate that would affect the Appendix R requirements. Therefore, the Appendix R Program is not affected by the by the 1.4-percent power uprate.

12.2.2 Environmental Impact Qualification

The Code of Federal Regulations (CFR) 10 CFR Part 50.49 requires that nuclear power plants maintain an Environmental Qualification (EQ) Program that addresses all design-basis events (DBEs). These DBEs include conditions of normal operation (including anticipated operational occurrences (AOO)), design-basis accidents (DBAs), external events, and natural phenomena.

The accident conditions that are included for review in the EQ Program are listed in the IP2 Updated Final Safety Analysis Report (UFSAR) as follows:

- Loss-of-coolant accident (LOCA)
- High-energy line breaks (HELBs)
- Main steam line breaks (MSLBs)

The current accident analysis for these conditions bounds the 1.4-percent power uprate conditions. Therefore, the accident analysis component of the EQ Program will not be affected by the 1.4-percent power uprate.

Normal-operation environmental conditions are also specified in the EQ Program for containment and portions of the Auxiliary and Turbine Buildings.

The 1.4-percent power uprate will not significantly alter any normal or AOO conditions or environmental evaluations. Therefore, the 1.4-percent power uprate will not affect any normal operation aspects of the EQ Program.

The 1.4-percent power uprate does not affect seismic aspects of the plant design. Therefore, there is no effect on the IP2 Seismic Qualification Program.

See Section 11 for the evaluation of the radiological aspects of equipment qualification.

12.2.3 Station Blackout

A station blackout (SBO) is defined as the complete loss of AC electric power to the essential and non-essential switchgear buses. (Loss-of-Offsite Electric Power System, concurrent with a turbine trip and the unavailability of the Onsite Emergency AC Power System.) An SBO does not involve the loss of available AC power to buses fed by station batteries through inverters. The event is considered to be terminated upon the restoration of power to the essential switchgear buses for any source, including the alternate AC (ACC) source that has been qualified as an acceptable coping mechanism. Station batteries have been sized to carry their expected shutdown loads following a plant trip and a loss of all AC power.

The 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 the required 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 steam generators, to support reactor heat removal due to the 1.4-percent power uprate.

Systems associated with SBO not affected by the uprate include Auxiliary Feedwater and Condensate Storage, Ventilation, Containment Isolation, and Reactor Coolant Inventory. There are no expected changes to any of these systems due to the uprate. Therefore, there is no effect on the SBO Program due to the 1.4-percent power uprate.

12.2.4 Flow-Accelerated Corrosion

The IP2 has a long-term Flow Accelerated Corrosion (FAC) Monitoring Program that consists of selected portions of single- and two-phase high-energy systems. This program conducts pipe inspections, wall thickness measurements, and erosion predictions. These activities are done to ensure that all applicable piping systems are adequate to continue operation through the next cycle.

The 1.4-percent power uprate results in changes in the operating pressure, temperature, and velocities in several of the BOP systems. Therefore, the FAC Program is affected by the power uprate. Note that although the system operating conditions have changed, the design pressure and temperatures have not changed for any systems because of the uprate. The evaluation performed did not identify any additional systems that would be added to the FAC Program since the current program includes the systems affected by the 1.4-percent power uprate.

Susceptible safety-related and non-safety-related systems are modeled at IP2 using the Electric Power Research Institute's (EPRI's) CHECWORKS software. CHECWORKS models will be revised to incorporate flow and process system conditions that are determined for the 1.4-percent power uprate conditions. The results of these upgraded models will be factored into future surveillance/pipe repair plans.

12.2.5 Safety-Related Motor-Operated Valves

The inputs discussed in Nuclear Regulatory Commission (NRC) Generic Letters 89-10 and 96-05 regarding motor-operated valve (MOV) thrust and torque requirements calculations and discussed in NRC Generic Letter 95-07 regarding MOV pressure locking thermal binding requirement calculations are based on the following:

1. Safety-related pump shutoff heads
2. Valve and tank elevations
3. Tank pressurization values
4. Safety and relief valve set points
5. Reactor Coolant System pressure and temperature limits during RHRS operations
6. Pressure/ temperature calculations for various accident scenarios

A review of items 1 through 6 listed above concluded that the 1.4-percent power uprate will not require any changes to the parameters listed. The pressure/temperature calculations for various accident scenarios are not effected by the 1.4-percent power uprate since these calculations used conservative inputs that bound the inputs for the uprate. Therefore, the 1.4-percent power uprate will not affect the MOV calculations discussed in NRC Generic Letters 89-10, 95-07, or 96-05.

The IP2 MOV Program Summary details the evaluation criteria used to determine the functional requirements for program valves. This states that the worst-case cycling scenarios are developed and used as input for ensuring required valve performance. The operating condition changes in systems will not alter the worst-case scenarios used as MOV Program inputs. Therefore, any system changes associated with the uprate will not affect the MOV Program.

The evaluation of the 1.4-percent power uprate effects on the BOP systems does not produce any changes to any of the other MOV Program systems listed above. Therefore, it is determined that the 1.4-percent power uprate does not affect the MOV Program.

12.2.6 Impact on Probabilistic Safety Assessment Results

The proposed power uprate has the potential to affect several areas in the IP2 Probabilistic Risk Assessment (PRA). These areas are:

- Initiating event frequency
- System success criteria
- Operator recovery timing
- Fission product inventory

Initiating Event Frequency

The likelihood of occurrence of an initiating event is not significantly affected as a result of the 1.4-percent power uprate and is bounded by the uncertainty in the initiating event frequency.

System Success Criteria

The success criteria for the systems modeled in the PRA are based on UFSAR criteria or specific calculations (e.g., SBO, room heatup, or steam generator boiloff) performed to support alternative criteria. Based on available system margins, no changes in system success criteria are expected as a result of the 1.4-percent power uprate.

Operator Recovery Timing

Operator actions to recover from potential core damaging scenarios are included in the PRA where appropriate. The time available to perform these actions is based on the particular scenario and the equipment available. Because of the uncertainty in the operator actions included in the PRA, no change in likelihood of operator success or failure is expected.

Fission Product Inventory

The source term associated with containment release categories will be slightly affected by the 1.4-percent power uprate. The Large Early Release frequency for IP2, however, is driven by bypass scenarios rather than specific fission product values and small changes in fission product inventory will not impact the delineation or frequency of Large Early Releases.

12.2.7 Impact on Generic Letter 96-06 Overpressurization

In September 1996, the NRC issued Generic Letter 96-06 to address (among other concerns) the concerns that thermally induced overpressurization of isolated water-filled piping sections in containment could:

- Jeopardize the ability of accident mitigating systems to perform their safety functions
- Lead to a breach of containment integrity via bypass leakage

The accident pressure/temperature values are still bounded by design. Current accident analyses (LOCA) are bounding for the 1.4-percent power uprate condition, and there are no changes to containment fan cooler (CFC) cooling water flow requirements or CFC accident conditions associated with the uprate. There is no increase in the possibility of overpressurization of isolated segments of safety-related piping inside containment, including penetrations, as a result of the 1.4-percent power uprate. Therefore, the Generic Letter 96-06 Program is not affected by the uprate.

12.3 ENVIRONMENTAL IMPACT CONSIDERATION

The proposed 1.4-percent core power uprate will increase the current licensed power level from 3071.4 MWt to 3114.4 MWt, and result in an upgrade of the Nuclear Steam Supply System (NSSS) thermal power from 3083.4 MWt to approximately 3126.4 MWt. The environmental review conducted for the proposed uprate considered the need for the power uprate and the resulting environmental impact associated with it. This review included considering the operating license and State Pollutant Discharge Elimination System (SPDES) permit limits and the information contained in the Final Environmental Statement (FES). Only a slight change in environmental conditions can be expected for the proposed 1.4-percent core power uprate as discussed below.

Final Environmental Statement

Environmental issues associated with the issuance of an operating license for IP2 were originally evaluated in the Indian Point Unit 2 FES that was approved by the NRC in September 1972.

The proposed 1.4-percent core power uprating is projected to increase the plant's rejected heat by a similar percentage. However, the NRC-approved FES related to operation of IP2 (Volume 1, page I-2, Section I) has already addressed plant operation up to a maximum calculated thermal power of 3,216 MWt. Thus, the slight increase in rejected heat has already been evaluated and determined to not significantly impact the quality of the environment. Also, the proposed increase in power involves no significant change in the types or significant increase in the amount of any effluents that may be released offsite that have not already been evaluated and approved in the FES for a power rating of 3216 MWt. Similarly, as enveloped by the FES, there would be no significant increase in individual or cumulative occupational radiation exposure. (Radiological effluent discharges are addressed in Section 11.)

Circulating Water Discharge Limits

The IP2 is required to maintain an SPDES permit. This permit specifies, in detail, requirements for discharge water quality, as well as quantity and temperature limitations on the circulating water flow.

The 1.4-percent power uprate will not affect the quality of any discharge water. The quantity of circulating water discharged is based on the number and speed of operating circulating water pumps. The capacity of these pumps is not affected by the uprate. The current circulating water flow is based on a balance between minimizing environmental impact (lower flow) and maintaining efficient plant operation (higher flow). As such, IP2 adjusts the circulation water flow based on plant operation and river-water temperature. This will not be affected by the uprate. The SPDES permit, and related legal documents, establish a goal for circulating water flow based on the time of year (river-water temperature). These flow requirements are not affected by the 1.4-percent power uprate. No changes are anticipated to any other (non-circulating water) discharges. Therefore, the quantity of discharged water is not affected by the 1.4-percent power uprate.

The heat rejection by the condenser to the circulating water system will increase slightly within the condenser design basis. The highest circulating water discharge temperatures and greatest circulating water flow conditions occur during the summer months. The 1.4-percent power uprate is expected to increase this temperature approximately 0.2°F. This minor change is not considered significant given the circulating water inlet temperatures (70°F during summer) and current condenser temperature rise (approximately 15°F during summer months). Therefore, as the quality, quantity, and temperature of the plant discharges are not affected by the uprate, the IP2 SPDES permit requirements and limitations are not affected by the 1.4-percent power uprate.

13 CONCLUSION

This report demonstrates that the 1.4-percent power uprate can be safely implemented at Indian Point Unit 2 (IP2). The analysis and evaluations described herein demonstrate that all applicable acceptance criteria will continue to be met based on operation at the 1.4-percent power uprate conditions at 3114.4 MWt, and that there are No Significant Hazards related to this power uprate according to the regulatory criteria of the Code of Federal Regulations (CFR) 10 CFR 50.92. Furthermore, the 1.4-percent power uprate will have no significant effect on the quality of human environment and does not involve an unreviewed environmental question.