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May 30, 2002
IPN-02-041

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Mail Stop O-P1-17
Washington, DC 20555-0001

Subject: Indian Point Nuclear Generating Unit No. 3
Docket No. 50-286
Request for License Amendment for 1.4% Measurement Uncertainty Recapture Power Uprate

References:

1. NRC Regulatory Issue Summary (RIS) 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 1993.

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

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

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

Dear Sir or Madam:

In accordance with 10 CFR 50.90, Entergy Nuclear Operations (ENO) hereby requests a change to Facility Operating License No. DPR-64 and to the Technical Specifications (TS) in Appendix A thereto for Indian Point Nuclear Generating Unit No. 3 (IP3). Pursuant to the requirements of 10 CFR 50.91, a copy of this letter with the attachments containing the proposed changes, safety evaluation and the marked up TS pages is being provided to the designated New York State Official.

The proposed license amendment will increase the licensed core power level for plant operation to 3067.4 MWt, which is 1.4% greater than the current level of 3025 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, as performed in accordance with the requirements set forth in 10 CFR 50, Appendix K.

A001

Improvement in core power measurement accuracy is possible through the reduction of the feed water flow measurement uncertainty utilized in the reactor power calorimetric calculation. The feed water flow measurement uncertainty is reduced through the use of improved measurement instrumentation provided by use of a Caldon LEFM (Leading Edge Flow Meter) Check 2000FC Cabinet. This new Caldon measurement system will improve accuracy in feed water flow, mass, and temperature measurement. This increase in accuracy will allow IP3 to operate at the 1.4% higher reactor thermal power level.

This proposed TS change involves revision of: (1) Licensed core thermal power levels from 3025 to 3067.4 MWt on page 1.1-5; (2) Safety Limit 2.1.1 referenced Figure 2.1-1 (Pressure/Temperature Limit Curve) on page 2.0-2; (3) % steam flow assumed in note (c) of Table 3.3.2-1 on pages 3.3.2-8, 3.3.2-9 and 3.3.2-11; (4) Effective Full Power Year (EFPY) values for LCO 3.4 Figures 3.4.3-1, 3.4.3-2, 3.4.3-3, 3.4.12-1, 3.4.12-2, 3.4.12-3, 3.4.12-4 on pages 3.4.3-3, 3.4.3-4, 3.4.3-5, 3.4.12-9, 3.4.12-10, 3.4.12-11, and 3.4.12-12, ; and (5) LCO 3.7 Main Steam Safety Valve Table 3.7.1-1 applicable neutron flux trip set points on page 3.7.1-3. In addition, several marked up TS Bases changes (on five separate pages) are included for information purposes.

Work supporting the IP3 1.4% power uprate has been performed consistent with Reference # 1. Further, evaluations performed by Westinghouse in support of this power uprate request are based upon Westinghouse methodology established in Reference # 2. This WCAP provides the methodology for evaluating power uprates such as the 1.4% Power Uprate on Nuclear Steam Supply Systems (NSSS), components and safety analyses. NRC is aware, from previous review of this WCAP, that this WCAP has been successfully utilized by other facilities in power uprate license amendment submittals. A comprehensive engineering review program consistent with the WCAP methodology has been performed for IP3 to evaluate the increase in IP3 licensed core power from 3025 to 3067.4 MWt. This report is included as part of the analysis portion of this submittal. Further, the use of a Caldon, Inc. LEFM Check System is based on Topical Reports and Supplements regarding this LEFM Check System that were previously approved by NRC as indicated in References # 3, 4, and 5.

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using the criteria of 10 CFR 50.92(c) and ENO has determined that this proposed change involves no significant hazards considerations. The bases for these determinations are included in the attached submittal. ENO has also reviewed the proposed license amendment against the criteria of 10CFR51.22 for environmental considerations. The proposed changes do not involve a significant hazards consideration, a significant change in the types or a significant increase in the amounts of effluents that may be released offsite, or a significant increase in individual or cumulative occupational radiation exposure. Therefore, ENO concludes that the proposed change meets the criteria as delineated in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.

Attachment I to this application contains the safety analysis for these proposed changes. Attachment II includes the proposed changes to the TS and the TS Bases. The 1.4% Measurement Uncertainty Recapture (MUR) Power Uprate Application Report, IP3-RPT-MULT-03614, Revision 0, is included as Attachment III, as the plant specific analyses in support of this license amendment request. This application report addresses those criteria explicitly identified in Reference # 1 in justification of this 1.4% power uprate amendment request.

We respectfully request review and approval of this TS amendment no later than October 4, 2002 consistent with NRC review expectations published in NRC RIS-2002-03.

There are no new commitments identified in this letter. Once approved, the amendment will be implemented within 30 days. If you have any questions or require additional information, please contact Mr. Kevin Kingsley, NRR Project Manager, at 914-734-6034.

I declare under penalty of perjury that the foregoing is true and correct.
Executed on 5-30-02.

Very truly yours,

Robert J. Barrett
Vice President, Operations - IP3
Indian Point Nuclear Generating Unit No. 3

Attachments:
I. Analysis of Proposed Technical Specification Changes
II. Proposed Technical Specification and Bases Changes (markup)
III. 1.4% Measurement Uncertainty Recapture Power Uprate Application Report

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ATTACHMENT I TO IPN-02-041

ANALYSIS OF PROPOSED

TECHNICAL SPECIFICATION CHANGE REGARDING

1.4% MEASUREMENT UNCERTAINTY RECAPTURE

POWER UPRATE

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

1.0 DESCRIPTION

This letter is a request to amend Operating License DPR-64, Docket No. 50-286 for Indian Point Nuclear Generating Unit No. 3.

The proposed changes to various sections of the Indian Point 3 (IP3) 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 IP3 to operate at a 1.4% higher reactor thermal power level, improving core power output from 3025 MWt to 3067.4 MWt. This 1.4% core power urpate 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 CHANGES

a. Indian Point 3 Technical Specification Definitions Section 1.1 for RATED THERMAL POWER (RTP) currently states:

   "RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3025 MWt."

   Definitions Section 1.1 for RATED THERMAL POWER (RTP) is revised to state:

   "RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3067.4 MWt."

b. Indian Point 3 Technical Specification SL Figure 2.1-1, which shows the loci of points of thermal power, reactor coolant system pressure and vessel inlet temperature for which the calculated DNBR is no less than the Safety Limit DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid, is being revised to indicate: (1) the core thermal limits in this figure change based upon the effect that the 1.4% power uprate has on the exit boiling limit lines shifting, as well as on a more conservative analysis of axial offset control performed in anticipation of a future change from Constant Axial

Offset Control (CAOC) to Relaxed Axial Offset Control (RAOC); and (2) the value for 100% rated thermal power is changed from 3025 to 3067.4 MWt.

c. The following change is made to Indian Point 3 Technical Specifications LCO 3.3.2, Table 3.3.2-1 (pages 1, 2 and 4 of 6), "Engineered Safety Feature Actuation System Instrumentation" note (c): 110% value for full steam flow load changed to 120% full steam flow load.

d. Indian Point 3 Technical Specification LCO 3.4.3 Figure 3.4.3-1(Heat up Curve), Figure 3.4.3-2 (Cool down Curve), Figure 3,4.3-3 (Hydrostatic and In-service leak test curve, Figure 3.4.12-1 (Maximum allowable nominal PORV set point for LTOP (Ops) curve), Figure 3.4-12.2 (Pzr Limits for OPS Inoperable curve for one charging pump feed to RCS), Figure 3.4.12-3 (Pzr Limits for OPS Inoperable curve for three charging pumps and/or one HHSI pump feed to RCS), and Figure 3.4.12-4 (Secondary side limits for RCP start curve with secondary side hotter than primary side) all changed the value for Effective Full Power Years (EFPY) to reflect a slight effect due to the mini-uprate on the applicable service period for all of these curves.

e. Indian Point 3 Technical Specifications LCO 3.7.1, Table 3.7.1-1 (page 1 of 1) indicating Operable Main Steam Safety Valves required versus Neutron Flux Trip set point in percent Rated Thermal Power changed to indicate lower "Applicable Neutron Flux Trip Setpoint" due to this mini-uprate.

Proposed changes to the TS Bases related to the above TS changes are included in Attachment II, for information purposes.

3.0 BACKGROUND

Indian Point Nuclear Generating Unit No. 3 is presently licensed for a full core power rating of 3025 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 IP3 core thermal power by 1.4% from 3025 MWt to 3067.4 MWt by employment of a Caldon LEFM Check System for improvement of accuracy of feed water system variables input into the secondary plant calorimetric program. 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 AND JUSTIFICATION OF REQUESTED CHANGES

Attachment III provides the detailed analysis and justification in the form of a 1.4% Measurement Uncertainty Recapture (MUR) Power Uprate Application Report. This in-depth report summarizes the various evaluations and analyses of the potential effects of the 1.4% core power uprate on plant systems, components, and safety analyses. All work supporting the IP3 1.4% power uprate, and the preparation of this report, has been performed consistent with NRC Regulatory Issue Summary (RIS) 2002-03. Affected and unaffected plant systems, components, and analyses have been clearly distinguished throughout the report according to this RIS 2002-03 guidance. Further, the proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met. These specific evaluations are presented throughout the Uprate Application Report (Attachment III).

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Entergy Nuclear Operations, Inc. (ENO) has evaluated the safety significance of the 1.4% increase in the licensed core thermal power identified in the IP3 TS according to the criteria of 10 CFR 50.92, "Issuance of Amendment." ENO has determined that the subject change does not involve Significant Hazards Consideration 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 evaluations associated with this proposed change to core power level have demonstrated that all 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 IP3. 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 evaluations of this proposed amendment show that all applicable acceptance criteria for plant systems, components, and analyses (including FSAR Chapter 14 safety analyses) will continue to be met for the proposed 1.4% power increase in IP3 licensed core thermal power. The subject increase in core thermal power will not result in conditions that could adversely affect the integrity (material, design, and construction standards) or operational performance of any potentially affected system, component, or analyses. The subject increase in core thermal power will not adversely affect the ability of any safety-related system to meet its safety function. Furthermore, the conditions associated with the subject increase in core thermal power will neither cause initiation of any accident, nor create any new credible limiting single failure. The power uprate does not result in changing the status of events previously deemed to be non-credible being made credible. Additionally, no new operating modes are proposed for the plant as a result of this requested change.

Therefore, the subject increase in core thermal power level will not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Response: No.

The evaluations associated with this proposed change show that all applicable acceptance criteria for plant systems, components, and analyses (including FSAR Chapter 14 safety analyses) will continue to be met for this proposed 1.4% increase in IP3 licensed core thermal power. The subject increase in core thermal power will not result in conditions that could adversely affect the integrity (material, design, and construction standards) or operational performance of any potentially affected system, component, or analysis. The subject power uprate will not adversely affect the ability of any safety-related system to meet its intended safety function. For example, most IP3 analyses already add a 2% uncertainty allowance to the nominal power level to account solely for power measurement uncertainty. These analyses have not been revised for the 1.4% uprate power level conditions because the sum of increased core power level (1.4%) and the improved power measurement accuracy (uncertainty less than 0.6%) is already bounded by the currently analyzed 2% uncertainty allowance.

Therefore, the subject increase in core thermal power will not involve a significant reduction in the margin of safety.

Based on the above, ENO concludes that the proposed license 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. Further, the proposed change has been evaluated to determine whether applicable regulations and requirements continue to be met. These specific evaluations are presented throughout the Technical Analyses of the Application Report.

ENO has determined that the proposed change does not require any exemptions or relief from regulatory requirements, other than those changes proposed in the IP3 TS. Additionally, this change does not affect conformance with any General Design Criteria (GDC) 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 change 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 51.2(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

Issuance by NRC of Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002, has provided the necessary criteria to pursue this MUR power uprate effort at IP3. In addition, NRC recognition of Westinghouse WCAP-10263, "A Review Plan for Uprating the Licensed Power of a Pressurized Water Reactor Power Plant," dated January 1983 and NRC SER approval of the various Caldon Topical Reports regarding LEFM Check System have paved the way for the submittal of this proposed license amendment.

Further, NRC license approval or submittal of MUR power uprate efforts at several other commercial nuclear power plants, utilizing the Caldon LEFM Check or CheckPlus System, has created a significant precedent for this proposed change to be implemented at IP3. Among those of most recent note include Comanche Peak 1, Watts Bar, Beaver Valley 1 & 2, Susquehanna 1 & 2, Sequoyah 1 & 2, Waterford 3, and Point Beach 1 & 2, among others.

The MUR power uprate license changes at several of these plants were reviewed for lessons learned in the creation of the IP3 submittal. The IP3 license submittal was created using various parts of some of those utilities that have received their license change approval for the 1.4% MUR uprate change. Therefore, there is ample precedent and pre-approval of the Uprate and specific Caldon LEFM Check System methodology preceding this submittal by IP3.

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 IPN-02-041

PROPOSED TECHNICAL SPECIFICATION AND BASES CHANGES (MARKUP)

Proposed Technical Specification pages revised:

1.1-5	3.4.3-5
2.0-2	3.4.12-9
3.3.2-8	3.4.12-10
3.3.2-9	3.4.12-11
3.3.2-11	3.4.12-12
3.4.3-3	3.7.1-3
3.4.3-4	

Proposed Technical Specification Bases pages revised:

B 3.3.1-50
B 3.3.2-14
B 3.6.6-5
B 3.7.1-3
B 3.7.6-2

ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

1.1 Definitions

MODE (continued) — vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

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

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

a. Described in FSAR Chapter 13, Initial Tests and Operations;

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

c. Otherwise approved by the Nuclear Regulatory Commission.

QUADRANT POWER TILT RATIO (QPTR) — QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

RATED THERMAL POWER (RTP) — RTP shall be a total reactor core heat transfer rate to the reactor coolant of ~~3025~~ MWt.

3067.4

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This curve does not provide allowable limits for normal operation.
(see LCO 3.4.1, Pressure, Temperature and Flow DNB limits, for DNB limits)

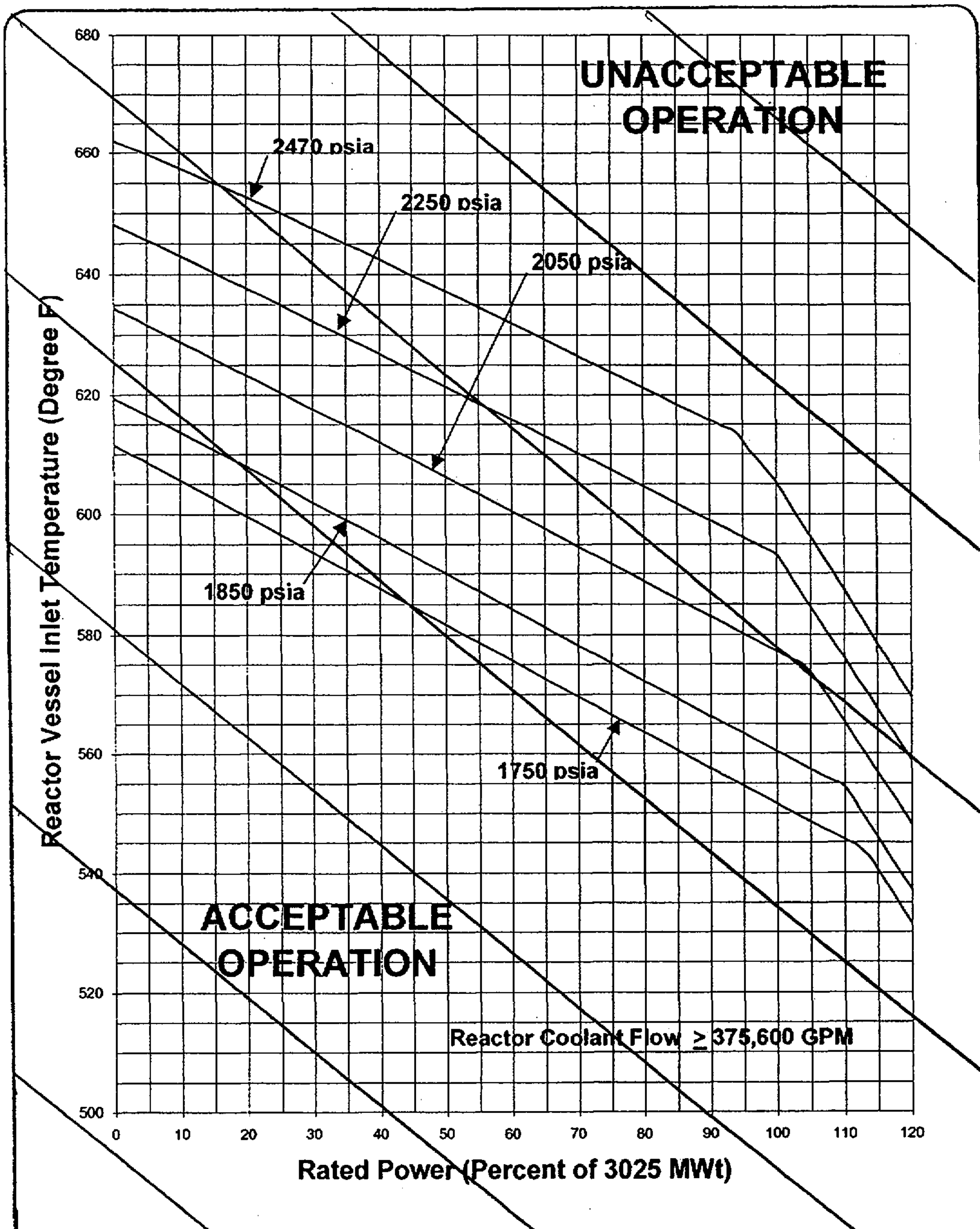


Figure 2.1-1
Rated Power (Percent of 3025 MWt)
100 PERCENT RATED POWER IS EQUIVALENT TO 3025 MWt
Pressures and temperatures do not include allowance for instrument error.

INSERT A

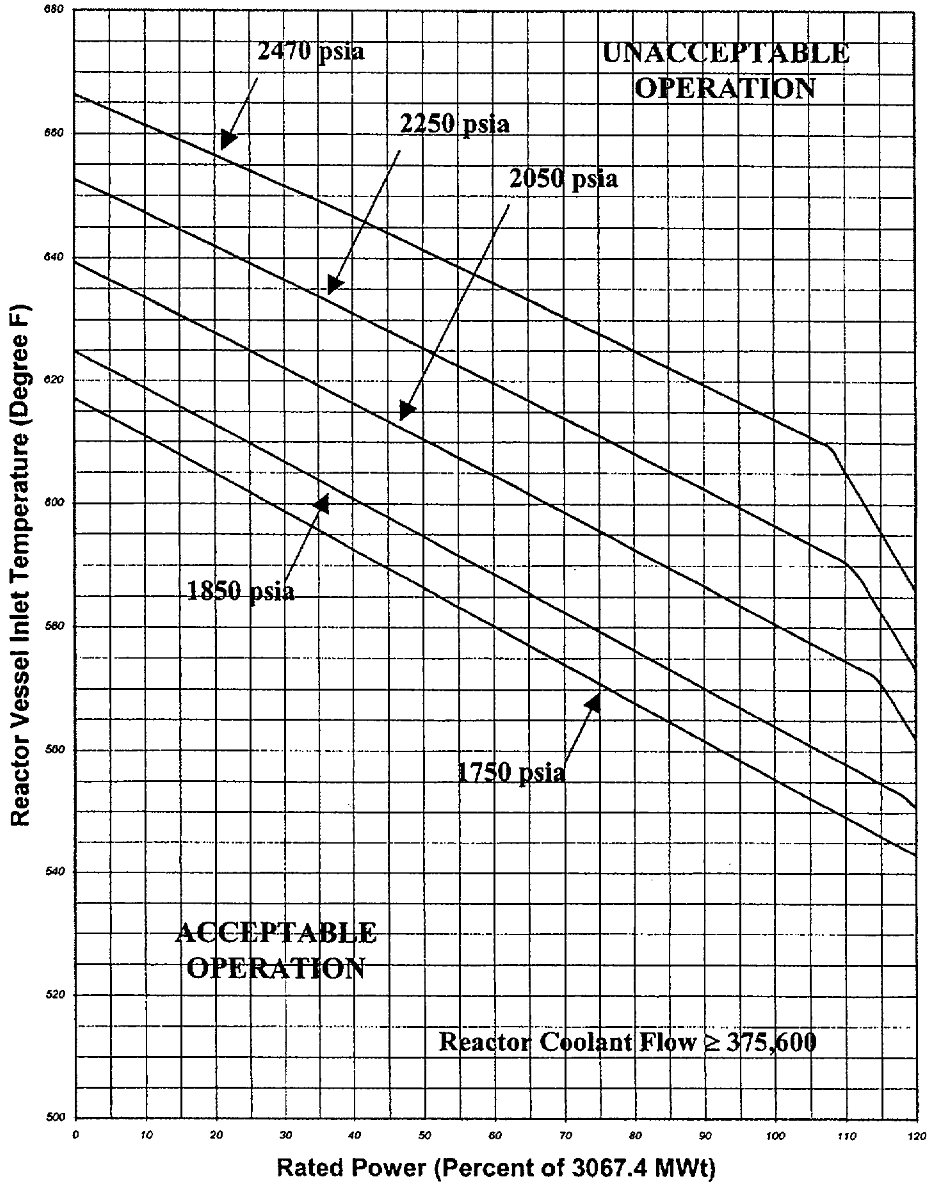


Figure 2.1-1
Rated Power (Percent of 3067.4 MWt)
100 PERCENT RATED POWER IS EQUIVALENT TO 3067.4 MWt
Pressures and temperatures do not include allowance for instrument error.

Table 3.3.2-1 (page 1 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Safety Injection					
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.6	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA
c. Containment Pressure-Hi	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ 4.80 psig
d. Pressurizer Pressure-Low	1,2,3(b)	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 1690 psig
e. High Differential Pressure Between Steam Lines	1,2,3	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	NA
f. High Steam Flow in Two Steam Lines	1,2(d),3(d)	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(c)
Coincident with T_{avg} - Low	1,2(d),3(d)	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 538 °F

(continued)

(a) Not used

(b) Above the Pressurizer Pressure interlock.

(c) Less than or equal to turbine first stage pressure corresponding to 54% full steam flow below 20% load, and increasing linearly from 54% full steam flow at 20% load to 110% full steam flow at 100% load, and corresponding to 110% full steam flow above 100% load. Time delay for SI ≤ 6 seconds.

(d) Except when all MSIVs are closed.

120%

120 %

Table 3.3.2-1 (page 2 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Safety Injection (continued)					
g. High Steam Flow in Two Steam Lines	1,2(d),3(d)	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(c)
Coincident with Steam Line Pressure-Low	1,2(d),3(d)	1 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 500 psig
2. Containment Spray					
a. Manual Initiation	1,2,3,4	2 per train, 2 trains	B	SR 3.3.2.6	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA
c. Containment Pressure (Hi-Hi)	1,2,3	2 sets of 3	E	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ 24 psig

(continued)

(c) Less than or equal to turbine first stage pressure corresponding to 54% full steam flow below 20% load, and increasing linearly from 54% full steam flow at 20% load to 110% full steam flow at 100% load, and corresponding to 110% full steam flow above 100% load. Time delay for SI ≤ 6 seconds.

(d) Except when all MSIVs are closed.

120%

120%

Table 3.3.2-1 (page 4 of 6)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. Steam Line Isolation					
a. Manual Initiation	$1,2^{(d)},3^{(d)}$	2 per steam line	F	SR 3.3.2.6	NA
b. Automatic Actuation Logic and Actuation Relays	$1,2^{(d)},3^{(d)}$	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA
c. Containment Pressure (Hi-Hi)	$1,2^{(d)},$ $3^{(d)}$	2 sets of 3	E	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ 24 psig
d. High Steam Flow in Two Steam Lines	$1,2^{(d)},$ $3^{(d)}$	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(c)
Coincident with T_{avg} - Low	$1,2^{(d)},$ $3^{(d)}$	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 538°F
e. High Steam Flow in Two Steam Lines	$1,2^{(d)},$ $3^{(d)}$	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(c)
Coincident with Steam Line Pressure - Low	$1,2^{(d)},$ $3^{(d)}$	1 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 500 psig

(continued)

(c) Less than or equal to turbine first stage pressure corresponding to 54% full steam flow below 20% load, and increasing linearly from 54% full steam flow at 20% load to 110% full steam flow at 100% load, and corresponding to 110% full steam flow above 100% load. Time delay for SI ≤ 6. seconds

(d) Except when all MSIVs are closed. 120%

120%

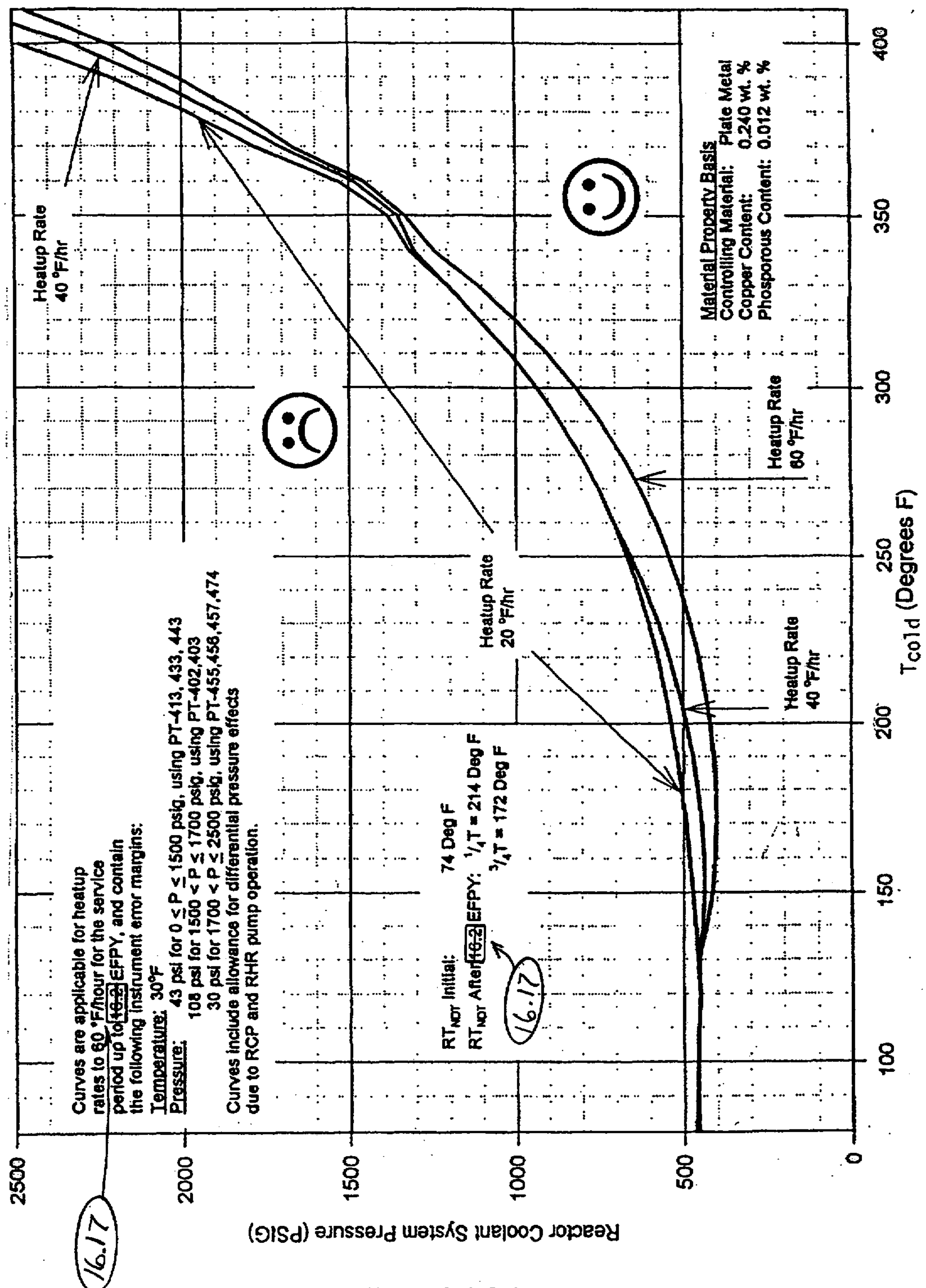


Figure 3.4.3-1:
Heatup Limitations for the Reactor Coolant System

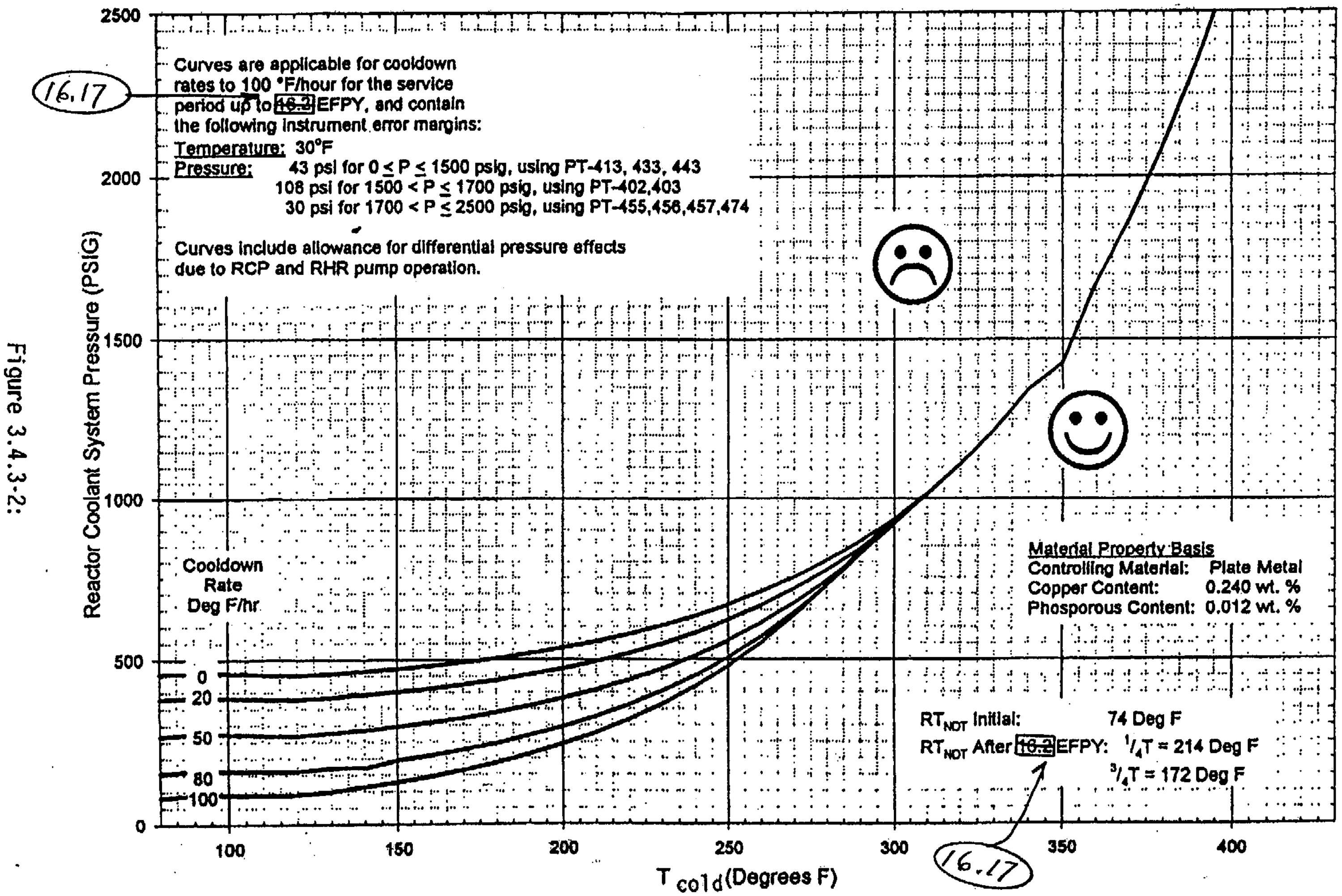


Figure 3.4.3-2:
Cooldown Limitations for the Reactor Coolant System

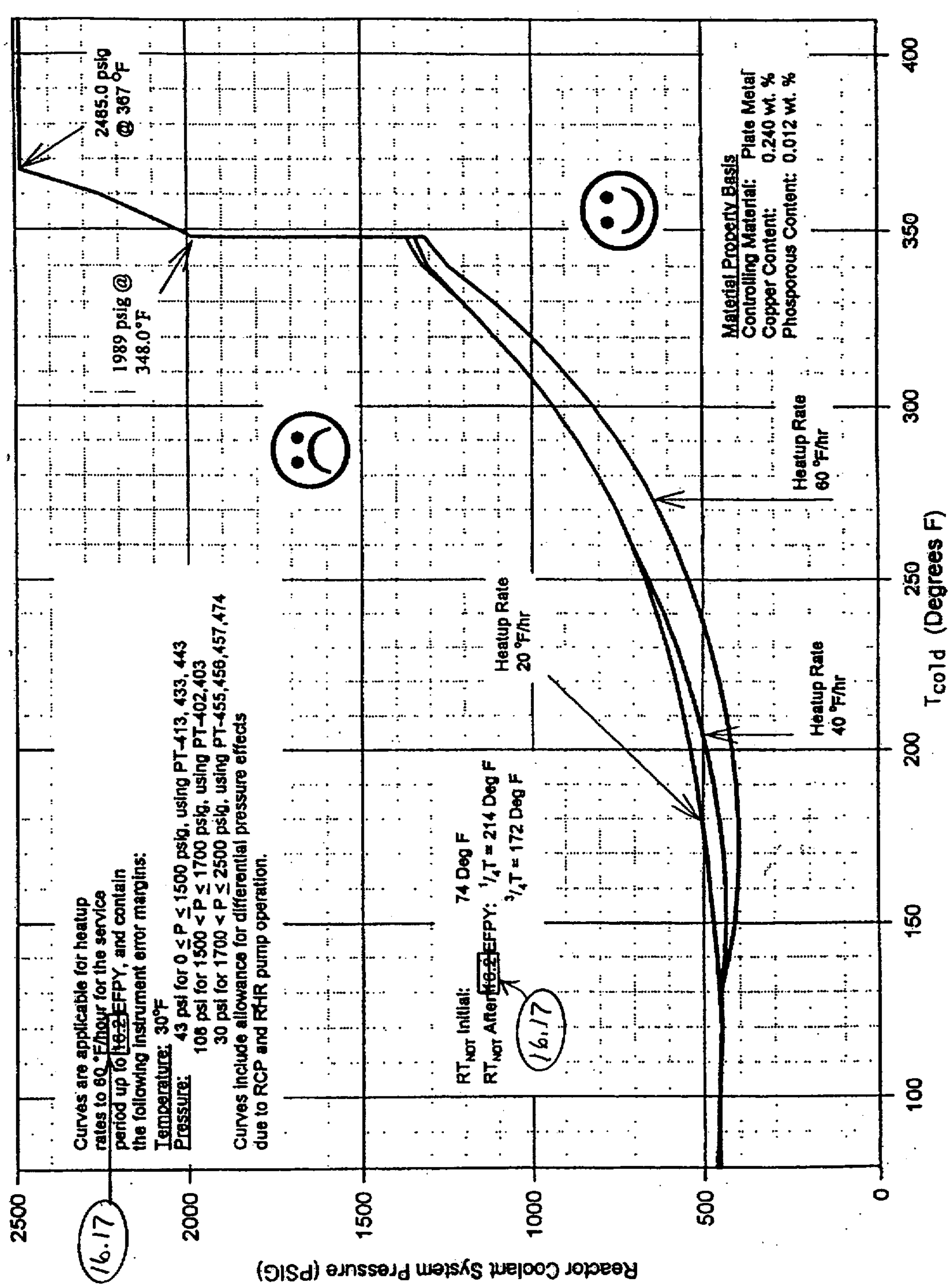
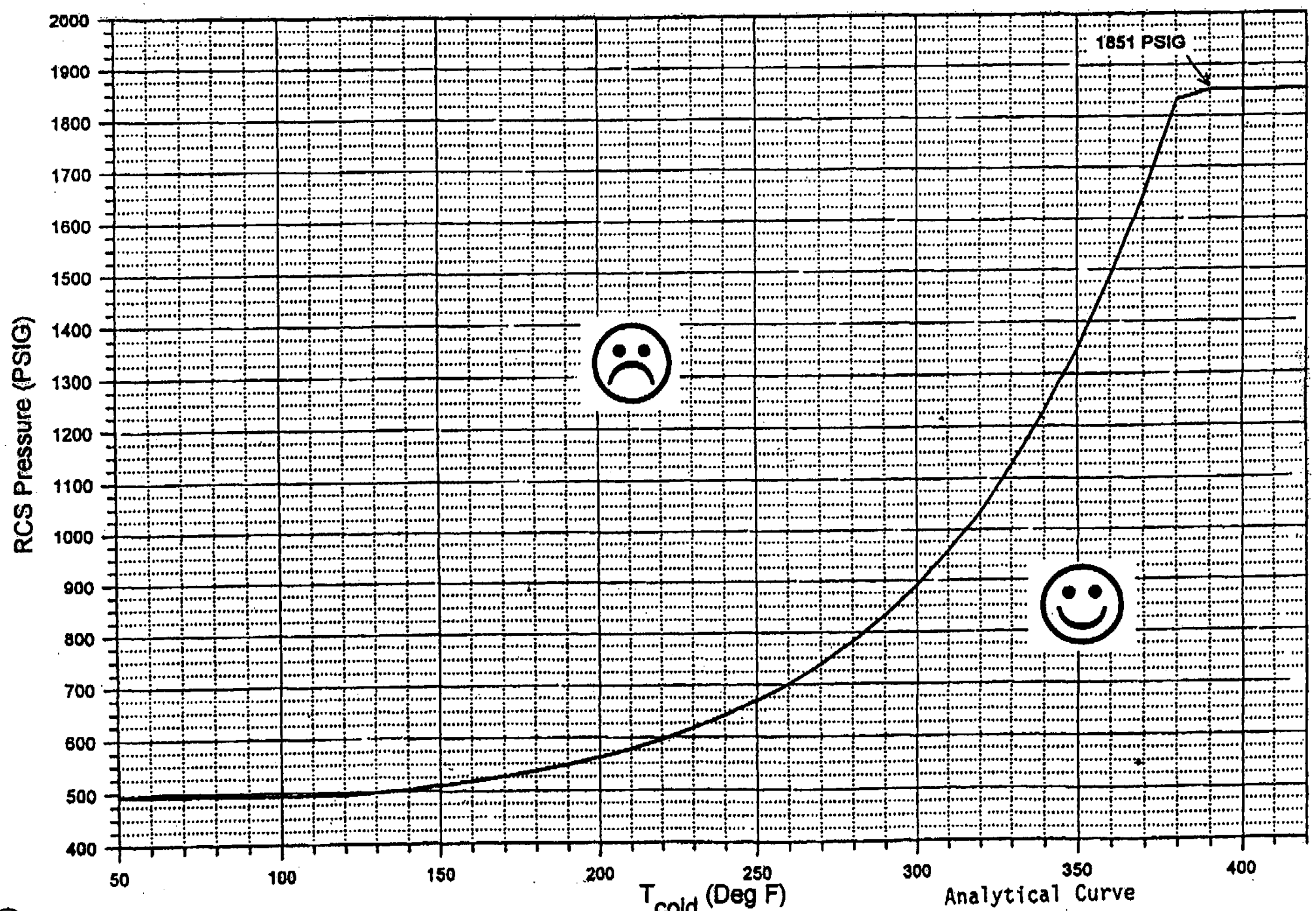


Figure 3.4.3-3:
Hydrostatic and Inservice Leak Testing Limitations for the Reactor Coolant System



Analytical Curve

Figure 3.4.12-1: Maximum Allowable Nominal PORV Setpoint for LTOP (OPS), $\boxed{16.2}$ EFPY

16.17

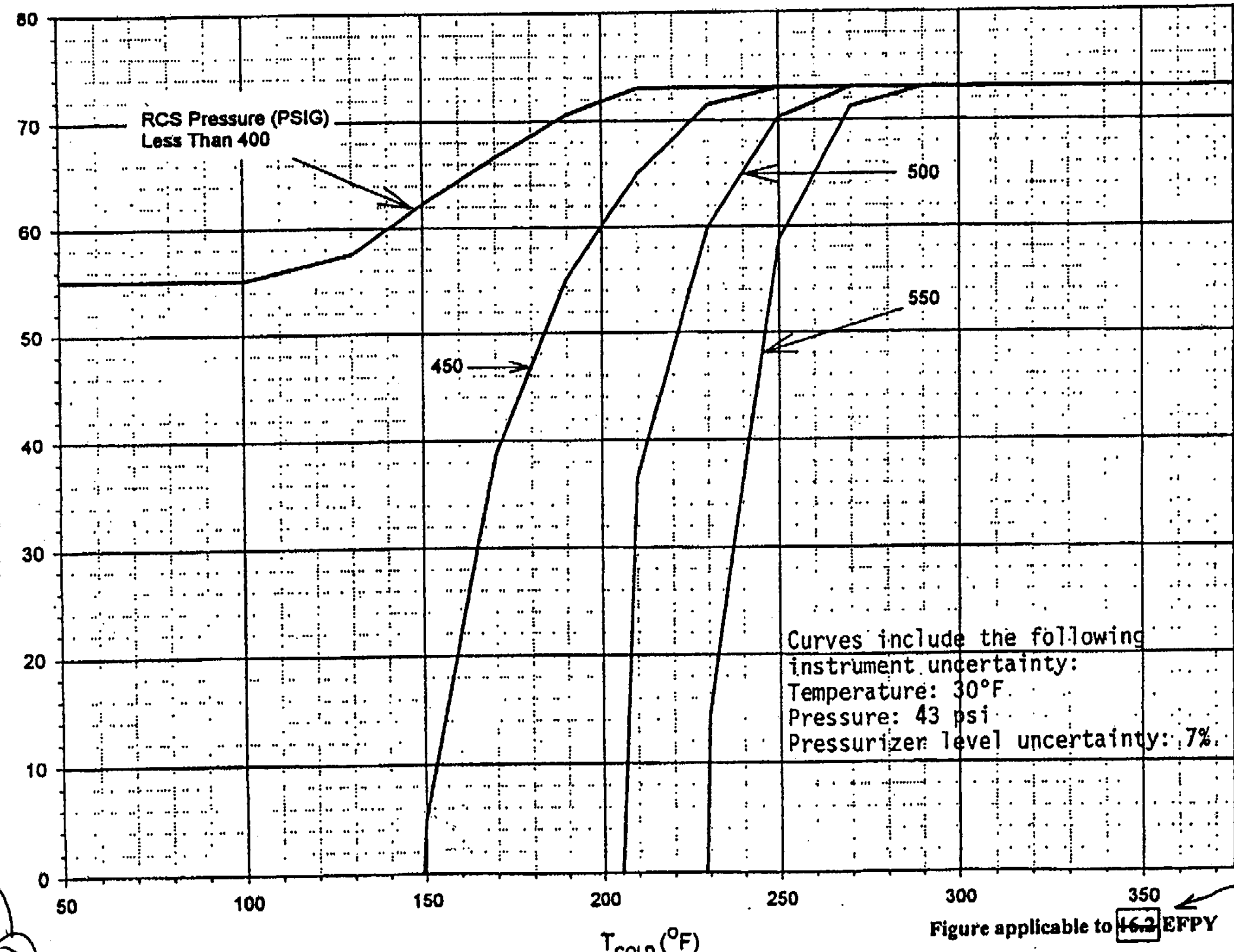
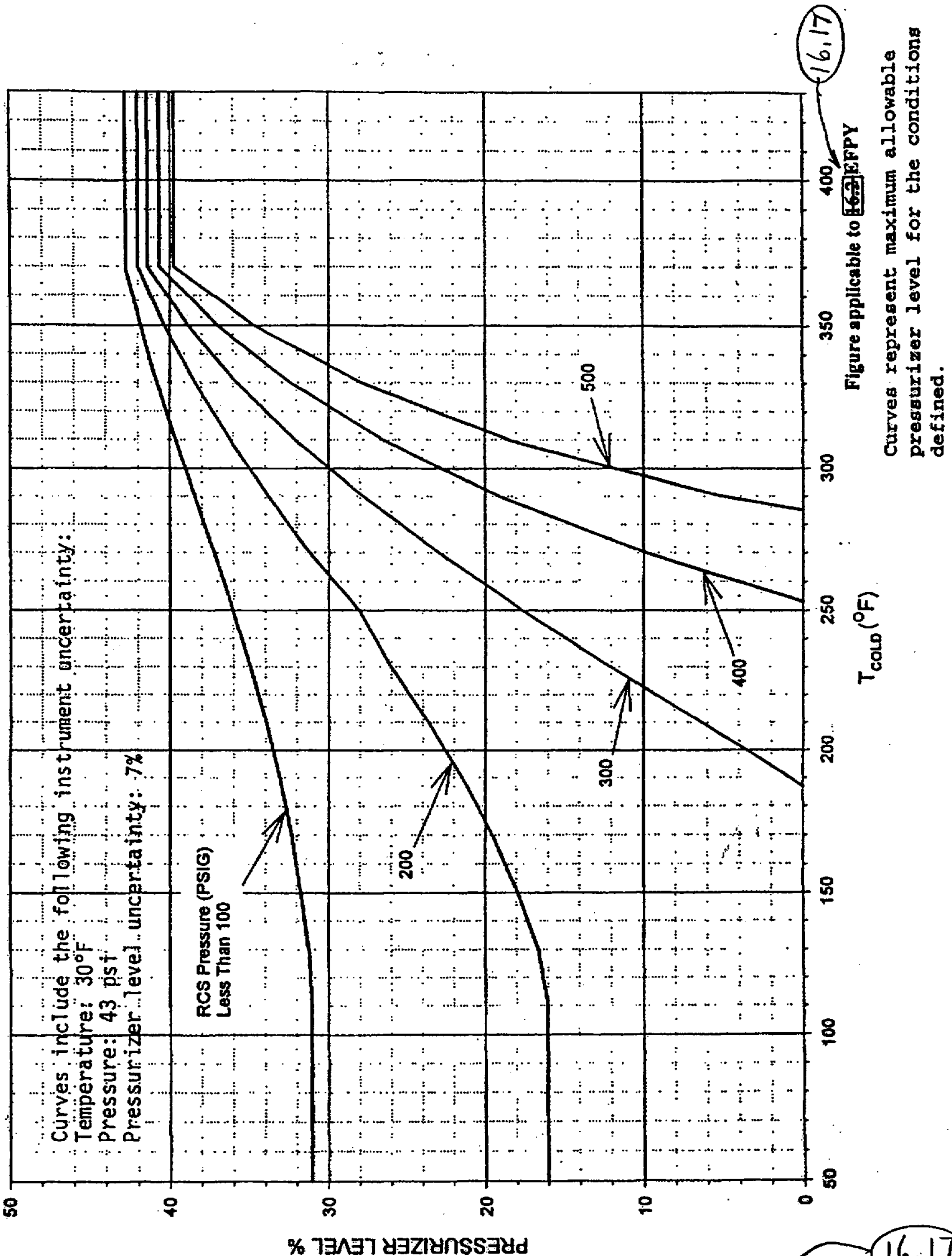


Figure applicable to ~~16.2~~ 16.17 EFPY

Curves represent maximum allowable pressurizer level for the conditions defined.

Figure 3.4.12-2: Pressurizer Limitations for OPS Inoperable, ~~16.2~~ 16.17 EFPY
(Up to one charging pump capable of feeding the RCS)



16.17

Figure 3.4.12-3: Pressurizer Limitations for OPS Inoperable. ~~16.2~~ EFPY
(Up to three charging pumps and/or one safety injection pump capable of feeding RCS)

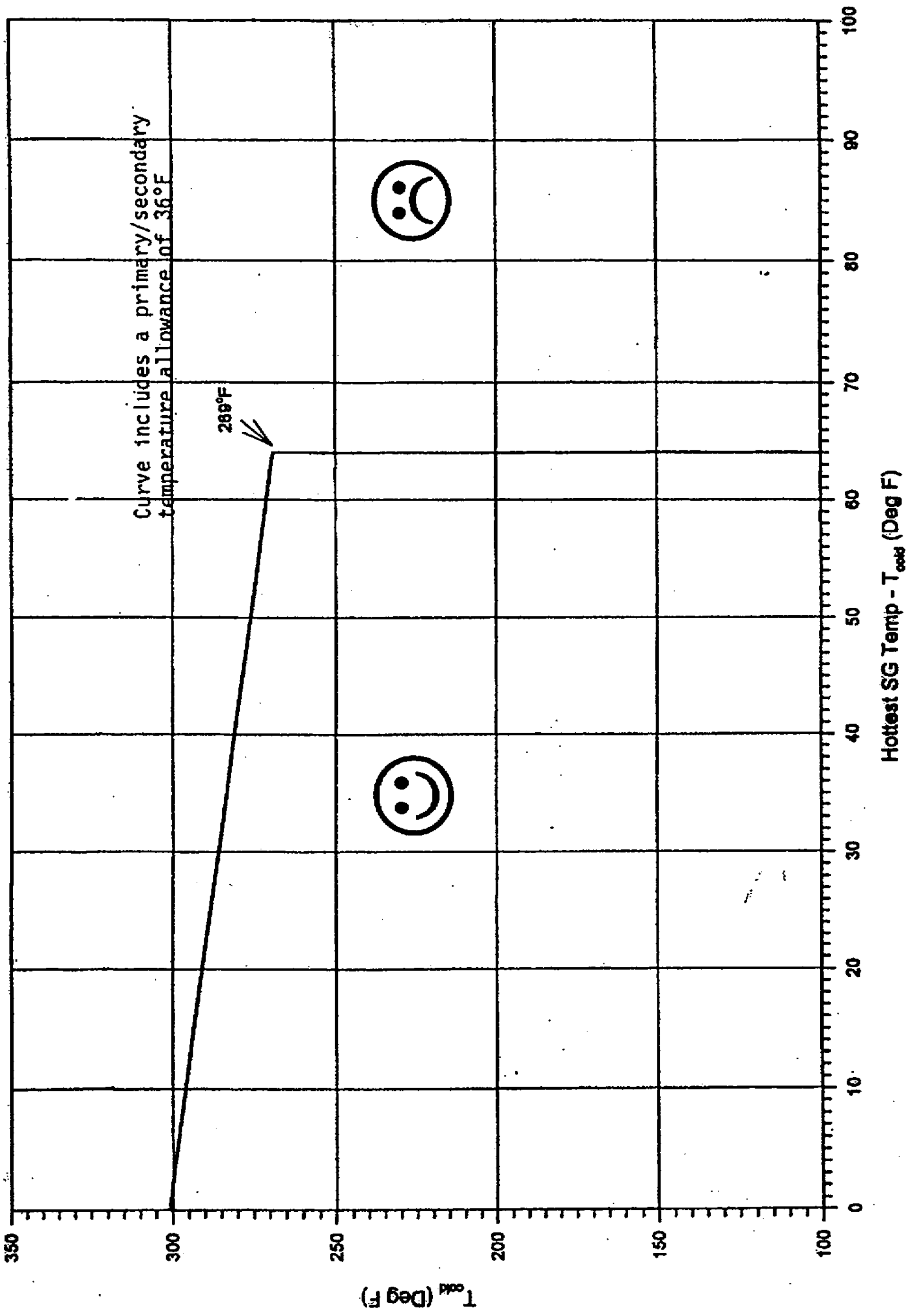


Figure 3.4.12-4: Secondary Side Limitations for RCP Start With Secondary Side Hotter than Primary Side, ~~16.2~~ EFPY

16.17

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus
Applicable Neutron Flux Trip Setpoint in Percent of RATED THERMAL POWER

MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	APPLICABLE Neutron Flux Trip Setpoint (% RTP)
4	≤ ~~61~~ 60
3	≤ ~~42~~ 41
2	≤ ~~23~~ 22

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.1.2 0.6%

SR 3.3.1.2 compares the calorimetric heat balance calculation to the NIS channel output every 24 hours. If the calorimetric exceeds the NIS channel output by > 2% RTP, the NIS is not declared inoperable, but must be adjusted. If the NIS channel output cannot be properly adjusted, the channel is declared inoperable.

Two Notes modify SR 3.3.1.2. The first Note indicates that the NIS channel output shall be adjusted consistent with the calorimetric results if the absolute difference between the NIS channel output and the calorimetric is > 2% RTP. The second Note clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and

0.6%

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO and APPLICABILITY (continued)

The Allowable Value for high steam flow is a linear function that varies with power level. The function is a turbine first stage pressure corresponding to approximately 54% of full steam flow between 0% and 20% load to approximately 110% of full steam flow at 100% load. The nominal trip setpoint is similarly calculated.

120%

With the transmitters located inside the containment (RCS temperature and steam line flow) or inside the auxiliary feedwater building (steam pressure), it is possible for them to experience adverse steady state environmental conditions during an SLB event. Therefore, the Allowable Value reflects both steady state and adverse environmental instrument uncertainties.

This Function must be OPERABLE in MODES 1, 2, and 3 when any MSIV is open because a secondary side break or stuck open valve could result in the rapid depressurization of the steam line(s). SLB may be addressed by Containment Pressure High (inside containment) or by High Steam Flow in Two Steam Lines coincident with Steam Line Pressure – Low, for Steam Line Isolation, followed by High Differential Pressure Between Two Steam Lines, for SI. This Function is not required to be OPERABLE in MODE 4, 5, or 6 because there is insufficient energy in the secondary side of the unit to cause an accident.

2. Containment Spray

Containment Spray provides three primary functions:

1. Lowers containment pressure and temperature after an HELB in containment;

(continued)

BASES

BACKGROUND (continued)

b) Three fan cooler trains (i.e., five fan cooler units); or,

c) One containment spray train and any two fan cooler trains (i.e., at least three fan cooler units).

This last configuration, one containment spray train and two fan cooler trains, is the configuration available following the loss of any safeguards power train (e.g., diesel failure).

APPLICABLE SAFETY ANALYSES

The Containment Spray System and Containment Fan Cooler System limit the temperature and pressure that could be experienced following a DBA. The limiting DBAs considered are the loss of coolant accident (LOCA) and the steam line break (SLB). The LOCA and SLB are analyzed using computer codes designed to predict the resultant containment pressure and temperature transients. No DBAs are assumed to occur simultaneously or consecutively. The postulated DBAs are analyzed with regard to containment ESF systems, assuming the loss of one safeguards power train, which is the worst case single active failure and results in one train of Containment Spray and one train of Fan Coolers being rendered inoperable.

The analysis and evaluation show that under the worst case scenario, the highest peak containment pressure and temperature may result from either a LOCA or SLB, depending on the cycle specific analysis (Refs. 4 and 6). Both results meet the intent of the design basis. (See the Bases for LCO 3.6.4, "Containment Pressure," and LCO 3.6.5 for a detailed discussion.) The analyses and evaluations assume a unit specific power level of 100.6 ~~102%~~ and initial (pre-accident) containment conditions of 130°F and 2.5 psig and a service water inlet temperature of 95° F. The analyses also assume a response time delayed initiation to provide conservative peak calculated containment pressure and temperature responses.

For certain aspects of transient accident analyses, maximizing the calculated containment pressure is not conservative.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued) 3081.4

Q = Nominal NSSS power rating of the plant (including reactor coolant pump heat) in Mwt (i.e., ~~3037~~ Mwt);

K = Conversion factor, 947.82 (Btu/sec)/Mwt;

ws = Minimum total steam flow rate capability of the operable MSSVs on any one steam generator at the highest MSSV opening pressure, including tolerance and accumulation, as appropriate, in lb/sec. ($ws = 150 + 228.61 * (4 - V)$ lb/sec, where V = Number of inoperable safety valves in the steam line of the most limiting steam generator).

h_{fg} = Heat of vaporization for steam at the highest MSSV opening pressure including tolerance and accumulation, as appropriate, Btu/lbm (i.e.,608.5 Btu/lbm).

N = Number of loops in plant (i.e., 4).

The calculated reactor trip setpoint is further reduced by 9% of full scale to account for instrument uncertainty and then rounded down.

The MSSVs satisfy Criterion 3 of 10 CFR 50.36.

LCO

The accident analysis requires five MSSVs per steam generator to provide overpressure protection for design basis transients
100.6% occurring at ~~102%~~ RTP. An MSSV will be considered inoperable if it fails to open on demand. The LCO requires that five MSSVs be OPERABLE in compliance with Reference 2. This is because operation with less than the full number of MSSVs requires limitations on allowable THERMAL POWER (to meet ASME Code requirements). These limitations are according to Table 3.7.1-1 in the accompanying LCO, and Required Action A.1.

The OPERABILITY of the MSSVs is defined as the ability to open within the setpoint tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced.

(continued)

BASES

BACKGROUND (continued)

To ensure CST pressure is maintained within its design limits while limiting the amount of air in contact with the condensate, two Category I, 100% capacity breather valves are installed on the dome of the CST. CST venting is required for the CST to perform both its normal and emergency function. The venting function can be met by either of the CST breather valves or equivalent venting capacity.

A description of the CST is found in the FSAR, Section 10.2 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The CST provides cooling water to remove decay heat and the minimum amount of water in the condensate storage tank is the amount needed to maintain the plant for 24 hours at hot shutdown following a trip from full power. When the condensate storage tank supply is exhausted, city water will be used.

The CST satisfies Criteria 2 and 3 of 10 CFR 50.36.

LCO

100.6%

To satisfy accident analysis assumptions, the CST must contain sufficient cooling water to remove decay heat while in MODE 3 for 24 hours following a reactor trip from ~~102%~~ RTP. In doing this, it must retain sufficient water to ensure adequate net positive suction head for the AFW pumps during cooldown, as well as account for any losses from the steam driven AFW pump turbine. When the condensate storage tank supply is exhausted, city water will be used.

The CST level required is equivalent to a total volume of ≥ 360,000 gallons, which is based on holding the unit in MODE 3 for 24 hours. This basis is established in Reference 1. The CST total volume includes allowances for instrument accuracy and the unuseable volume in the CST.

(continued)

ATTACHMENT III TO IPN-02-041

1.4% MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE APPLICATION REPORT

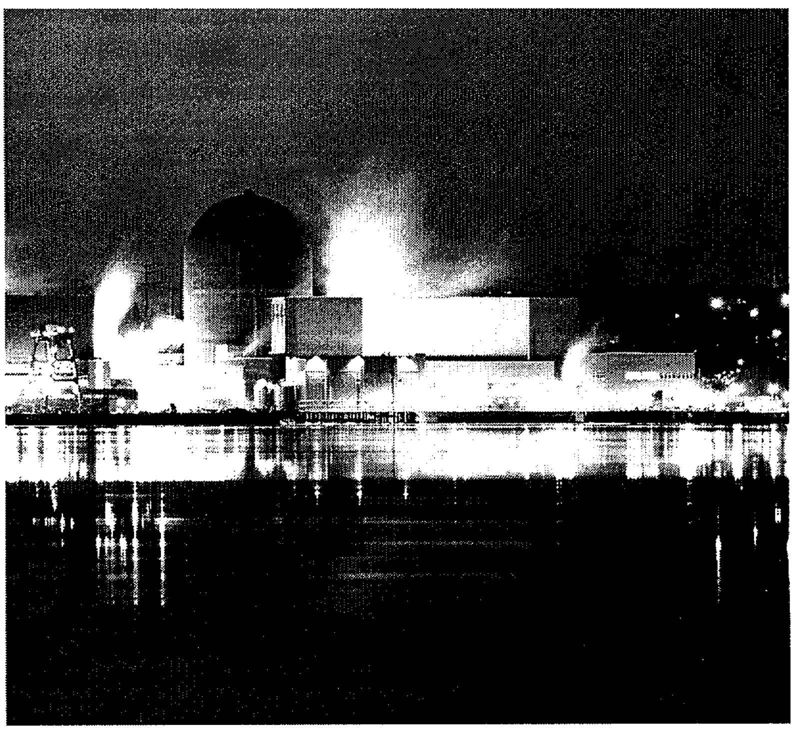
ENTERGY NUCLEAR OPERATIONS, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
DOCKET NO. 50-286

May 2002

Entergy Document No.:
IP3-RPT-MULT-03614,
Revision 0

Entergy Nuclear Operations, Incorporated

Indian Point Nuclear Generating Unit No. 3



1.4% Measurement Uncertainty Recapture Power Uprate Application Report

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LIST OF ACRONYMS

AC	Alternating Current
AFW	Auxiliary Feedwater
AFWS	Auxiliary Feedwater System
AISC	American Institute of Steel Construction
AMSAC	ATWS Mitigation System Actuation Circuitry
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrences
ARL	Alden Research Laboratory
ART	Adjusted Reference Temperature
ARV	Atmospheric Relief Valves
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	Anticipated Transient Without Scram
AV	Allowable Value
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
CC&F	Containment Cooling and Filtration
CCW	Component Cooling Water
CCWS	Component Cooling Water System
CF	Chemistry Factor
CFR	Code of Federal Regulations
CFU	Cumulative Fatigue Usage
CRDM	Control Rod Drive Mechanism
CSS	Containment Spray System
CST	Condensate Storage Tank
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
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EDS	Electrical Distribution System

LIST OF ACRONYMS (Cont'd)

EFPY	Effective Full-Power Year
EOL	End of Life, or License
EOP	Emergency Operating Procedures
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
FES	Final Environmental Statement
FF	Fluence Factor
FOA	Forced-Oil-Air Cooled
FOSAR	Foreign Object Search And Retrieval
FRV	Feedwater Regulating Valve
GDC	General Design Criteria
HELB	High Energy Line Break
HFP	Hot Full Power
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
IP3	Indian Point Unit 3
ISA	Instrument Society of America
LAR	License Amendment Request
LBB	Leak Before Break
LBLOCA	Large-Break Loss-of-Coolant Accident
LCO	Limiting Conditions of Operation
LEFM	Leading Edge Flow Meter
LOCA	Loss-of-Coolant Accident
LONF	Loss of Normal Feedwater
LOAC	Loss of All AC
LTOP	Low Temperature Overpressurization Protection
MT	Main Transformer
MCO	Moisture Carryover
MFP	Main Feedwater Pump

LIST OF ACRONYMS (Cont'd)

MOV	Motor Operated Valve
MPT	Main Power Transformer
MSIV	Main Steam Isolation Valve
MSLB	Main Steamline Break
MSR	Moisture Separator Reheater
MSS	Main Steam System
MSSV	Main Steam Safety Valve
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
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
OPΔT	Overpower Delta Temperature
OTΔT	Overtemperature Delta Temperature
ΔP	Delta Pressure
PID	Proportional Integral Derivative
PORV	Power-Operated Relief Valve
PRA	Probabilistic Risk Assessment
PRT	Pressurizer Relief Tank
PSS	Primary Sampling System
PT	Potential Transformer
P-T	Pressure-Temperature
PTS	Pressurized Thermal Shock
PU	Per Unit
PWR	Pressurized Water Reactor
QA	Quality Assurance
QSPDS	Qualified Safety Parameter Display System
RCCA	Rod Cluster Control Assembly
RCL	Reactor Coolant Loop
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RHRS	Residual Heat Removal System
RIS	Regulatory Issue Summary
RPS	Reactor Protection System
RSE	Reload Safety Evaluation

LIST OF ACRONYMS (Cont'd)

RTDP	Revised Thermal Design Procedure
RT_{NDT}	Reference Temperature for Nil Ductility Transition
RTP	Rated Thermal Power
RT_{PTS}	Reference Temperature, Pressurized Thermal Shock
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
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
SGTP	Steam Generator Tube Plugging
SGTR	Steam Generator Tube Rupture
SIS	Safety Injection System
SPDES	State Pollutant Discharge Elimination System
SSE	Safe Shutdown Earthquake
SST	Station Service Transformer
SWS	Service Water System
TDF	Thermal Design Flow
TSC	Technical Support Center
TSP	Tube Support Plate
UAT	Unit Auxiliary Transformer
UFSAR	Updated Final Safety Analysis Report
USE	Upper Shelf Energy
V&V	Verification and Validation
WOG	Westinghouse Owners Group

1 INTRODUCTION

1.1 BACKGROUND

The purpose of this report is to support United States Nuclear Regulatory Commission (NRC) review and approval of the Indian Point Unit 3 (IP3) 1.4% Measurement Uncertainty Recapture Power Uprate License Amendment Request (LAR). IP3 is presently licensed for a core power rating of 3025 MWt (see Section 1.4). The 1.4% power uprate, which is enabled through the use of more accurate feedwater flow measurement techniques, will increase the IP3 licensed core thermal power to 3067.4 MWt.

The June 1, 2000 NRC rulemaking regarding 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% in loss-of-coolant-accident (LOCA) analyses. This rulemaking provides licensees with the option of either maintaining the 2% 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% 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 the Code of Federal Regulations (CFR) 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 IP3 licensing bases analyses already include a 2% core power allowance, a demonstrated core power uncertainty of 0.6% effectively enables a 1.4% 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% core power uprate on plant systems, components, and analyses.

1.2 LICENSING APPROACH

All work supporting the IP3 1.4% 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% 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% power uprate.

Furthermore, Westinghouse has addressed the potential effects of the 1.4% 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 being submitted 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 performance 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 IP3 to evaluate the increase in the licensed core power from 3025 MWt to 3067.4 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 IP3 analyses assume either a nominal core or nominal NSSS power level. These analyses have either been evaluated or revised for the 1.4% 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% power uprate conditions.

2. Some IP3 analyses assume a core power level in excess of the 1.4% uprate core power level of 3067.4 MWt. These analyses were previously performed at a higher power level that bounds the current IP3 power level and the 1.4% uprate power level. This higher power level is typically the original IP3 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% uprate.

3. Most IP3 analyses already add a 2% uncertainty allowance to the nominal power level to account solely for power measurement uncertainty. These analyses have not been revised for the 1.4% uprate power level conditions because the sum of increased core power level (1.4%) and the improved power measurement accuracy (uncertainty less than 0.6%) is already bounded by the currently analyzed 2% uncertainty allowance.

   The power calorimetric uncertainty calculation described in Section 4 demonstrates that, with the Caldon Leading Edge Flow Meter (LEFM) instrumentation installed, the power measurement uncertainty (based on a 95% probability at a 95% confidence interval) is less than 0.6%. Since these analyses only need to account for the 0.6% power measurement uncertainty, the existing 2% uncertainty allowance can be allocated to account for the 0.6% uncertainty in the analyses and enable the 1.4% increase in licensed core thermal power. In addition, these analyses also employ other conservative assumptions that are unaffected by the 1.4% 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% power conditions, or do not model power at all. By definition, these analyses are unaffected by the 1.4% increase in core thermal power and have not been revised.

1.4 SUMMARY OF TECHNICAL SPECIFICATION CHANGES

The primary IP3 Technical Specification (Tech Spec) changes associated with 1.4% core thermal power uprate project are:

- A change to the core power from 3025 MWt to 3067.4 MWt in the definition of Rated Thermal Power (RTP) on page 1.1-5 of the Tech Specs,

- A change to the Reactor Core Safety Limits curves and the associated 100% core thermal power level identified as Figure 2.1-1 on page 2.0-2 of the Tech Specs,

- A change to the Applicable Neutron Flux Trip Setpoints (%RTP) based on a minimum number of operable main steam safety valves (MSSVs) as identified in Table 3.7.1-3 (page 3.7.1.1 of the IP3 Tech Specs), and

- A change to the high steam flow safety injection initiation Tech Spec Allowable Value (AV) for 100% power from 110% to 120% in Table 3.3.2-1 on pages 3.3.2-8, 3.3.2-9, and 3.3.2-11 of the Tech Specs.

There are secondary changes to the various heatup and cooldown figures and Low Temperature Overpressurization Protection (LTOP) figures to reflect a slight effect on the applicable service period for these figures (as described in Section 7.2 of this report). These Tech Spec Figures are 3.4.3-1, 3.4.3-2, 3.4.3-3, 3.4.12-1, 3.4.12-2, 3.4.12-3, and 3.4.12-4.

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% 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% power calorimetric uncertainty which, in turn, justifies the 1.4% power uprate. This section also addresses the potential effects on Reactor Protection System (RPS)/Engineered Safety Features Actuation System (ESFAS) 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 (RHR), and control systems) and components (e.g., reactor vessel, pressurizer, reactor coolant pumps (RCPs), 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 areas (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 balance of plant (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.

- **Section 13** provides the overall report conclusion.

The analyses and evaluations described herein demonstrate that all applicable acceptance criteria will continue to be met based on operation at the 1.4% power uprate conditions at 3067.4 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 DESIGN PARAMETERS

The NSSS primary and secondary system design parameters are the fundamental system condition inputs (temperatures, pressures, and flow) that are used as the basis for all of the NSSS analyses and evaluations.

Revised design parameters were developed to reflect the 1.4% increase in the IP3 licensed core power from 3025 MWt to 3068 MWt (3067.4 MWt conservatively rounded up to 3068 MWt). The new parameters are shown in Table 2-1. As discussed throughout this report, these parameters have been reconciled with the applicable systems and components evaluations, as well as safety analyses, performed in support of the 1.4% power uprate.

2.1 INPUT ASSUMPTIONS

The IP3 NSSS primary and secondary system design parameters were developed based on conservative inputs such as a conservatively low primary system flow (thermal design flow (TDF)) and bounding steam generator tube plugging (SGTP) levels. The resulting primary and secondary-side design conditions will bound actual plant operations at the 3067.4 MWt uprate power level.

The method and mathematical model used to calculate the IP3 design parameter values in Table 2-1 use basic thermal/hydraulic and engineering principles, including energy and mass balances. Westinghouse has codified the method and mathematical model to facilitate more efficient performance of the calculations. The code used to determine the NSSS design parameters is called SGPER (Steam Generator PERformance). Explicit NRC approval is not needed for SGPER, since it is used to facilitate fundamental engineering calculations that could be performed by hand. The code, method, and mathematical model have been successfully used to support all previous uprates for Westinghouse plants.

Three sets of design performance parameters were developed for IP3. The following assumptions were common to all three sets:

- Westinghouse Model 44F steam generators,
- NSSS uprated power level of 3082 MWt (3068 MWt core power + a conservatively high value of 14 MWt net heat input from the primary Reactor Coolant System (RCS) RCP),
- Nominal feedwater temperature (T_{feed}) of 427.4°F,
- Westinghouse 15x15 VANTAGE+ fuel, and
- Total design core bypass flow of 5.2% that accounts for Intermediate Flow Mixing (IFM) Grids and a protective bottom grid.

The three sets are distinguished as follows:

Set 1 presents the parameter values applicable for most NSSS system and component analyses. They maintain the current analysis basis of TDF = 89,700 gpm/loop, reactor vessel T_{avg} of 571.5°F, and 0% SGTP.

Set 2 presents the parameter values applicable for the Updated Final Safety Analysis Report (UFSAR) Chapter 14 Safety Analysis. They maintain the current safety analysis basis of TDF = 80,900 gpm/loop, reactor vessel T_{avg} of 571.5°F, and 24% SGTP.

Set 3 presents a separate set of parameter values for Non-LOCA Safety Analyses that address asymmetric SGTP. These parameters maintain the current safety analysis basis of TDF = 80,900 gpm/loop, reactor vessel T_{avg} of 574.7°F, and 24% avg/30% peak SGTP.

The 1.4% power uprate resulted in only minor changes to the values for some of the NSSS design parameters relative to the current design basis. These changes were evaluated for each of the analytical areas discussed in this report. For the base 0% SGTP case, they are:

- Primary system hot-leg coolant temperature (T_{hot}) increased by 0.4°F,
- Primary system cold-leg coolant temperature (T_{cold}) decreased by 0.4°F, and
- Steam pressure decreased by 9 psia.

TABLE 2-1

1.4% UPRATE NSSS DESIGN PARAMETERS – IP3

THERMAL DESIGN PARAMETERS	------------1.4% Uprate ------------ Set 1	Set 2	Set 3
NSSS Power, %	101.4	101.4	101.4
MWt	3,082	3,082	3,082
10^6 Btu/hr	10,516	10,516	10,516
Reactor Power, MWt	3,068	3,068	3,068
10^6 Btu/hr	10,468	10,468	10,468
Thermal Design Flow, Loop gpm	89,700	80,900	80,900
Reactor 10^6 lb/hr	136.3	123.4	122.9
Reactor Coolant Pressure, psia	2250	2250	2250
Core Bypass, %	5.2	5.2	5.2
Reactor Coolant Temperature, °F			
Core Outlet	603.7	607.0	610.0
Vessel Outlet	600.8	603.8	606.9
Core Average	574.2	574.6	577.9
Vessel Average	571.5	571.5	574.7
Vessel/Core Inlet	542.2	539.2	542.5
Steam Generator Outlet	541.9	538.9	542.2
Steam Generator			
Steam Temperature, °F	512.7	498.9	502.4
Steam Pressure, psia	762	674	696
Steam Flow, 10^6 lb/hr total	13.26	13.23	13.24
Feed Temperature, °F	427.4	427.4	427.4
Moisture, % max.	0.10	0.10	0.10
Tube Plugging, %	0	24	24
Zero-Load Temperature, °F	547	547	547
HYDRAULIC DESIGN PARAMETERS			
Mechanical Design Flow, gpm total	396,400 (99,100 per loop)		
Minimum Measured Flow, gpm total			
Technical Specifications	375,600		
Fuel/Core Analyses	330,800		
Minimum Allowable Flow, gpm total (other plant analyses)	366,800		

3 CALDON LEFM CALCULATION

3.1 CALDON LEFM ULTRASONIC FLOW MEASUREMENT

The Caldon 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 NRC for power uprate applications as documented in Caldon Topical Reports ER-80P and ER-160P. At IP3, there are LEFM Check Flow Elements in each of the four feedwater lines, located at approximately six 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 will be transmitted to the plant computer for real-time calculation of reactor thermal power. The mass flow and temperature outputs will also be 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 is out of service as described in Section 3.3.

3.3 CALDON LEFM CHECK 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 3067.4 MWt may continue for the allowed outage time. The allowed outage time for operation at the 1.4% uprate level with an LEFM out of service is seven days, as long as steady-state conditions persist during the seven days (i.e., no power changes in excess of 10% during the period). There are five bases for this proposed time period:

- IP3 Operations personnel will operate based on alternate plant instruments, which will be benchmarked to the LEFMs last good reading as soon as the LEFM would become 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 seven-day period as long as steady-state conditions persist.

- It is considered prudent to provide IP3 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 an eight-hour shift, the seven days gives plant personnel ample time to trouble shoot, repair and verify normal operation of the LEFM System within its original uncertainty bounds at the same power level as before the failure.

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

If the plant experiences a power change of greater than 10% during the seven-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 seven-day outage period is exceeded, then the plant will operate at a power level consistent with the accuracy of the alternate plant instruments. The plant will implement procedures and guidance as required according to operator actions when the LEFM Check System is unavailable.

The LEFM Check System at IP3 is installed in each of the four feedwater lines. Failure of any one of the LEFMs will result in calculation of thermal power based on operation of the operable LEFMs and on operation of alternate plant instruments (flow nozzle and thermocouples) in the feedwater line with the LEFM out of service. If more than one LEFM is out of service, substitute values will be used for the associated feedwater lines, permitting operation with all four LEFMs out of service at a power level consistent with operation entirely on alternate instruments.

3.4 MAINTENANCE AND CALIBRATION

Maintenance of the Caldon LEFM will be 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 will be performed using site-specific procedures developed from ER-80P and the User's Manual.

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

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

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

IP3 I&C personnel will be trained per the I&C training program on the LEFM System before work or calibration may be performed. Formal training by Caldon will be provided to site personnel.

The LEFM Check is designed and manufactured in accordance with Caldon's 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 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 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 or system grooming will be performed by qualified personnel. At IP3 the LEFM Check System is 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 IP3'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 Ultrasonic Feedwater Measurement System (which also supplies feedwater temperature values), all other instrument components that provide fluid condition data for calculation of RTP will be controlled, calibrated and performance monitored to the conditions represented in the overall calorimetric uncertainty evaluation done for the IP3 1.4% power uprate.

The IP3 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 was originally installed at IP3 in 1982. The original electronic unit was upgraded with the Caldon LEFM 8300 electronic unit in 1997. The upgrade to the LEFM Check Electronic Unit, which meets the requirements of the approved Topical Reports ER-80P and ER-160P, is scheduled for the Summer of 2002. Since installation the LEFM has been used to provide trend basis data for tracking the alternate plant instruments to be used when the LEFM is out of service.

The Caldon LEFM Check System will be installed at IP3 in the Summer of 2002. The installations will be 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 to be installed at IP3 is representative of the Caldon LEFM Check System discussed in the Topical Report ER-80P, and will be bounded by the requirements set forth in this topical report.

3.6 UNCERTAINTY DETERMINATION METHODOLOGY

The methodology used to calculate the Caldon LEFM Check uncertainties is consistent with ASME PTC 19.1 and ISA 67.04 as approved in Topical Report ER-80P. IP3 currently uses this methodology in the development of the majority of the IP3 calculated instrument uncertainties.

With respect to the Caldon LEFM Check uncertainties, uncertainty calculations have been performed and determined a mass flow accuracy of better than 0.5% of rated flow for IP3.

Additionally, the Caldon LEFM Check uncertainty calculations are performed to achieve a 95% confidence interval, 95% probability flow measurement.

IP3 maintenance procedures and Caldon LEFM system operating instructions will 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. The Caldon LEFM Check flow element calibrations were based upon Alden Research Laboratory (ARL) testing of a population of seven flow elements with identical inside diameters and dimensions. The LEFM Check flow elements at IP3 are installed 5.8 diameters downstream from two non-planar elbows separated by 10 diameters. The uncertainty analysis expressly considers the additional uncertainty for these features, and their effects on the LEFM Check flow measurement. Further, the actual plant velocity profiles at IP3 have been compared to straightpipe profiles at ARL and the effects on the LEFM Check measurement have been thoroughly addressed in Caldon Report ER-262. These measurements assure that the actual LEFM Check measurements at IP3 are bounded by the uncertainty analysis and addressed appropriately in Topical Report ER-80P.

4 CONTROL AND PROTECTION SYSTEM SETPOINTS AND UNCERTAINTIES

Westinghouse WCAP-12128, "Westinghouse Improved Thermal Design Procedure Instrument Uncertainty Methodology for New York Power Authority Indian Point Unit 3," dated January 1989 provides the basis for the (RTDP) uncertainties that are used in the IP3 UFSAR Chapter 14 safety analyses. These include T_{avg} (rod) control, pressurizer pressure control, RCS flow measurement (calorimetric) and indication, and power measurement (calorimetric). The effect of the 1.4% power uprate on the power calorimetric and RTDP uncertainties is discussed in the following subsections.

4.1 POWER CALORIMETRIC

While not covered by WCAP-15824, "Power Calorimetric for the 1.4% Uprating for Entergy Nuclear Indian Point Unit 3," 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 2.0% of RTP. The 1.4% 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 2.0% 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 LEFM measurement system to determine total feedwater flow. The uncertainty allowance for feedwater system flow is ±0.5%. The flow error, in combination with the remaining uncertainty components, results in a total 95/95 power measurement uncertainty of ±0.6% RTP. A power measurement uncertainty of ±0.6% allows a power uprate of 1.4% RTP. The methodology used to determine the power calorimetric uncertainties is documented in WCAP-15824.

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 IP3 design parameters for the 1.4% power uprate conditions. Therefore, the 1.4% power uprate does not require changes to the uncertainties documented in WCAP-12128 for these controllers.

4.2.2 Reactor Coolant System Flow Calculation

The RCS flow calculation uses nominal plant conditions for feedwater temperature and steam pressure as part of the input assumptions for the uncertainty calculations. The small changes in these plant parameters due to the 1.4% power uprate conditions do not change the final calculated RCS flow uncertainties. Therefore, the 1.4% power uprate does not require changes to these uncertainties as documented in WCAP-12128.

4.3 RPS/ESFAS UNCERTAINTIES

The license basis accident analyses, which model Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS) initiations for mitigation, have been reviewed. It has been confirmed that their results remain acceptable (see Section 8.0) using the currently applied safety analyses assumptions (some of which are Tech Spec Analytical Limits) and the initial condition inputs based on the 1.4% 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% power uprate on total uncertainties calculated as the basis for implemented RPS/ESFAS setpoints and Tech Spec documented Allowable Values (AVs). The fluid condition changes assumed for this review are given in Section 2.0 of this report. The results of this uncertainty evaluation are provided below.

4.3.1 RPS Functions

Power Range Neutron Flux – High

As a result of T_{avg} being approximately unchanged by the 1.4% power uprate condition, aggregate neutron leakage (vessel) characteristics are expected to be unchanged and flux monitoring sensitivity, therefore, also unchanged. The existing total uncertainty for this protective function includes a full 2.0% calorimetric allowance and this value will not be reduced as a result of the 1.4% power uprate. The existing uncertainty value for this function, therefore, is not changed as a result of the 1.4% power uprate. However, the Tech Spec modified Nuclear Instrumentation System (NIS) Power Range High Flux trip setpoints associated with the plant degraded conditions caused by having one, two or three MSSVs inoperable on any Main Steamline must be revised. See proposed revisions to the Tech Spec Table 3.7.1-1 for the new limiting setpoints required as a result of the 1.4% power uprate.

Power Range Neutron Flux – Low

The conditions described above for the high trip function of the NIS power range are also applicable (except for the MSSVs consideration) to the low trip function. Power ascension to an unchanged full power T_{avg} value will ensure similar sensitivity conditions at low power when transitioned from the current licensed power level to the 1.4% uprate power level. Calorimetric allowance is also the same. The existing uncertainty value for this function, therefore, is not changed as a result of the 1.4% power uprate.

Overtemperature ΔT (OTΔT)

RCS full-power T_{hot} and T_{cold} values will be slightly changed (full-power ΔT will increase by ~0.8°F) but uprate full-power T_{avg} is intended to be approximately the same as with the previous licensed power level. New full-power values are all within existing instrument spans and the T_{hot} 3-point radial sensing arrangement should be more than adequate to accurately monitor any changes that might occur relative to thermal streaming affects in the hot legs. The pressurizer pressure penalty is virtually unaffected since containment temperature conditions are virtually unaffected by the 1.4% uprate power, thus ensuring no meaningful change to the impulse leg correction for this variable. See the NIS discussions above for confirmation that the FΔI penalty is unaffected. The existing uncertainty value for this function, therefore, is unaffected as a result of the 1.4% power uprate.

Overpower ΔT (OPΔT)

See discussion on OTΔT above relative to T_{hot}, T_{cold} and T_{avg} considerations with the 1.4% power uprate. The overpower protective algorithm does not utilize inputs from any other variables. The existing uncertainty value for this function, therefore, is unaffected as a result of 1.4% power uprate.

Pressurizer Pressure – Low

Nominal pressurizer pressure (i.e. 2250 psia) is unaffected by the 1.4% power uprate. The pressurizer pressure monitoring system is unaffected since containment temperature conditions are virtually unaffected by the 1.4% power uprate, thus ensuring no meaningful change to the impulse leg correction for this variable. The existing uncertainty value for this function, therefore, is unaffected as a result of the 1.4% power uprate.

Pressurizer Pressure – High

See "Pressurizer Pressure – Low" above.

Pressurizer Water Level – High

As discussed in the sections addressing pressurizer pressure, impulse leg corrections are unaffected as a result of the insignificant changes in containment temperature with the 1.4% power uprate. Also, assumed initial conditions of pressurizer level (a programmed value which is a function of T_{avg}) are also unchanged since it is anticipated that full-power T_{avg} will be unaffected with the implementation of the 1.4% power uprate (thus keeping the full power level control point the same). Should a small change in Average T_{avg} actually occur, no consequences would arise since the existing level program ascends to a final level value at Average T_{avg} equaling 571.5°F (which is approximately 4°F above the current full-power Average T_{avg} value of between 567 and 568°F).

Reactor Coolant Flow – Low

Conservatively assuming that the total change in loop ΔT for the 1.4% power uprate is accomplished by reducing T_{cold} by the full 0.8°F, we would induce a specific gravity bias on flow monitoring uncertainty on the order of 0.1% of loop flow. This additional uncertainty is bounded by the margin existing in the currently implemented setpoint and the margin existing in the currently documented Tech Spec AV (used for determining loop operability).

Underfrequency RCPs (6.9 kV Bus)

Bus underfrequency relay uncertainties are not subject to influences associated with fluid parametric changes caused by the 1.4% power uprate. Also, nominal frequency 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% power uprate.

Steam Generator Narrow Range (NR) Water Level – Low-Low

Process Measurement uncertainties considered in the steam generator level calculation take into account specific effects such as fluid pressure/specific gravity variations and Reference Leg Growth effects. None of the uncertainties associated with these effects are significantly altered by the 1.4% power uprate conditions. Containment temperature under normal conditions is not significantly changed as a result of the 1.4% power uprate conditions, and as a result, impulse leg offsets and transmitter temperature effects are not effected by this change in plant power level. Steam generator pressure under 1.4 % power uprate conditions will be approximately 10 psi lower than under current power conditions. The static pressure effect magnitude is computed, however, on the basis of maximum SG pressure conditions (at zero power), which is unchanged for the 1.4% power uprate, and thus the static pressure effect is unchanged. The existing total loop uncertainty value for this function, therefore, is unaffected as a result of the 1.4% power uprate.

4.3.2 ESFAS Functions

Containment Pressure – High and High-High

Containment pressure sensing instrument uncertainties are not subject to influences associated with fluid system parametric changes caused by the 1.4% power uprate. Containment pressure and temperature limiting Initial Conditions for Containment Response Analyses are also unchanged as a result of the 1.4% power uprate influences. Therefore, implemented setpoints and Tech Spec AVs will remain the same.

Pressurizer Pressure – Low

Nominal pressurizer pressure (i.e. 2250 psia) is unaffected by the 1.4% power uprate. The pressurizer pressure monitoring system is unaffected since containment temperature conditions are virtually unaffected by the 1.4% power uprate thus ensuring no meaningful change to the impulse leg correction for this variable. The existing uncertainty value for this function, therefore, is not changed as a result of the 1.4% power uprate.

High Steam Flow in Two Lines

All fluid conditions assumed in the analysis of instrument uncertainties for this function are adequately conservative and thus ensure protection of analytical limits (which become re-defined on a percentage basis given the new definition of 100% plant power). However, to assure accurate propagation of the setpoint ramp function up to the newly defined 100% plant load point, various of the instruments in the loops (all of which have adequate range for the 1.4% power uprate condition) will have their calibration cardinal points extended, as appropriate, to encompass the 1.4% escalation in the parameter inputs to this function. Additionally, the 100% power AV in the Tech Spec Section 3.3.2, Table 3.3.2-1 is being revised upward to more closely reflect actual calculated limiting conditions. This is needed to preclude the possibility for unnecessarily identifying an inoperable condition in loop as-found conditions as the parameter values (i.e. first stage turbine pressure and main steam flow) increase under the 1.4% power uprate conditions. See Section 6.3.2, and associated markups for Tech Spec Section 3.3.2, Table 3.3.2-1.

T_{avg} – Low

RCS T_{hot} and T_{cold} values will be slightly changed, but the uprate T_{avg} is intended to be approximately the same as with the previous licensed power level. New full power values are all within existing instrument spans and the T_{hot} three-point radial sensing arrangement should be more than adequate to accurately monitor any changes that might occur relative to thermal streaming affects in the hot legs.

Steamline Pressure – Low

Steamline pressure sensing instrument uncertainties are not subject to influences associated with fluid parametric changes caused by the 1.4% power uprate. The only effect of the 1.4% power uprate on this function will be a slight shortening of the time to trip as a result of nominal main steam pressure being slightly lower.

Steam Generator Water Level – High-High and Low-Low

Process measurement uncertainties are considered in the calculation which take into account other effects such as fluid pressure variations/specific gravity effects and reference leg growth effects. None of the effects considered are significantly affected by the change to uprate power conditions. Containment temperature under normal conditions is not significantly changed as a result of power uprate conditions, and as a result, impulse leg offsets and transmitter temperature effects are not effected by this change in plant power level. SG pressure under the 1.4% power uprate conditions will be approximately 8 psi lower than under current power conditions. The static pressure effect magnitude is computed, however, on the basis of maximum steam generator pressure conditions (at zero power) which is unchanged for the 1.4% power uprate and so the static pressure effect is unchanged. The existing total loop uncertainty value for this function, therefore, is not changed as a result of the 1.4% power uprate.

Loss-of-Offsite Power (480 VAC)

Bus undervoltage relay uncertainties are not subject to influences associated with fluid parametric changes caused by the 1.4% power uprate. ESFAS loads are also unchanged and thus initial bus voltage conditions assumed in the analyses are also unchanged.

Degraded Grid (480 VAC)

Bus undervoltage relay and timer uncertainties are not subject to influences associated with fluid parametric changes caused by the 1.4% power uprate. ESFAS loads are also unchanged and thus initial bus voltage conditions assumed in the analyses are also unchanged.

5 DESIGN TRANSIENTS

5.1 NUCLEAR STEAM SUPPLY SYSTEM DESIGN TRANSIENTS

The IP3 1.4% power uprate results in a slight change in the NSSS design parameters. 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% 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% power uprate.

5.1.1 Design Transient Background

The NSSS design transients are included in the various component design specifications, and are used to perform fatigue analyses. The transients describe variations in the following parameters:

- 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")
- Feedwater temperature (generally reported as "variation" or "change from initial")

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 IP3 were initially generated back in the late 1960s to early 1970s. Unlike the UFSAR Chapter 14 accident analyses, these were developed on a generic basis, using a conservative basis for the definition of the NSSS parameters for a particular analysis. In addition, conservatisms were applied to the particular initial condition for the analysis. For example, the RCS temperature might have been artificially increased, or the steam generator steam temperature might have been artificially decreased. This was done to arrive at conservative bounding design transient parameter variations that could be used on a generic basis as much as possible.

5.1.2 Design Transient Evaluation

The method used for the design transient applicability evaluation was to review the differences in the 100% power condition parameter values between those for the existing plant power level and those for the 1.4% power uprate. Table 5-1 compares these parameter values. The parameters that are reported in the design transients are mainly limited to T_{hot}, T_{cold}, and T_{steam}. The following identifies the small change in the full-power design temperature values due to the 1.4% power uprate (based on 0% SGTP):

T_{hot} – increased by 0.4°F (at reactor vessel outlet)

T_{cold} – decreased by 0.7°F (at steam generator outlet)

T_{steam} – decreased by 1.3°F

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% to 100% power followed by an unloading from 100% to 0% power occurs every day. For the upset transients, it is assumed a reactor trip from 100% 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 Core 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 feedbacks, no credit for any control systems, and no reactor trip till 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% power uncertainty allowance, which bounds the 1.4% power uprate plus the 0.6% power measurement uncertainty.

- The existing design transients have generally been developed using a conservative starting point that results in a greater parameter transient variation than would actually occur based on either the existing design operating conditions or those developed for the 1.4% uprate. This is the case for the RCS T_{hot} and T_{cold}; the transients are actually developed based on values which give greater transient parameter variations than would actually occur if the transients were analyzed based on the 1.4% power uprate parameters.

These conservatisms would typically ensure that the current design transients would continue to remain conservatively applicable to the 1.4% power uprate. However, since the 1.4% uprate full-power steam generator steam pressure and associated steam temperature are lower than those used in the existing design transients, Westinghouse revised certain design transients. These revised transients ensure sufficient conservatism to accommodate the following effects:

- The lower steam pressure due to the 1.4% power uprate would result in a greater steam temperature (i.e., T_{steam}) parameter change when going between no-load and full-power temperatures.
- The steam generator primary-to-secondary pressure differential must stay below the design limit of 1550 psid during any Normal Condition transient and must not be exceeded by more than 110% during any Upset Condition transient.

To address the above issues, the Unit Loading and Unloading, Loss of Flow, and Reactor Trip design transients were revised to reflect the lower, more conservative value for full-power steam pressure noted due to the 1.4% power uprate. The only parameter that changes as a consequence was T_{steam}. T_{hot} and T_{cold} did not require revision.

The lower steam pressure would result in a greater T_{steam} variation during any transient associated with a power level change. While the change is considered small enough (9 psi and 1.3°F as noted in Table 5-1) to not materially affect the design transient response, the design transients that reflect the present full-power steam generator conditions were revised to reflect the lower steam pressure for the 1.4% power uprate conditions.

A review of the primary-to-secondary pressure differential indicated that the limiting Upset transient (Loss of Load) would have resulted in a violation of the primary-to-secondary differential pressure limit. However, this transient was reanalyzed based on improved modeling.

The following peak values for the steam generator primary-to-secondary pressure differential resulted from this design transient effort:

- <u>Normal Condition transients</u>: (Limiting transient is 10% Step Load Increase)
  - Peak Pressure Differential: 1521 psid
  - Design Pressure Differential Limit: 1550 psid
- <u>Upset Condition transients</u>: (Limiting transient is Loss of Load)
  - Peak Pressure Differential: 1610 psid
  - Design Pressure Differential Limit: 1705 psid (110% of 1550 psid Normal Condition design limit)

There are no other changes required to the existing design transients. The frequency of occurrences of all of the design transients are unaffected by the 1.4% power uprate.

5.2 AUXILIARY EQUIPMENT DESIGN TRANSIENTS

The NSSS auxiliary equipment design transients were reviewed based on a comparison between the NSSS design parameters for the 1.4% power uprate described in Section 2 and the NSSS design parameters that form the basis of the current auxiliary equipment design transients.

This review determined that the only current auxiliary equipment transients that could be potentially affected by the 1.4% power uprate are those temperature transients that are affected by the full-load NSSS design temperatures, namely T_{hot} and T_{cold}. These transients are currently based on an assumed full-load NSSS T_{hot} of 630°F and T_{cold} of 560°F. These NSSS temperatures were originally selected to ensure that the resulting design transients would be conservative for a wide range of NSSS design temperatures.

A comparison of the 1.4% power uprate NSSS design temperatures for reactor vessel outlet T_{hot} (600.8°F) and reactor vessel inlet T_{cold} (542.2°F) with the T_{hot} and T_{cold} reference values used to develop the design transients indicates the uprate design temperatures are less than the reference design values. Therefore, the actual temperature transients would be less limiting than the current design temperature transients.

Since the temperature transients based on the 1.4% power uprate conditions are less limiting than those that established the current auxiliary equipment design transients, then all of the current auxiliary equipment design transients for IP3 remain conservatively applicable for the 1.4% power uprate conditions.

TABLE 5-1 IP3 PLANT OPERATING CONDITIONS		
	Original Power Rating	**1.4% Power Uprate**
Power, MWt	3025	3082
Reactor Coolant Flow, gpm/loop	89,700	89,700
Reactor Vessel Outlet Temperature, T_{hot}, °F	600.4	600.8
Reactor Vessel Average Temperature, T_{avg}, °F	571.5	571.5
Steam Generator Outlet* Temperature, T_{cold}, °F	542.6	541.9
Steam Temperature, °F	514	512.7
Steam Pressure, psia	771	762
Feedwater Temperature, °F	427.2	427.8
Feedwater/Steam flow (total), 10^6 lb/hr	13.01	13.26

* Reactor vessel inlet is only 0.4°F higher.

6 NUCLEAR STEAM SUPPLY SYSTEMS

This section discusses the evaluations performed on the NSSS systems in support of the revised design parameters discussed in Section 2. The systems that could potentially be affected by the IP3 1.4% power uprate that are discussed in this section are the NSSS fluid systems, the NSSS/BOP interface systems, and NSSS control systems. The performance and integrity of these systems, except Residual Heat Removal System (RHRS) performance, are unaffected by the 1.4% power uprate. For RHRS performance, the IP3 plant cooldown cases were analyzed based on the 1.4% power uprate and shown to still meet applicable acceptance criteria.

6.1 NSSS FLUID SYSTEMS

6.1.1 Reactor Coolant System

RCS Design Parameters

The NSSS design performance parameters for the 1.4% power uprate are discussed in Section 2. The primary changes in parameters that affect RCS performance are core power and the resulting full-power T_{cold} and T_{hot} temperatures. The steady-state RCS pressure (2235 psig), no-load RCS temperature (547°F), and RCS design flows have not changed. The change in full-power RCS temperatures at 0% SGTP are shown below:

RCS Temperatures	Current Parameters	Uprated Parameters
T_{cold} (SG Outlet)	542.6°F	541.9°F
T_{hot} (Vessel Outlet)	600.4°F	600.8°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. The RCS design pressure and temperatures continue to bound the 1.4% power uprate design performance conditions. Since the RCS design temperature and pressure continue to bound the 1.4% power uprate conditions, the integrity of the RCS pressure boundary is maintained within the original design limits, and it is unaffected by the 1.4% power uprate.

RCS Heat Capacity

The RCS heat capacity is defined as the amount of heat (Btu's) to raise or lower the RCS temperature by 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 fluid(s) and component masses, both of which are unaffected by the 1.4% power uprate. Therefore, the RCS heat capacity is, in turn, unaffected by the 1.4% power uprate.

Reactor Coolant Pump Net Positive Suction Head

This section addresses RCP Net Positive Suction Head (NPSH), as it relates to RCS flow. Adequate RCP NPSH, at the RCP suction nozzle, is monitored 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 hydraulic losses. As indicated by the 1.4% power uprate design parameters (Section 2), RCS flow is unaffected by the 1.4% power uprate. Since there are no plant changes for the 1.4% uprate that could affect the RCS hydraulic performance for RCP NPSH, including RCS flow, the RCP NPSH is unaffected by the 1.4% 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 at the 1.4% power uprate conditions, there is no effect on pressurizer spray performance as a result of these RCS temperature changes. Also, a reactor vessel flow of 358,800 gpm was used for determining pressurizer spray flow performance. The applicable minimum allowable flow for this calculation is 366,800 gpm (see Table 2-1). Since the minimum allowable RCS flow for the 1.4% power uprate condition is greater than the 358,800 gpm flow assumed in the existing spray performance analysis, the pressurizer spray flow performance is unaffected by the 1.4% power uprate.

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. Sufficient flow is ensured to maintain the temperature of the spray lines, valves, spray nozzle, and surge line above 500°F. Low temperature alarm setpoints are specified as 500°F with a tolerance ±5°F. Since the 1.4% power uprate zero-power and full-power RCS hot and cold leg temperatures will be well above the maximum high-side temperature setpoints of these instruments (505°F), these instrument setpoints are unaffected by the 1.4% power uprate.

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% 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 earlier for RCS design parameters and RCS design temperature and pressure, RCS pressure has not changed for the 1.4% power uprate conditions. Also, pressurizer level is not affected by the 1.4% power uprate. Since both RCS pressure and pressurizer level are unaffected by the 1.4% power uprate, the PRT performance is unaffected as a result of the 1.4% power uprate conditions, including the associated PRT setpoints.

6.1.2 Natural Circulation Cooldown Capability

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

6.1.3 NSSS Auxiliary Systems Evaluation

The following NSSS Auxiliary Systems are addressed in this section:

- Chemical and Volume Control System (CVCS)
- Emergency Core Cooling System
- 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 heat exchanger design inlet T_{cold} is 555°F, which bounds the highest RCS T_{cold} temperature associated with the RCS no-load temperature of 547°F. Since the no-load RCS temperature has not changed, and the full-power uprate T_{cold} temperature has decreased by a negligible amount, there is negligible effect on the performance of the regenerative heat exchanger at 1.4% power uprate conditions due to any minor change in letdown flow (due to the slight change in design full-power RCS T_{cold} temperature) or design full-power 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 1.4% power uprate conditions, as discussed in the previous section, there will be a negligible effect on the performance of the non-regenerative heat exchanger. Any minor difference in performance would easily be accommodated by the automatic response of 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 heat exchanger design inlet T_{cold} is 555°F, which bounds the highest T_{cold} associated with the RCS no-load temperature of 547°F (see earlier RCS section). Since the no-load RCS temperature has not changed, and the full-power uprate T_{cold} has decreased by a negligible amount, there is a negligible effect on the performance of the excess letdown heat exchanger at 1.4% power uprate conditions due to the slight 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 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 earlier discussion for the excess letdown heat exchanger). Since the no-load RCS temperature has not changed, and the full-power uprate T_{cold} has decreased by a negligible amount, there is a negligible effect on the performance of the seal water heat exchanger at 1.4% power uprate conditions due to the change in RCS T_{cold} temperature.

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

As discussed earlier for the various CVCS heat exchangers, there is negligible effect on their performance as a result of the 1.4% power uprate conditions. Therefore, there will also be a negligible effect on the charging and letdown performance provided by the CVCS (including RCP seal injection). The flow capacity performance of the RCS makeup system is independent of the change in RCS conditions resulting from the 1.4% power uprate 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.

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 1.4% power uprate conditions as discussed earlier, the 1.4% power uprate does not affect the capability of the makeup system to perform these makeup system functions.

The boric acid storage tanks and refueling water storage tank (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 1.4% power uprate is expected to have a small effect on the boration requirements that must be met by the CVCS boration capabilities. This boration capability will be addressed before implementation of the 1.4% power uprate by the Westinghouse Reload Safety Evaluation (RSE) process, according to the Westinghouse Reload Methodology in WCAP-9272-P-A[7-6]. This process is designed to address boration capability due to routine plant changes, such as core reloads, and infrequent plant changes such as a power uprate that results in a change to core operating conditions. Therefore, boration capability will be addressed as part of an RSE revision to be prepared prior to implementation of this plant uprate for the current cycle core. Future RSEs will also consider the 1.4% power uprate 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 unaffected by the 1.4% power 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 (CSS) performance. Subsequent to ECCS actuation, the Safety Injection System (SIS) draws water from the RWST during the injection phase and delivers to the RCS, while the CSS simultaneously draws from the RWST and sprays the containment atmosphere. During RWST drain down, at the point when SI is terminated at the Low-Low level alarm, the SIS is switched over to the containment recirculation alignment, drawing fluid from the containment sump. At the time when the "Empty" alarm is reached, RWST drain down is concluded and operation of the CSS is terminated. The SIS can also provide recirculation spray to the CSS, if required for continued containment cooling during the recirculation phase.

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

Residual Heat Removal System

The 1.4% uprate affects plant cooldown times since no additional margin (e.g., 102% reactor power) has been applied to the core power level assumed in the current cooldown analysis of record. Therefore, updated cooldown cases were analyzed to account for the 1.4% power uprate conditions. 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 10 CFR 50 Appendix R fire hazards analysis. The following considerations were applied to these cooldown analyses:

- The replacement component cooling water (CCW) heat exchanger data, assuming 5% tube plugging, was used which includes higher heat transfer performance capability and therefore, generally improved CCWS performance (improved normal cold shutdown and Appendix R cooldown performance).

- Accounted for the RHRS 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 IP3 RHRS design.

- The spent fuel pool (SFP) heat load was increased by 1.4% to account for the increased uprated core power. This was done to maintain the same level of margin as exists in the current analysis of record.

- With the improved CCW heat exchanger performance, the letdown heat exchanger heat load is now included in the CCWS auxiliary heat loads for the Appendix R analysis.

The following cooldown analyses results were obtained:

Cases	Time to Cool Down to 140°F (hours after Reactor Shutdown)	Time to Cool Down to 200°F (hours after Reactor Shutdown)	Time RHR Initiated @ 350°F (hours after Reactor Shutdown
1) Normal Cooldown with CCW Auxiliary Heat Loads	94.0	17.0	4.0
2) Normal Cooldown without CCW Auxiliary Heat Loads	83.6	14.0	4.0
3) Appendix R with CCW Auxiliary Heat Loads	N/A	68.9	* 12.0
* RHR is initiated at this time to prevent RCS heatup (above 350°F) after RHR cut-in.			

Cases 1 and 2 are the two-train normal cooldown cases. Case 1 includes the CCWS auxiliary heat loads, while Case 2 does not include auxiliary heat loads. Case 3 is the single train Appendix R cooldown case. The IP3 UFSAR will be updated according to these revised cooldown times. It is noted that the Case 3 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 because these sample locations set the limiting process conditions that govern the design of the 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 earlier, the RCS operating pressure has not changed due to the 1.4% power uprate. Therefore, the design duty of the PSS is unaffected by the 1.4% power uprate.

Component Cooling Water System

Normal Plant Operations (at Power and Refueling)

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

- Charging Pumps (Bearing and Fluid Drive Oil Coolers)
- Seal Water Heat Exchanger (RCP No. 1 Seal Leak Off Return and Excess Letdown)
- Non-Regenerative Heat Exchanger
- Primary Sample Heat Exchanger (Pressurizer Steam, Pressurizer Liquid, RCS)
- Steam Generator Blowdown Sample Heat Exchanger
- Radiation Monitor Condenser Sample Cooler
- Gross Failed Fuel Detector Cooler
- Excess Letdown Heat Exchanger (During Plant Heatup)
- Reactor Vessel Support Cooling Blocks
- RCP Motor Bearing Oil Coolers (Upper and Lower)
- RCP Thermal Barrier Heat Exchanger
- SFP Heat Exchanger
- Waste Gas Compressors (Seal Water Cooling and Seal Water Makeup)

Discussed in the preceding sections were the only CCWS heat loads with a potential to affect the CCWS during normal plant operation and, of those, only the SFP heat load affects the CCWS. (See Section 10.10 regarding the Spent Fuel Pool Cooling System [SFPCS] for further detail.) All other heat loads were determined to be unaffected by the 1.4% power uprate during normal (at power) plant operation.

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

The CCWS provides cooling to the RHRS heat exchangers during plant cooldown. See the earlier RHRS Section 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 earlier RHRS Section, the same CCW supply temperatures of record have been maintained. For normal cooldown, the CCW supply temperature limit is maintained at 120°F. For the Appendix R cooldown, the CCW supply temperature limit is maintained to 125°F. Therefore, the performance of the CCW system is unaffected by the 1.4% power uprate during plant cooldown.

Post-LOCA Plant Cooldown

The CCWS supports post-LOCA ECCS operation during recirculation by providing cooling to the RHR heat exchangers. As described earlier in the ECCS section, there could be a slight increase in sump fluid temperature during recirculation since decay heat increases slightly with power level. However, it has already been determined that the post-LOCA containment sump temperature performance is unaffected by the 1.4% uprate conditions. Therefore, the CCWS is unaffected by ECCS performance at the 1.4% power uprate conditions.

6.2 NSSS/BALANCE-OF-PLANT INTERFACE SYSTEMS

Five BOP fluid systems were reviewed based on the NSSS design parameters presented in Section 2 to assess compliance with the NSSS/BOP interface guidelines. The BOP systems included in this interface evaluation are:

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

The NSSS design performance parameters for the 1.4% power uprate were compared with the current design parameters for systems and components. The comparison indicated differences that could affect the performance of the above BOP systems. For example, the increase in NSSS power due to a conservative RCP net heat input of 14 MWt, along with the 1.4% core power increase, would result in about a 1.9% increase in a steam/feedwater mass flow rates and a reduction in full-load steam pressure of 1.2%.

Evaluations of the above BOP systems relative to compliance with Westinghouse NSSS/BOP interface guidelines were performed to address the NSSS design parameters for power uprate analyses that maintain T_{avg} (571.5°F), assumed SGTP level (0%), and decreased assumed moisture carryover (MCO) (0.25% to 0.1%). As mentioned, these NSSS design parameters coupled with an NSSS power increase due to a 1.4% core power increase, with an RCP net heat input of 14 MWt, result in a decrease in steam generator outlet pressure and an increase in steam/feedwater mass flow rate. The results of the NSSS/BOP interface evaluations are delineated below.

6.2.1 Main Steam System

The 1.4% power uprate, coupled with the potential reduction in full-load steam pressure to the design value of 762 psia, adversely affects main steam line pressure drop. At the design steam generator dome pressure of 762 psia, the full-load steam mass flowrate would increase by approximately 1.9%; however, due to the reduced operating pressure and the lower-density steam, the volumetric flowrate would increase by approximately 3.2% and steam line pressure drop would increase by approximately 5.2%. Note that main steam line pressure drop affects plant economics, since an increase in pressure drop results in a corresponding increase in plant heat rate over the life of the plant. (An increase in steam line pressure drop of 1 psi is equivalent to an increase of approximately 2 Btu/KW-hr in plant heat rate.) Initial plant design studies indicated that a pressure drop in the range of 25 to 40 psi at rated load provided an acceptable economic balance between the value of a lower heat rate over the life of the plant and the capital cost of larger-bore, longer-length pipes.

The current NSSS design performance parameters for the current core power of 3025 MWt resulted in a steam line pressure drop of about 39.3 psi and a pressure of 732 psia at the turbine inlet valves. Based on the NSSS design performance parameters associated with the 1.4% power uprate to a core power of 3068 MWt (Table 2-1), the reduced steam generator dome pressure of 762 psia would result in a steam line pressure drop of 41.3 psi and a pressure at the turbine inlet valves of approximately 721 psia.

The following subsections summarize the evaluation of the major steam system components relative to the 1.4% power uprate NSSS design performance parameters. The major components of the MSS include the steam generator MSSVs, steam generator power-operated atmospheric relief valves (ARVs), main steam isolation valves (MSIVs), and non-return/check 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 ASME Boiler and Pressure Vessel (B&PV) Code. Since the design pressure of the steam generators has not changed with the 1.4% 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% 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 (IP3 UFSAR). Based on this requirement, Westinghouse applies the conservative criterion that the valves should be sized to relieve 100% of the maximum calculated steam flow at an accumulation pressure not exceeding 110% of the MSS design pressure. Additionally, the capacity of any single safety valve is presently limited to 890,000 lb/hr at 1100 psia based on the steam line break analysis of record for a stuck-open steam generator safety valve.

IP3 has twenty safety valves with a total capacity of 15.108 x 10^6 lb/hr, which provides about 113.9% of the maximum uprated full-load steam flow of the 13.26 x 10^6 lb/hr. 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 Atmospheric Relief Valves

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 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 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. Under such circumstances, the ARVs in conjunction with the 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 integral derivative (PID) controller compares steam line pressure to the pressure setpoint, which is manually set by the plant operator.

In the event of a tube rupture event in conjunction with loss of offsite power, 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. 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 about 10% of rated steam flow at no-load pressure. This capacity permits a plant cool down to RHRS operating conditions (350°F) in four hours (at a rate of about 50°F/hr) assuming cooldown starts two 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 basis in terms of cool down capability can still be achieved for 1.4% power uprate NSSS design performance parameters.

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 (IP3 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 flowrate 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% 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% 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% power uprate has no significant effect on the interface requirements for the MSIV bypass valves.

6.2.2 Steam Dump System

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

- Loading at 5% of full power per minute with automatic reactor control.
- Unloading at 5% of full power per minute with automatic reactor control.
- Instantaneous load transients of plus or minus 10% of full power (not exceeding full power) with automatic reactor control.
- Load reductions of 50% 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% of the rated steam flow at full-load steam pressure to permit the NSSS to withstand an external load reduction of up to 50% of plant rated electrical load without a reactor trip. To prevent a trip, this transient requires all NSSS control systems to be in automatic, including the Reactor Control System, which accommodates 10% of the load reduction. A steam dump capacity of 40% of rated steam flow at full-load steam pressure also prevents MSSV lifting following a reactor trip from full power.

Steam Dump System Major Components

IP3 is equipped with twelve 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. The total capacity provides a steam dump capability of about 45.3% of the current maximum guaranteed steam flow (13.01×10^6 lb/hr), or 5.894×10^6 lb/hr at a design full-load steam generator dome pressure of 771 psia versus the Westinghouse sizing criterion of 40% of rated steam flow.

The NSSS design parameters for the 1.4% power uprate will result in reduced steam dump capability, since design full-load steam generator dome pressure is reduced (from 771 psia to 762 psia) and full-load steam flow increases (from 13.01 x 10^6 lb/hr to 13.26 x 10^6 lb/hr). An evaluation indicated that the steam dump capacity would only be reduced to 43.8% of rated steam flow (13.26 x 10^6lb/hr), or 5.808x10^6 lb/hr at a full-load steam pressure equal to 762 psia. Therefore, condenser steam dump valves have adequate margin relative to the Westinghouse sizing criterion (40% of rated steam flow) to accommodate the NSSS design parameters for the 1.4% power uprate.

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 in order to control flow. Positioning response may be slower with a maximum full stroke time of 20 seconds. These requirements are still applicable for the NSSS design parameters for the 1.4% power uprate.

The NSSS controls systems analysis provides an evaluation of the adequacy of the steam dump control system at the 1.4% power uprate NSSS design parameters.

6.2.3 Condensate and Feedwater System

The C&FS must automatically maintain steam generator water levels during steady-state and transient operations. The 1.4% power uprate NSSS design parameters will result in a required feedwater volumetric flow increase of up to 2.0% during full-power operation. As noted earlier, the higher feedwater flow will have an effect on system pressure drop, which may increase by as much as 3.9%. Also, a comparison of the 1.4% power uprate NSSS design parameters with the current NSSS design parameters (based on 0% SGTP) indicates that the steam generator full-power operating steam pressure will decrease by 9 psi (from 771 psia to 762 psia).

The major components of the C&FS are the main feedwater control valves (FCVs), and the C&FS pumps.

Main Feedwater Isolation/Feedwater Control Valves

The main FCVs are located outside containment. The valves function in conjunction with the backup trip signals to the feedwater pumps and the feedwater pump discharge isolation 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 (IP3 UFSAR).

The quick-closure requirements imposed on the FCVs and backup feedwater pump discharge isolation valves cause 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 zero-power conditions with the conservative assumption that all feedwater pumps are in service providing maximum flow following the break. Since these conservative assumptions are not affected by the 1.4%

power uprate, the design loads and associated stresses resulting from rapid closure of these valves are unaffected.

Condensate and Feedwater System Pumps

The C&FS available head in conjunction with the FCV characteristics must provide sufficient margin for feed control to ensure adequate flow to the steam generators during steady-state and transient operation. A continuous, steady feed flow should be maintained at all loads. To assure stable feedwater control, with variable speed feedwater pumps, the pressure drop across the FCVs at rated flow (100% power) should be approximately equal to the dynamic losses from the feed pump discharge through the steam generator (i.e., equal to the frictional resistance of feed piping, high pressure feed water heaters, feed 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% of rated flow with a 100 psi pressure increase above the full-load pressure with the FCVs fully open. However, based on the IP3 FCV design (full open C_v of 790 at 2.5 inches lift) and the system layout, the present pump speed control program was set to provide a FCV pressure drop of about 135 psi to achieve about a 73% valve lift at full load.

For the 1.4% power uprate NSSS design parameters, the present speed control program results in a small change in FCV pressure drop (approximately 3.5 psi) and a corresponding small change in valve lift (approximately 2%) at 100% power. Therefore, operation of the FCVs (in conjunction with the present feedwater pump speed control program) has been determined to be acceptable for both steady-state and transient operation consistent with this evaluation.

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 uprated conditions.

6.2.4 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 Safeguards System. In the latter function, the AFWS 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 1.4% power uprate does affect safety analyses performed at a nominal 100% power level, evaluations were performed to confirm that the AFWS performance is acceptable at the 1.4% 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 engineered safety features (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 loss-of-offsite power transient. The IP3 licensing basis dictates that in the event of a loss of offsite power, sufficient CST useable 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 IP3 analysis of record concluded that the CST should be designed to accommodate a minimum useable inventory of 290,000 gallons. Accordingly, the IP3 Technical Specifications ensure a contained volume of 360,000 gallons or greater.

The minimum required useable inventory of 290,000 gallons is based on reactor trip from 102% of a maximum power of 3220 MWt (or 3284.4 MWt), which bounds the original IP3 design basis power level of 3216 MWt. Since the 1.4% power uprate is based on improved calorimetric accuracy, no change in the analysis or the IP3 Technical Specifications is required with regard to auxiliary feedwater storage requirements at the 1.4% uprate power level.

6.2.5 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 secondary sides of the steam generators.

The blowdown flowrates required during plant operation are based on chemistry control and tubesheet sweep requirements to control the buildup of solids. The blowdown flowrate 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, allowable primary-to-secondary leakage, and the performance of the condensate polishers. Since these variables are not affected by 1.4% power uprate, the blowdown required to control secondary chemistry and steam generator solids will not be unaffected by the 1.4% 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 original NSSS design parameters permit a maximum decrease in steam pressure from no-load to full-load of 265 psi (i.e. from 1020 psia to 755 psia). Based on the revised range of 1.4% power uprate NSSS design parameters, the no-load steam pressure (1020 psia) remains the same and the minimum full-load steam pressure (762 psia) is 7 psi higher than the original 755 psi full-load design pressure for this system. This small decrease in blowdown system inlet pressure will not significantly affect the required maximum lift of the blowdown flow control valves. Therefore, the range of design parameters approved for 1.4% power uprate will not affect blowdown flow capability.

6.3 PLANT OPERABILITY

The incorporation of the 1.4% power uprate can affect plant operability in two ways, as follows:

- **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 operability**: this includes the plant response to the normal design basis plant operability transients, including 5% per minute loading and unloading, 10% step load increase or decrease, large load rejection, and turbine trip without reactor trip below the P-8 permissive. These transients must be handled without resulting in a reactor trip or ESFAS actuation and not challenging the pressurizer or steam generator safety valves.

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 original 3025 MWt power level and for the 1.4% power uprate is shown in Table 6-1.

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

Table 6-1 IP3 Plant Operating Conditions (at 0% SGTP)		
	Original Power Rating	**1.4% Uprate**
Power, MWt	3025	3082
Flow, GPM/loop	89,700	89,700
T_{hot}, °F	600.4	600.8
T_{avg}, °F	571.5	571.5
T_{cold}, °F	542.6	541.9*
T_{steam}, °F	514	512.7
P_{steam}, psia	771	762
$T_{feedwater}$, °F	427.2	427.8
FW/Steam flow (total), 10^6 lb/hr	13.01	13.26
* Value is S/G outlet; reactor vessel inlet is 0.4°F higher.		

6.3.1 Pressure Control Component Sizing

The various pressure 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% power uprate conditions. The results obtained from the IP3 Replacement Steam Generator Program were used as a basis for the evaluation.

The following pressure control components will be evaluated separately:

- Pressurizer heaters
- Pressurizer spray
- Pressurizer PORVs

Pressurizer Heaters

The pressurizer heaters are sized to be able to heat up the pressurizer liquid at a 200°F/hr 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 which result in pressurizer outsurge events. These include the initial part of a 10% step load increase transient, a 5% per minute plant unloading transient, or events resulting in a reactor trip. Generic analyses 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%, 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% power uprate. The heatup maneuver would be essentially the same as that which IP3 presently experiences. Therefore, the installed pressurizer heater capacity is acceptable for the 1.4% power uprate.

Pressurizer Sprays

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

Because of the age of the IP3 plant and the current IP3 spray sizing analysis of record, Westinghouse reanalyzed the 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:

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

- Plant initially at nominal T_{avg} + 4°F uncertainty,
- Transient is a step load reduction from the noted 100.6% turbine load to 90% load, and
- Fuel reactivities at conservative BOL conditions.

The analysis resulted in a peak pressurizer pressure that was below the pressurizer PORV setpoint of 2350 psia (2335 psig). Therefore, the installed pressurizer spray capacity is adequate to maintain the pressurizer pressure below the pressurizer PORV actuation setpoint for a 10% step load decrease transient for the 1.4% power uprate conditions.

Pressurizer Power Operated Relief Valves

The sizing basis for the pressurizer PORVs is to prevent the pressurizer pressure from reaching the high pressurizer pressure reactor trip setpoint for the design basis load rejection with steam dump transient. For IP3, the limiting transient is a reduction in the turbine load from 100% to 50% power.

Because of the age of the IP3 plant and the current IP3 PORV sizing analysis of record, Westinghouse reanalyzed the PORV sizing based on current-day analysis methodologies to ensure acceptability. This included the following assumptions:

- Plant initially at 100.6% of the 1.4% uprate power level. The standard analysis methodology is to assume a power uncertainty allowance of 2%. The 1.4% power uprate plus the 0.6% instrument uncertainty is bounded by this original 2% power measurement uncertainty allowance.
- Plant initially at nominal T_{avg} + 4°F uncertainty,
- Steam generator heat transfer coefficient increased to the maximum credible value (0% fouling, 0% SGTP),
- Transient is a step load reduction from the noted 100.6% turbine load to 50% load,
- Fuel reactivities at conservative BOL conditions, and
- Credit taken for automatic operation of all NSSS control systems (reactor control, pressurizer pressure and level control, feedwater control, and steam dump control).

The analysis showed that the installed pressurizer PORV capacity is adequate to maintain the pressurizer pressure below the high pressurizer pressure trip setpoint for a 50% large load rejection transient for the 1.4% power uprate conditions.

6.3.2 Plant Operability

The design basis operability transients for IP3 include the following:

- 5% per minute unit loading from 15% to 100% power,
- 5% per minute unit unloading from 100% to 15% power,
- 10% step load increase; limiting case is from 90% to 100% power,
- 10% step load decrease; limiting case is from 100% to 90% power,
- Large step load decrease with steam dump; limiting case is from 100% to 50% power, and
- Turbine trip without reactor trip below the P-8 power permissive.

All of these transients should be accommodated without generating a reactor trip or ESFAS actuation.

Out of the above, there is a limiting set of transients that are chosen for analysis. Successful results obtained from these analyses provide assurance that the remaining transients can be handled successfully. The transients chosen for detailed analysis were the following:

- 10% step load decrease; limiting case is from 100% to 90% power. This is the same transient described for the sizing of the pressurizer spray. The same acceptance criteria are applied here. This transient should not result in actuation of the pressurizer PORVs (actuated at 2,335 psig).

- Large step load decrease with steam dump; limiting case is from 100% to 50% 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 2385 psig). In addition, this is the limiting operational transient for actuation of the overtemperature delta temperature (OTΔT) or overpower delta temperature (OPΔT) trip setpoints; the plant transient response acceptability should be that the trip setpoint is not reached.

- Turbine trip without reactor trip below the P-8 power permissive (no greater than 50% of rated thermal power) transient, the criteria is that the pressurizer PORVs will not be challenged even assuming any credible control system single failure.

10% Step Load Decrease

The analyses done for the pressurizer spray capacity verification are also directly applicable for this 10% step load decrease operability transient analysis. The results showed a maximum pressurizer pressure of 2302 psia. This is below the pressurizer PORV setpoint of 2335 psig (2350 psia). Therefore, there is acceptable margin to the pressurizer PORV actuation setpoint for a 10% step load decrease transient actuated from the 1.4% power uprate conditions.

Large Step Load Decrease with Steam Dump

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

- Plant initially at 100% of the 1.4% power uprate level,
- Plant initially at nominal T_{avg} uncertainty,
- Nominal steam generator heat transfer coefficient (0% SGTP condition),
- Transient is a step load reduction from the noted 100% turbine load to 50% load,
- Fuel reactivities at conservative BOL conditions, and
- Credit taken for automatic operation of all NSSS control systems (reactor control, pressurizer pressure and level control, feedwater control, and steam dump control).

An initial analysis for the 1.4% power uprate resulted in the OTΔT setpoint being challenged. The cause was the slow response of the steam dump control system with the present setpoints. The following change to the trip-open setpoint on the first bank of steam dump valves achieved successful results.

High (T_{avg} - T_{ref}) valve trip open (TC-412K) 11°F (from present value of 12°F)

The results show acceptable margin to all reactor trip and ESFAS setpoints. The minimum margin to the limiting setpoint (OTΔT) is about 3.5%. The peak pressurizer pressure was no greater than the pressurizer PORV actuation setpoint. Therefore the 50% load rejection can be accommodated for the 1.4% power uprate without challenging any of the reactor trip setpoints.

Turbine Trip without Reactor Trip below the P-8 Power Permissive

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 turbine trip without reactor trip below the P-8 permissive effort. The results of that effort concluded that the PORVs would not be challenged for any credible control system single failure. That analysis was done assuming a 2% power uncertainty allowance above the initial power level of 50% RTP. Therefore, for the 1.4% power uprate, this current analysis remains valid as long as the P-8 setpoint is no greater than 50.4% of the 1.4% power uprate level (52% of current analysis power level).

Based on these limiting analyses, all of the normal plant operability transients can be accommodated for the 1.4% power uprate conditions without challenging any reactor trip or ESFAS setpoints.

Other Considerations

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

NSSS Control Systems Setpoints

The 1.4% power uprate will not affect any of the control system setpoints except for the steam dump control system. For the steam dump control system, the trip-open setpoint for steam dump bank 2 will be modified as follows:

High (T_{avg} - T_{ref}) valve trip open (TC-412K) 11°F

The 1.4% power uprate could cause the FCVs to operate at a few percent (judged to be less than 5%) higher open position than they presently do. The issue is that there could be some small oscillations in the FCV position at full-power operation. This is more likely if the feedwater/steam flow increases from the present value. If this happens, the header-to-header ΔP program may be increased to result in the feedwater control valves operating at about the same positions they were at before the 1.4% power uprate. The potential need to adjust the header-to-header ΔP program cannot be definitively foreseen, but FCV stroke positions and Cv utilizations at the current power level suggest that the oscillation phenomena will probably not occur as a result of the 1.4% power uprate. However, this is an area that will be monitored during the implementation phase of the 1.4% power uprate, and contingency actions to mitigate significant oscillations will be available and taken as needed. The above cited possibility of eliminating oscillations is identified based on Westinghouse experience at some plants where slow valve oscillations were observed when they were too far open to result in smooth and stable feedwater control.

NSSS Instrument Spans

The limiting transmitter affected by the 1.4% power uprate is considered to be the steamline flow transmitter in each loop. The steam flow transmitter is used for two functions:

- **Protection-related functions**: ESFAS actuation on high steam line flow in conjunction with low steam line pressure

- **Control-related functions**: Steam flow input to steam generator level control system.

The pressure drop across the transmitter varies as the square of the flow and is affected by the steam density. A comparison based on the information shown in Table 6-1 was made as follows:

Current design:	13,010,000 lb/hr total at 771 psia
1.4% power uprate:	13,260,000 lb/hr total at 762 psia

The steam flow transmitter can basically be characterized as measuring flow as:

$$W = K * \sqrt{\Delta P * \rho}$$

where W = flow in lb/hr, ΔP is pressure drop, psid, ρ is steam density in lb/ft^3 and K is the transmitter flow coefficient and units conversion.

Now, taking the ratio of conditions for the 1.4% power uprate and current plant design, the transmitter pressure drop ratios can be expressed as:

$$\Delta P2 / \Delta P1 = (W2 / W1)^2 * \rho 1 / \rho 2$$

Using the conditions for the 1.4% power uprate and current power rating, the pressure drop across the steam line flow transmitter at the new 100% power operating conditions would increase about 5.2% due to the 1.4% power uprate.

With the requirement for being able to measure a maximum of 107.3% of full-load steam flow at a low steamline pressure setpoint of 616 psig for ESFAS actuation/steamline isolation, the pressure drop across the transmitter at this condition is 1.421 times the nominal 100% power pressure drop. Now, for the 1.4% power uprate condition, the steam flow is 1.019 times the value for the current design condition. With no change in the pressure drop associated with the ESFAS actuation/steamline isolation setpoint, the pressure drop across the transmitter at this condition is 1.351 times the pressure drop associated with the 1.4% power uprate full-power condition.

In order to actuate the high steamline flow ESFAS/steamline isolation function, a steamline flow of 105.3% of the 1.4% power uprate condition would have to occur at a steamline pressure of no less than 616 psig (vs. the present case of actuating at 107.3% of the nominal steamline flow). During normal operational transients this condition would not occur unless there was some system or equipment fault. However, the margin to actuation has been reduced about 7% (pressure drop ratio at actuation vs. nominal 100% power condition of 1.352 vs. 1.421) due to the higher steam flow. Considering the potential adverse plant effects of an ESFAS/steamline isolation, the steamline ESFAS setpoints will be rescaled to reflect the 1.4% power uprate steam flow.

Relative to the steam flow transmitter's input to the 3-element Steam Generator Level Control System, rescaling is not necessary, since the transmitters are currently spanned to 4.00 million lbm/hr. This is approximately 120% higher than the 1.4% power uprate design steam flow per steam generator of 3.315 million lbm/hr. The feedwater transmitters are spanned for this same mass flow rate, thereby providing a completely matched condition of flow inputs to this control system.

6.3.3 Conclusions

Based on this review, the IP3 1.4% power uprate is not expected to result in unacceptable plant operations. The existing pressurizer pressure control component sizing is acceptable for the 1.4% power uprate conditions. The plant operability transients (5% per minute loading/unloading, 10% step, large load rejection, and turbine trip without reactor trip below P-8 permissive) can be accommodated and meet the existing design basis requirements. Revisions are not needed for any control system setpoints except for the trip-open setpoint of the first bank of steam dump valves. A setpoint re-scaling will be done, however, for the Steamline Flow ESFAS setpoint to recapture desired operating margin, while still maintaining conservatism relative to the calculated limiting setpoint.

6.4 SECTION 6 REFERENCE

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

7 NSSS COMPONENTS

7.1 REACTOR VESSEL STRUCTURAL EVALUATION

The IP3 reactor vessel structural integrity has been evaluated for potential effects due to the 1.4% power uprate. The evaluation addressed the potential effects of the uprate design performance parameters (Table 2-1) on the most limiting locations with regard to ranges of stress intensity and fatigue usage factors in each of the regions as identified in the original reactor vessel stress report. The reactor vessel design transients were not affected as result of the 1.4% power uprate. However, the normal steady-state vessel inlet temperature decreased from 554.8°F (design temperature used in the original reactor vessel stress analysis) to 542.2°F. This would affect the T_{cold} variation during normal Plant Loading and Plant Unloading. Therefore, normal Plant Loading and Plant Unloading were evaluated for those regions of the reactor vessel that are in contact with vessel inlet water during normal operation in order to assure that the reactor vessel stress report results remain conservatively applicable. The evaluation concluded that no changes to the stress report results were necessary since the temperature variation from the zero-load temperature of 547°F during Plant Loading and Unloading is greater with the current vessel inlet temperature of 554.8°F. The normal vessel outlet temperature decreases from 613.0°F (design temperature used in the original reactor vessel stress analysis) to 600.8°F for the 1.4% power uprate. Since the effects of Plant Loading and Unloading with the 1.4% power uprate are bounded by the current analysis of record (having the larger temperature variation), reactor vessel operation based on the 1.4% power uprate for the remainder of the current operating license is justified.

The current design basis LOCA hydraulic forces on the reactor vessel continue to remain bounding for the IP3 1.4% power uprate. Faulted-condition LOCA plus seismic loads, which bound the effects of the current LOCA forces alone, were previously evaluated and justified for application to the reactor vessel interface locations. However, those evaluation results had not been included in the IP3 reactor vessel stress report. Though not related to the 1.4% power uprate, the interface loads and faulted-condition stress intensity results were summarized for incorporation into an IP3 reactor vessel stress report addendum. No additional faulted-condition evaluations were performed.

The vessel inlet temperature for the 1.4% power uprate results in a reversal in the original temperature variations during normal Plant Loading and Plant Unloading for those regions of the reactor vessel assumed to be in contact with vessel inlet water during normal reactor operation. The temperature variation for Plant Loading changes from a temperature increase to a temperature decrease, and the temperature variation for Plant Unloading changes from a temperature decrease to a temperature increase. However, the original temperature variation during Plant Loading and Unloading is greater in magnitude than corresponding temperature variation with the 1.4% power uprate parameters. Therefore, the current reactor vessel analysis of record remains bounding for regions in contact with vessel inlet temperature during normal operation, but the stress intensities for Plant Loading become the stress intensities for Plant Unloading and vice versa. The remaining regions (including the control rod drive mechanism (CRDM) housings, main closure flanges and outlet nozzles) are assumed to be in contact with reactor coolant at or near the vessel outlet temperature during normal operation.

The reactor vessel main closure flange assembly was evaluated for the potential effects of the increased T_{hot} variations during the normal Plant Loading and Plant Unloading transients. The evaluation was also based upon the results of a previous evaluation for the main closure that was not included in the original

IP3 reactor vessel stress report. This evaluation correctly considers that the temperature in the upper head region is near the vessel outlet temperature whereas the original stress report assumed that the closure head internal surfaces are in contact with vessel inlet water. In addition, Operating Basis Earthquake (OBE) seismic loads are included in the evaluation that were not considered in the original stress report. (This was documented in WCAP-15859, "Addendum to Analytical Report for the Indian Point Vessel Unit No. 3 - Mini-Uprate Evaluation.) The evaluation results show that the maximum ranges of primary-plus-secondary stress intensity, and maximum cumulative fatigue usage (CFU) factors reported for the main closure in the original IP3 reactor vessel stress report change as a result of the analysis for higher upper head temperatures using finite element methods and the addition of the OBE loads. The maximum range of stress intensity for the closure head flange decreases from 50.5 ksi to 45.37 ksi. The maximum range of stress intensity for the vessel flange increases from 45.4 ksi to 52.14 ksi. The maximum ranges for these flanges compare favorably with the $3S_m$ limit of 80.1 ksi. The maximum bolt service stress for the closure studs increases from 95.9 ksi to 109.4 ksi. This result compares to a limit of 110.4 ksi. The maximum CFU factors for the head flange and vessel flange remain relatively small at 0.0107 and 0.0229, respectively. However, the maximum CFU factor for the closure studs increases from 0.313 to 0.9078 as a result of the higher upper head temperature. This fatigue calculation was based on the original 40-year design life of the component.

The CRDM housing maximum range of stress intensity and maximum CFU factor were found to be unchanged as a result of the upper head temperature increase and the 1.4% power uprate. This result is based upon the conclusion that the original stress report results remain conservative for the higher head temperatures since differential thermal expansion between the CRDM housing tubes and the head compensates for the high pressure stresses at the attachment weld.

The maximum range stress intensity for the outlet nozzle increases from 45.46 ksi to 49.39 ksi due to the addition of the OBE seismic interface load at the nozzle internal protrusion. This result is still much less than the $3S_m$ limit of 80.1 ksi. The maximum CFU factor for the outlet nozzles increases from 0.022 to 0.259 due to consideration of the OBE seismic load.

The faulted-condition primary stress intensities for the IP3 reactor vessel do not change as a result of the 1.4% power uprate because there are no changes in the faulted-condition reactor vessel/reactor internals interface loads or other faulted-condition loads as a result of the 1.4% power uprate. However, the previously faulted-condition stress intensity results were summarized and compared to the applicable limits of Appendix F in the 1974 Edition of Section III of the ASME B&PV Code for incorporation into an IP3 reactor vessel stress report addendum.

There are no changes to the reactor vessel stress report, related to the 1.4% power uprate, as a result of flow-induced vibration, changes in flow rates, high-energy line break locations, or jet impingement and thrust forces.

The structural integrity of the IP3 reactor vessel continues to satisfy the applicable requirements of the 1965 Edition of Section III of the ASME B&PV Code, with addenda through the Winter 1965 Addendum, in accordance with the reactor vessel design requirements. Faulted-conditions satisfy the applicable requirements of the 1974 Edition of Section III of the ASME B&PV Code. Therefore, the structural integrity of the IP3 reactor vessel is unaffected by the 1.4% power uprate.

7.2 REACTOR VESSEL INTEGRITY – NEUTRON IRRADIATION

Reactor vessel integrity would be affected by any changes in plant design parameters that affect neutron fluence levels or temperature/pressure transients. Changes in neutron fluence resulting from the proposed IP3 1.4% power uprate have been assessed to determine the potential effect on reactor vessel integrity. This assessment included a review of the current material surveillance capsule withdrawal schedule, applicability of the plant heatup and cooldown pressure-temperature limit curves, applicability of the Emergency Response Guideline (ERG)[7-1] limits, the effect on the RT_{PTS} values (10 CFR 50.61, known as the Pressurized Thermal Shock (PTS) Rule), and a review of the updated inlet temperature.

The most critical area, in terms of reactor vessel integrity, is the beltline region of the reactor vessel. The beltline region is defined in ASTM E185-82[7-2] as "the irradiated region of the reactor vessel (shell material including weld regions and plates or forgings) that directly surrounds the effective height of the active core and adjacent regions that are predicted to experience sufficient neutron damage to warrant consideration in the selection of surveillance material."

7.2.1 Description of Analyses/Evaluations Performed

The primary objectives of the reactor vessel integrity evaluation for the IP3 1.4% power uprate were to:

1. Review the reactor vessel surveillance capsule removal schedule for IP3 to determine if changes are required as a result of changes in vessel fluence due to the 1.4% power uprate. This evaluation is consistent with the recommended practices of ASTM E185-82[7-2] and meets the requirements of Appendix H of 10 CFR Part 50.

2. Review existing Pressure-Temperature (P-T) limit curves to determine if a new applicability date needs to be calculated due to the effects of the 1.4% power uprate fluence projections. The methodology of NRC Regulatory Guide 1.99, Revision 2 (May 1988), will be used in any required calculations.

3. Calculate new RT_{PTS} values to determine if the effects of the 1.4% power uprate fluence projections cause an increase in RT_{PTS} for the beltline materials in the IP3 reactor vessel at end-of-life (EOL) (27.1 effective full-power years [EFPY]). The current PTS Rule, 10 CFR Part 50.61, will be used to ensure that the screening criteria are met.

4. Review the Upper Shelf Energy (USE) values at EOL for all reactor vessel beltline materials in the IP3 reactor vessel to assess the potential effect from the 1.4% power uprate fluence projections.

5. Review the inlet temperature for IP3 to verify that it maintains an acceptable level with the 1.4% power uprate.

Uprated Fluence Projections

Calculated fluence projections on the reactor vessel were evaluated for the 1.4% power uprate for input to the reactor vessel integrity evaluations. Typically, fluence values are used to evaluate the EOL transition temperature shift (EOL ΔRT_{NDT}) for development of the surveillance capsule withdrawal schedules, adjusted reference temperature (ART) values for determining the applicability of the heatup and cooldown curves, ERG limits, and RT_{PTS} values. The calculated fluences used in this 1.4% power uprate evaluation comply with NRC Regulatory Guide 1.190.

7.2.2 Acceptance Criteria for Analyses/Evaluations

Surveillance Capsule Withdrawal Schedule

The proposed surveillance capsule removal schedule developed for IP3 following the 1.4% power uprate shall meet the requirements of ASTM-185-82[7-2]. A satisfactory number of surveillance capsules shall remain in the reactor vessel so that further analysis, such as may be needed for life extension, can be completed as necessary.

Applicability of Heatup and Cooldown Pressure-Temperature Limit Curves

The applicability date for the heatup and cooldown curves presently identified by IP3 Tech Spec 3.4.3 shall be known.

ERG Limits

The ERG limits shall be known in order to establish guidelines for operator action in the event of an emergency situation, such as a PTS event.

Pressurized Thermal Shock

The 1.4% power uprate RT_{PTS} values for all beltline materials shall not exceed the screening criteria of the 10 CFR 50.61 PTS Rule. Specifically, the RT_{PTS} values of the base metal (plates or forgings) shall not exceed 270°F, while the girth weld metal RT_{PTS} values shall not exceed 300°F through the EOL (27.1 EFPY).

Upper Shelf Energy

At the 1.4% power uprate conditions, the EOL USE values for all reactor beltline materials meet the requirements of 10 CFR 50, Appendix G.

Inlet Temperature

The reactor vessel inlet temperature must be maintained in the range of 525°F to 590°F for current analyses described herein to remain valid.

7.2.3 Results

The following results are based on increased neutron fluence projections for IP3 for the 1.4% power uprate.

Surveillance Capsule Withdrawal Schedule

The current surveillance capsule withdrawal schedule for IP3 is documented in WCAP-11057. As part of the 1.4% power uprate, the capsule fluence values only changed slightly from those documented in WCAP-11057. Therefore, a new surveillance capsule withdrawal schedule was calculated based on ASTM E185-82[7-2]. Per ASTM E185-82, the withdrawal of a capsule is to be scheduled at the nearest vessel refueling outage to the calculated EFPY established for the particular surveillance capsule withdrawal.

The capsules removed from the IP3 vessel to date meet the intent of ASTM E185-82. Since the capsule fluences for the 1.4% power uprate have changed from the capsule fluences used in development of the current withdrawal schedule, a new withdrawal schedule was developed and documented in Table 7-1. Table 7-2 shows that the maximum ΔRT_{NDT} using the 1.4% power uprate fluences for IP3 at 27.1 EFPY is 217.1°F. Per ASTM E185-82, this ΔRT_{NDT} value would require five capsules to be withdrawn from IP3.

Applicability of Heatup and Cooldown Pressure-Temperature Limit Curves

IP3 Tech Spec 3.4.3 shows the current applicability date of 16.2 EFPY. Because of the 1.4% power uprate fluences, the applicability date has been decreased slightly to 16.17 EFPY as shown in Table 7-3. As shown, this change equates to less then one month of operating time and is based on current Accumulated Lifetime Burnup, the time remaining on the current P-T curves and the 1.4% power uprate.

ERG Limits

For IP3, the peak surface RT_{PTS} value at EOL was calculated to be 263°F based on the 1.4% power uprate fluences. This is actually a 1°F improvement over the current IP3 calculation value of 264°F. The limiting material for IP3 was the Lower Shell Plate B2803. The 1.4% power uprate is compared to EOL RT_{PTS} ERG limits in Table 7-4. As is the case for the current EOL RT_{PTS} limits, Plate B2803 is longitudinally-oriented and above 250°F, therefore, IP3 will still maintain a plant-specific ERG analysis for EOL.

Pressurized Thermal Shock

Due to the 1.4% power uprate, new PTS evaluations were completed using the latest procedures specified by the NRC in the PTS Rule. These new calculations for EOL are shown in Table 7-5. Using the 1.4% power uprate fluences, the most limiting beltline material in the IP3 reactor vessel remains below the screening criteria values of 270°F at EOL (27.1 EFPY).

Upper Shelf Energy

Based on the 1.4% power uprate, all beltline materials are expected to have an upper shelf energy (USE) greater than 50 ft-lb through end of license (EOL, 27.1 EFPY) as required by 10 CFR 50, Appendix G. The EOL (27.1 EFPY) USE was predicted using the EOL 1/4T fluence projection and the predictions are shown in Table 7-6.

Table 7-6 shows that all beltline materials will be maintained above 50 ft-lb through EOL.

Inlet Temperature

The inlet temperature must be maintained within the range of 525°F to 590°F to uphold the basis of the equations and tables from NRC Regulatory Guide 1.99, Revision 2 and 10 CFR 50.61. As shown in Table 2-1, the inlet temperature is maintained within this range.

7.2.4 Conclusions

The fluence projections for the 1.4% power uprate were considered in the calculations of the withdrawal schedule, pressure-temperature limit curves, emergency response guideline category, pressurized thermal shock and upper shelf energy. Based on these calculations, the IP3 1.4% power uprate will not have a significant effect on the reactor vessel integrity up to EOL (27.1 EFPY).

TABLE 7-1 RECOMMENDED SURVEILLANCE CAPSULE WITHDRAWAL SCHEDULE				
Capsule	**Capsule Location**	**Lead Factor**	**Withdrawal EFPY[(b)]**	**Fluence (n/cm^2)[(a)]**
T	40°	3.51	1.4	2.63×10^{18}
Y	40°	3.49	3.2	6.92×10^{18}
Z	40°	3.48	5.5	1.04×10^{19}
S	40°	3.47	(c)	(c)
U	4°	1.50	15 (d)	8.57×10^{18} (d)
V	4°	1.50	EOL	1.38×10^{19}
W	4°	1.50	EOL	1.38×10^{19}
X	4°	1.50	EOL	1.38×10^{19}

Notes:

a. Updated for the IP3 1.4% Power Uprate

b. EFPY from Plant Startup

c. IP3 tried to remove Capsule S in May of 2001; however, the capsule was not retrievable.

d. Capsule U reaches the 15 EFPY criteria (4th Capsule) before it meets the EOL fluence criteria.

TABLE 7-2
EOL (27.1 EFPY) ΔRT_{NDT} VALUES FOR ALL IP3 BELTLINE MATERIALS

Material		CF(a)	f @ 27.1 EFPY(b)	FF(c)	ΔRT_{NDT}(d)
Intermediate Shell Plate	B2802-1	137	0.898	0.969	132.8
Intermediate Shell Plate	B2802-2	152	0.898	0.969	147.3
Intermediate Shell Plate	B2802-3	136	0.898	0.969	131.8
Lower Shell Plate	B2803-1	128	0.898	0.969	124.0
Lower Shell Plate	B2803-2	150	0.898	0.969	145.35
Lower Shell Plate	B2803-3	160	0.898	0.969	155.0
→ Using Surveillance Capsule Data	B2803-3	170.9	0.898	0.969	165.6
Intermediate to Lower Shell Weld Longitudinal Weld Seams (Heat 34B009)	2-042 A,B,C 3-042 A,B,C	224	0.898	0.969	217.1
Intermediate to Lower Shell Circumferential weld Seams (Heat 13253)	9-042	189	0.898	0.969	183.1

Notes:

a. The Chemistry Factors (CFs) are documented in WCAP-15024, dated April 2001.

b. f at 27.1 EFPY is the peak 27.1 EFPY fluence at the clad/base metal interface. (x 10^{19} n.cm^2, E > 1.0 MeV)

c. Fluence Factor (FF) = $f^{(0.28 - 0.1 \log f)}$, where f is the clad/base metal interface fluence.

d. $\Delta RT_{NDT} = CF * FF$

TABLE 7-3 APPLICABILITY DATES FOR HEATUP AND COOLDOWN CURVES AT IP3	
Current EFPY	**New EFPY (1.4% Power Uprate)**
16.2	16.17

TABLE 7-4 ERG PRESSURE-TEMPERATURE LIMITS	
Applicable RT_{NDT} (ART) Value[(a)]	**ERG Pressure-Temperature Limit Category**
RT_{NDT} < 200°F	Category I
200°F < RT_{NDT} < 250°F	Category II
250°F < RT_{NDT} < 300°F	Category IIIb

Notes:

a. Longitudinally-oriented flaws are applicable only up to 250°F. Circumferentially-oriented flaws are applicable up to 300°F.

ART = Adjusted Reference Temperature

TABLE 7-5 **RT_{PTS} CALCULATIONS FOR IP3 BELTLINE REGION MATERIALS AT 27.10 EFPY WITH 1.4% POWER UPRATE FLUENCES**							
Material	**Fluence (n/cm^2, E>1.0 MeV)**	**FF**	**CF (°F)**	**ΔRT_{PTS}[c] (°F)**	**Margin (°F)**	**$RT_{NDT(U)}$[a] (°F)**	**RT_{PTS}[b] (°F)**
Intermediate Shell Plate	0.898	0.969	137	132.8	34	5	172
Intermediate Shell Plate	0.898	0.969	152	147.3	34	-4	177
Intermediate Shell Plate	0.898	0.969	136	131.8	34	17	183
Lower Shell Plate	0.898	0.969	128	124.0	34	49	207
Lower Shell Plate	0.898	0.969	150	145.4	34	-5	174
Lower Shell Plate	0.898	0.969	160	155.0	34	74	263
→ Using S/C Data	0.898	0.969	170.9	165.6	17	74	257
Intermediate and Lower Shell Weld Longitudinal Weld Seams (Heat 34B009)	0.898	0.969	224	217.1	65.5	-56	227
Intermediate to Lower Shell Circumferential weld Seams (Heat 13253)	0.898	0.969	189	183.1	56	-54	185

Notes:

a. Initial RT_{NDT} values are measured values except the intermediate and lower longitudinal welds.

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

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

CF = Chemistry Factor

FF = Fluence Factor

TABLE 7-6 **PREDICTED EOL (27.1 EFPY) USE CALCULATIONS FOR ALL THE BELTLINE REGION MATERIALS**					
Material	**Weight % of Cu**	**1/4T EOL Fluence (1019 n/cm2)**	**Unirradiated USE (ft-lb)**	**Projected USE Decrease(a) (%)**	**Projected EOL USE (ft-lb)**
Intermediate Shell Plate B2802-1	0.20	0.535	102	14	88
Intermediate Shell Plate B2802-2	0.22	0.535	97	15	82
Intermediate Shell Plate B2802-3	0.20	0.535	95	14	82
Lower Shell Plate B2803-1	0.19	0.535	72	13	63
Lower Shell Plate B2803-2	0.22	0.535	94	15	80
Lower Shell Plate B2803-3	0.24	0.535	68	17	56
Intermediate and Lower Shell Weld Longitudinal Weld Seams (Heat 34B009)	0.19	0.535	112	17	93
Intermediate to Lower Shell Circumferential weld Seams (Heat 13253)	0.22	0.535	111	18	91

Notes:

a. Values are deduced from Figure 2 of NRC Regulatory Guide 1.99, Revision 2, Predicted Decrease in Upper Shelf Energy as a function of Copper and fluence.

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 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% 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 which 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 as follows.

Core Bypass Flow Calculation

Core 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 5.2% of the total reactor vessel flow. The effect of the 1.4% power uprate has an insignificant effect on the actual core bypass flow, and the total design core bypass flow value of 5.2% remains unaffected.

Rod Control Cluster Assembly Drop Time

An evaluation was performed to demonstrate that the RCCA drop time to dashpot entry will remain within the current value of 2.7 seconds (required by the IP3 Technical Specifications) for the 1.4% power uprate design conditions. The revised design conditions for the RCCA drop time consist of the core power and the core inlet temperature (T_{cold}). The core power will increase by 1.4% from 3025 MWt to 3067.4 MWt. The design core inlet temperature has decreased by 0.4°F . The evaluation indicated that the revised conditions (primarily T_{cold}) will have a negligible effect on the drop time. The effect of the increase in core power and a 0.4°F decrease in T_{cold} increased the calculated RCCA drop times by less than 0.01 seconds. This change is considered negligible and the RCCA drop time will still be bound by the Tech Spec limit of 2.7 seconds.

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 show that the downward force remains essentially unchanged, indicating that the reactor internals would remain seated and stable for the 1.4% 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 cause 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.

In order to guard against fuel rod failures from flow-induced vibration, the cross-flow 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 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 one of these. The pressure differential across the baffle plate remains unchanged due to the 1.4% power uprate, likewise the baffle gap width and fuel assembly modal response. Therefore, the IP3 baffle joint momentum flux is unaffected by the 1.4% power uprate.

7.3.2 Mechanical Evaluations

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

With regards to flow-induced vibration, the vessel/core inlet coolant temperature decreases 0.4°F. The vessel outlet coolant temperature increases 0.4°F. These slight temperature changes cause a change in water density that have a negligible effect on the vibratory response of the reactor internals. The current design basis parameters and the 1.4% power uprate parameters remain essentially the same. Therefore, the performance of the reactor internals with regard to flow-induced vibration is unaffected.

7.3.3 Structural Evaluations

Evaluations were performed to demonstrate that structural integrity of the reactor internal components is not adversely affected by the 1.4% 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 are accounted for in the design and analysis of various components.

The core support structure components affected by the 1.4% 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% 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 CFU factors were unaffected by the 1.4% 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 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 1.4% power uprate design conditions. The CFU factor of the lower core plate (see Table 7-7), including the effects of the increase in the heat generation rates, is small, and the lower core plate structural integrity is maintained for the 1.4% power uprate.

Baffle-Barrel Region Evaluations

The baffle-barrel region consists of the core barrel into which baffle plates are installed. The baffle plates are supported by bolting interconnecting former plates to the baffle and core barrel.

The baffle-to-former bolts restrain the motion of the baffle plates that surround the core. These bolts are designed for primary loads consisting of deadweight, hydraulic pressure differentials, LOCA and seismic loads, as well as secondary loads consisting of pre-load, 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.

The thermally-induced displacements of the baffle-former bolts for the 1.4% power uprate relative to the original design conditions were calculated for a bounding range of conditions. The results demonstrated that the 1.4% power uprate conditions have a smaller thermally-induced bolt displacement than the original design conditions. Therefore, the current baffle-barrel region thermal and structural analysis results bound the revised design conditions associated with the 1.4% 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 that exits 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. Four equally spaced core plate alignment pins restrain the lateral movement.

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% power uprate was small. The results show that the structural integrity of the upper core plate is maintained for the 1.4% power uprate conditions. The CFU factor of the upper core plate caused by the increase in the heat generation rates remains less than 1.0 (see Table 7-7).

No new cumulative usage factor calculations were performed for the baffle-barrel region components since it has been demonstrated that the 1.4% power uprate conditions are bounded by the current design operating conditions.

TABLE 7-7 MARGINS OF SAFETY AND FATIGUE SUMMARY			
Limiting Component	**Calculated Stress[1], psi**	**Allowable Stress, psi**	**Cumulative Fatigue Usage Factor, ΣU**
Lower Core Plate	47,000	48,600	0.237
Baffle/Barrel Assembly	See Note 2	See Note 2	See Note 2
Upper Core Plate	22,000	48,600	0.062

Notes:

1. Primary plus Secondary Stress Intensity (NG-3222.2)
2. No new cumulative usage factor calculations were performed for the baffle-barrel region based on the demonstration in this report that the 1.4% power uprate conditions are bounded by the original design operating conditions.

7.4 PIPING AND SUPPORTS

7.4.1 Nuclear Steam Supply System Piping

The potential effect of the 1.4% power uprate on the IP3 existing reactor coolant loop (RCL) and pressurizer surge line analyses was evaluated. The parameters associated with the 1.4% 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% 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.

- NSSS Design Performance Parameters (Table 2-1)
- Thermal Design Transients (Section 5)
- LOCA Hydraulic Forcing Functions (Section 8.1)

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% 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% power uprate is addressed in Section 5 of this report. Based on the small changes in operating temperatures, Section 5 concluded that the NSSS (primary side) transients were unaffected by the 1.4% power uprate conditions. Therefore, there is no effect on the current design basis analyses for the RCL and the pressurizer surge line. Note that the RCL piping is evaluated to the USAS Code B31.1.0-1967[7-3]. 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 stipulated in NRC Bulletin 88-11 and a detailed fatigue evaluation.

The potential effect on the LOCA hydraulic forcing functions due to the 1.4% power uprate is addressed in Section 8.1 of this report. As specifically approved by the NRC[7-4], the postulation of pipe breaks in the RCS primary loop piping is no longer required for IP3. As detailed in the RCL analysis and considered in conjunction with the SG snubber elimination program for IP3, the RCL piping is evaluated for pipe breaks at the RCL branch nozzles for the pressurizer surge line, RHR, and accumulator lines. Based on the 1.4% power uprate parameters, no changes to LOCA hydraulic forcing functions are needed. Therefore, the design basis LOCA analyses performed for the RCL are unaffected by the 1.4% power uprate and still applicable. Additionally, the RCL is also evaluated for postulated pipe ruptures at the main steam and feedwater nozzles to the steam generators. The current evaluations performed for these breaks also bound the 1.4% power uprate conditions.

Based on the evaluations of the NSSS Design Parameters, Thermal Design Transients, and the LOCA Hydraulic Forcing Functions, the current design basis analyses for the IP3 RCL and pressurizer surge line remain applicable for the IP3 1.4% power uprate conditions. Additionally, there are no changes to any of the steam generator or reactor coolant loop displacements, 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% power uprate.

7.4.2 Reactor Coolant Loop Support System

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, 1969 Edition, for the revised loading associated with the deactivation of all steam generator snubbers.

The 1.4% 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 IP3 snubber elimination program remains applicable for the 1.4% power uprate.

The RCL supports were shown to meet the allowable stresses for all loading combinations for the IP3 snubber elimination program.

The steam generator, RCP, and reactor vessel supports have been qualified for piping and component loads resulting from the snubber elimination program. Since the 1.4% power uprate does not significantly change the loads exerted upon the support structures, the supports remain qualified for the 1.4% power uprate conditions.

7.4.3 Leak-Before-Break Analysis

The current leak-before-break (LBB) evaluation was performed for the primary loop piping to provide technical justification for eliminating pipe rupture as the structural design basis for IP3[7-5].

In order to demonstrate the elimination of RCS primary loop pipe breaks, the following objectives had to 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 demonstrated to be met in WCAP-8228 Volume 1, Revision 1[7-5].

There is no significant change in loads due to the 1.4% power uprate parameters as indicated in Section 7.4.1. The effect of material properties due to the changes in temperature, shown in Table 2-1, will have a negligible effect on the LBB margins[7-5]. The existing LBB analysis conclusions[7-5] remain applicable for the 1.4% power uprate conditions for IP3.

Therefore, based on the above assessment, the LBB margins will be negligibly affected, and the conclusions of WCAP-8228 Volume 1, Revision 1 for LBB[7-5] remain unchanged and applicable for the 1.4% power uprate conditions for IP3.

7.5 CONTROL ROD DRIVE MECHANISMS

The CRDM parameters are based on the hot leg data, which is the vessel outlet temperature data presented in Table 2-1. The evaluation was performed for the 1.4% power uprate NSSS power of 3082 MWt (3067.4 MWt core power). The upper bound vessel outlet temperature is shown to decrease from the current design operating temperature of 601.5°F to 600.8°F. The higher temperature evaluated for the current analysis remains bounding for the 1.4% power uprate. The design temperature and pressure for the CRDMs are 650°F and 2485 psig, respectively.

The revised pressure changes (ΔPs) and temperature changes (ΔTs) of the transients are less than those previously evaluated and continue to bound the 1.4% power uprate, with the exception of the Unit Loading/Unloading transient applied to the CRDM middle joint cono-seal hold down sleeve. The original design transient analysis for this location considered an alternating stress range of:

$$S_a = \frac{1}{2} \cdot K_f \cdot E \cdot \alpha \cdot \Delta T$$

where: K_f = stress concentration factor, taken as 4,

E = modulus of elasticity, taken as 25.4 x 10^6 psi for austenitic stainless steel at a temperature of 600°F,

α = mean coefficient of thermal expansion, taken as 9.82 x 10^{-6} inches/inch/°F for austenitic stainless steel between temperatures of 70°F and 600°F, and

ΔT = temperature difference, taken as 25°F in the original analysis.

The required ΔT is 65.6°F for the Unit Loading/Unloading transient. This difference in temperature results in an increase in the cumulative usage factor from 0.066 to 0.118, using the design fatigue curve in the Code of Record, which is the 1965 Edition of the ASME Code, Section III, with Addenda through Summer 1966. This CFU factor remains well below the allowable limit of 1.0. The acceptability of the CRDMs is thus demonstrated for the 1.4% power uprate conditions.

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 RCL. The reactor vessel inlet (RCP outlet) temperature at 100% power is 542.2°F for the 1.4% 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.

The revised ΔPs and ΔTs of the transients are less than those previously evaluated and continue to bound the 1.4% power uprate, with the exception of the heat-up/cool down transient considered for the pump casing. The original design analysis considered a ΔT of 433°F for this transient, while the transient applicable to IP3 has a ΔT of 447°F. This represents an increase of about 3% (447°F/433°F – 1 = 0.032) in the ΔT. The safety margin available between the calculated and allowable stresses for this condition is approximately 20% (i.e., 50,100 psi/41,898 psi – 1 = 0.196), so that is an increase of 3% in ΔT and the associated stress is easily accommodated. A ΔP of 2500 psi was considered in the calculation of the stress range, which envelopes the ΔPs required for all applicable transients. Therefore, the casing stress remains acceptable, and the RCP structural integrity is maintained for the 1.4% power uprate conditions.

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 IP3 and Indian Point Unit 2 (IP2) RCP motors were developed based on the IP3 impellers, steam generator outlet temperature of 515.5°F, and a single loop flow of 90,700 gpm. The results show a hot-loop motor load of 5,696 hp and a cold-loop motor load of 7,367 hp. The IP3 RCP motors have a nameplate rating of 6,000 hp hot and a design limit of 7,500 hp cold. Since the loads are less than the nameplate rating of the motors, no further analysis was necessary.

The above bounding motor evaluation applicable to IP3 was based on a flow of 90,700 gpm per loop. The 1.4% power uprate best-estimate flow (BEF) is 92,400 gpm and is, therefore, bounded by the above evaluated flow. In addition, the steam generator outlet temperature of 515.5°F associated with the above motor evaluation is lower than the steam generator outlet temperature of 541.9°F for the 1.4% power uprate. The 1.4% power uprate steam generator outlet temperature is, therefore, bounded by the above evaluated steam generator outlet temperature. Since the 1.4% power uprate BEF and steam generator outlet temperature are bounded by the above evaluated flow and temperature, those motor loads remain bounding and applicable for the 1.4% power uprate conditions.

Based upon this evaluation, the above IP3 RCP motor evaluation is bounding for the 1.4% power uprate, and the IP3 RCP motors are acceptable for operation at the 1.4% power uprate conditions.

7.7 STEAM GENERATORS

Evaluations of the thermal-hydraulic performance, structural integrity, and mechanical hardware have been performed for the IP3 Model 44F steam generators to address operation at 1.4% power uprate conditions.

7.7.1 Thermal-Hydraulic Evaluation

The thermal-hydraulic evaluations of the IP3 steam generators focus on the changes to secondary-side design operating conditions for the 1.4% power uprate. The 1.4% power uprate design operating conditions considered are shown in Table 2-1. The evaluations discussed in this section were performed to confirm the acceptability of the steam generator secondary-side parameters. The results of the thermal-hydraulic evaluations are summarized in Table 7-8. Based on these evaluations, the IP3 steam generators are qualified to operate at the 1.4% power uprate conditions based on 0% SGTP.

Bundle Mixture Flow Rate

The steam flowrate increases proportionally with the 1.4% power uprate when operating with the same T_{avg} and feedwater temperature. With the 1.4% power uprate and 0% SGTP, the calculated steam flowrate per generator increases from 3.26 to 3.31 million lb/hr, and the circulation ratio decreases from 3.70 to 3.65. Since the tube bundle mixture flowrate is the product of the circulation ratio and the steam flowrate, the resulting bundle flowrates are 12.05 and 12.06 million lb/hr, respectively, or essentially the same at both the current power level and at the 1.4% uprate power level.

The secondary fluid velocities in the U-bend region are 2% higher at the 1.4% power uprate conditions. The fluid velocities in the downcomer and at the wrapper opening are within 0.1% of the current plant operating condition values. The 1.4% power 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 the available heat transfer area in the tube bundle, and by the average primary fluid temperature. With the 1.4% power uprate and the design condition T_{avg} of 571.5°F, Westinghouse used the GENF code to perform a more rigorous thermal-hydraulic analysis. This analysis resulted in a calculated steam pressure decrease from 771.6 psia to 766.2 psia, which is in the acceptable range.

GENF is a one-dimensional steady-state thermal and hydraulic performance code developed by Westinghouse specifically for feedring 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

Average heat flux in the steam generator is directly proportional to heat load, and inversely proportional to the heat transfer area in service. For the 0% SGTP plugging case, the average heat flux increases from 59,640 Btu/hr-ft^2 at the current RTP to 60,484 Btu/hr-ft^2 at the 1.4% power uprate level.

A measure of the margin for DNB transition in the bundle is a check of the ratio of the local quality, to the estimated quality at DNB transition, or (X/XDNB). Westinghouse used the ATHOS code analyses to show that the maximum (X/XDNB) increases from 0.803 at the current RTP to 0.811 at the 1.4% power uprate level with 0% SGTP. This shows a minimal effect due to the 1.4% 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 geometry pre-processor, ATHOS solution, and post-processor modules. 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 geometry model links thermally with the primary side coolant flow. This thermal link allows the ATHOS module to calculate heat transfer from the primary coolant flow to the secondary side fluid. Therefore, the ATHOS code will calculate both heat flux and tube wall temperature in addition to typical parameters such as liquid velocity, vapor velocity, 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 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

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 one installed in the IP3 steam generators. The test results indicate that the separators are highly effective. The measured MCO was near 0.01% at full power. The operating parameters that can have an effect on moisture performance are steam flow (power), steam pressure, and water level. The MCO value for the IP3 1.4% power uprate conditions were calculated using the GENF computer code. The calculated MCO increases from 0.0025% of steam flow at the current RTP to 0.0031% at the 1.4% power uprate conditions with 0% SGTP. The predicted MCO value based on 24% SGTP increases to a maximum of 0.0042%. The MCO will be well below the 0.1% limit at the 1.4% power uprate conditions. This demonstrates that the 1.4% power uprate will have negligible effect on the moisture separator performance of the model 44F steam generators at IP3.

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 thermal-hydraulic parameters, such as flow rate or water level, will reduce in amplitude, and the steam generator will return to stable operation. The damping factor decreases from –562 hr^{-1} at nominal power to -566 hr^{-1} at the 1.4% power uprate conditions with 0% SGTP (-609.7 with 24% SGTP). The IP3 steam generators will continue to operate in a hydrodynamically stable manner at the 1.4% power uprate operating conditions.

Steam Generator Secondary Fluid Inventory

Secondary side fluid inventory consists of the mass of liquid and vapor phases. The vapor mass is approximately 6% of total inventory. With the 1.4% power uprate, the secondary fluid mass decreases from 86,899 lb to 86,515 lb, a change of less than 1%. The small changes in inventory will not affect the steam generators.

Steam Generator Secondary Side Pressure Drop

The secondary side pressure drop increases from 27.7 psi to 28.6 psi as result of 1.4% power uprate with 0% SGTP. This small increase in pressure drop should have no significant effect on the feed system operation.

Thermal-Hydraulic Evaluation Conclusions

In conclusion, the thermal-hydraulic characteristics of the IP3 Model 44F steam generators are within acceptable ranges for the 1.4% power uprate conditions with 0% SGTP.

7.7.2 Structural Integrity Evaluation

The structural evaluation focused on the critical steam generator components as determined by the stress ratios and fatigue usage. The following discussions address the evaluations of the primary-side and secondary-side components. The mechanical repair hardware evaluations are discussed later in this section.

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 updated for the 1.4% 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 differential pressure for the baseline and 1.4% power uprate conditions.

For the secondary-side components, such as the feedwater nozzle and secondary manway studs, the decrease in secondary-side pressure was the basis for determining the applicable scale factor. The scale factor was 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.

Input Parameters and Assumptions

The 1.4% power uprate structural evaluation was performed for 3082 MWt NSSS power and 0% SGTP. The applicable NSSS design parameters used for the steam generator's structural evaluation are shown in Table 2-1. The Design Transients discussed in Section 5 of this report, and the results of the primary-to-secondary side ΔP calculation were used to generate scaling factors with respect to the original stress reports results. The scaling factors were based on the steam temperature of 512.7°F, corresponding to a design steam pressure of 762 psi.

The primary side temperatures resulting from the 1.4% power uprate, (see Table 2-1), when compared to those considered in the original design basis analysis, show an increase of less than 1°F. The significant thermal stresses produced during the heat-up transients occur early in the heat-up cycle. The temperature difference for this portion of the transient would be significantly less than 1°F. Also, note that the plant heat-up is controlled through procedural constraints that limit the maximum rate of heat-up and cool-down, thereby limiting the maximum stress during the early portion of the transient. Therefore, thermally, the 1.4% power uprate will not affect the steam generator primary side components.

Likewise, secondary side components exhibit their maximum stress ranges early in the transients when the temperature differences are greatest. The steam temperature at 100% power increases by approximately 1°F. However, the no-load steam temperature remains unchanged at 547°F. Because the change in steam temperature is so small, the thermal stresses were unchanged as a result of the 1.4% power uprate.

Description of Steam Generator Component Structural Analyses and Evaluation

Scaling factors were calculated based on the change in the differential pressure between the primary side and secondary side components from the referenced stress report condition and the 1.4% power uprate condition. The scale factors were used to calculate the primary-plus-secondary stress range, and for calculation of the fatigue usage for each of the applicable transient combinations affected by the 1.4% power uprate.

For operation at the 1.4% power uprate condition, with 0% SGTP, the design steam pressure (P_{steam}) is predicted to be 762 psi. This structural evaluation was performed for a design P_{steam} of 762 psi. The original design qualification for these generators was based on a value of P_{steam} = 770 psi as reported in the original design basis stress reports. The scaling factors were calculated based on this ratio of steam pressures (762 psi vs. 770 psi).

The calculated scale factors were applied to the stresses and fatigue usage factors for all applicable transient conditions that were the basis for the existing IP3 Model 44F steam generators. Applying the scale factors to the stresses approximated the calculated values that would occur for operation at the 1.4% power uprate conditions.

Primary Side Components

The stresses in the primary side components are dependent on the differential pressure (ΔP) between the primary side and the secondary side. The additional pressure stress due to a reduction in steam pressure from the reference stress report evaluation (P_{steam} = 770 psia) to the 1.4% power uprate condition (P_{steam} =762 psia) was taken into consideration to calculate the stress range and the resulting increase in the fatigue usage as a result of the 1.4% power uprate.

Secondary Side Components

The change in stress in the secondary side components is dependent on the changes in the steam pressure as a result of increasing power by 1.4%. The increase in stress that results from a reduction in steam pressure for the 1.4% power uprate condition was calculated. The additional stress was then included in calculating the increase in stress and the resulting changes in the fatigue usage for operation at the 1.4% power uprate conditions.

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 report. 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 (ASME B&PV Code, Section III, 1965 edition through Summer 1966 Addenda) consistent with the original design basis analysis. A CFU factor of less than 1.0 demonstrates design 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. The results of the evaluation are summarized in Table 7-9.

Evaluation of Primary-to-Secondary-Side Pressure Differential

An analysis was performed to determine if the ASME B&PV Code (1965, Summer 1966 Addenda) limits on design primary-to-secondary ΔP are exceeded for any of the applicable transient conditions for the IP3 1.4% power uprate parameters. The design pressure limit for primary-to-secondary pressure differential is 1550 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 1550 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% of those defined for Design Conditions. In other words, as long as the Upset

Condition pressure values are less than 110% of the design pressure values, no additional analysis is necessary. For the IP3 steam generators, 110% of the design pressure limit corresponds to 1705 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 on Table 2-1. The pressure differentials across the primary-to-secondary side pressure boundary are calculated for these defined full-power conditions.

The analysis determined that the maximum Normal/Upset operating condition primary-to-secondary side differential pressures would be 1547 psi for Normal operating condition transients, and 1529 psi for Upset condition transients. The results show that the maximum primary-to-secondary pressure gradients are less than the allowable values of 1550 psi and 1705 psi for Normal and Upset operating conditions, respectively.

Therefore, the design pressure requirements of the ASME Code continue to be satisfied.

Evaluation of Mechanical Repair Hardware

The IP3 Model 44F replacement steam generators entered service in 1989. During the fabrication of one of the steam generators, several Westinghouse shop welded plugs were installed. These components were evaluated for the operating conditions and transients associated with the 1.4% power uprate. Also, 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 and now for operation at the 1.4% power uprate conditions. In addition, since there are circumstances that may require tube ends to be reamed, a 40% tube undercut was considered and the resulting reduced weld joint geometry was qualified for continued service.

Mechanical Plugs

The enveloping condition for the Westinghouse mechanical plug (Alloy 690 shell material) is the one that results in the largest pressure differential between the primary and the secondary sides of the steam generator. The 1.4% power uprate design performance parameters (Table 2-1) and the NSSS design transients (Section 5.1) were used to determine the effect of the 1.4% power uprate on the mechanical plugs. The most important parameter for the mechanical plug evaluation was the primary side hydrostatic pressure test in which the differential pressure across the plug is 3107 psi, and which is independent of the 1.4% power uprate.

A structural evaluation was performed for the mechanical plug for the 1.4% power uprate condition. This evaluation was performed to the applicable requirements of ASME B&PV Code, 1965 Edition through Summer 1966 Addenda.

Acceptance Criteria

The Westinghouse mechanical tube plug was evaluated for the effects of changes to the transients due to the uprate. The primary stresses due to Design, Normal, Upset, and Test conditions must remain within the respective ASME Code-allowable values. The maximum range of primary-to-secondary stresses must be less than the $3S_m$ limit. The CFU 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 parameter from the design of the plugs is the primary-to-secondary differential pressure. These calculations were based on a primary system pressure of 2250 psia.

All stress/allowable ratios are less than 1.0, 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 B&PV Code. It was also determined that adequate preload and friction are available to prevent dislodging of the plug for the limiting steady-state and transient loads.

Since mechanical plugs are components that are installed in the steam generator after initial operation, and were not part of the original steam generator, these components are typically fabricated to the requirements of the 1989 ASME B&PV Code Edition. An additional evaluation was performed based on the 1989 code year requirements. It was determined that the mechanical plugs are also acceptable for the 1.4% power uprate based on the 1989 ASME Code edition.

In summary, results of the analyses performed for the mechanical plug used in the IP3 steam generators show that both the long and short mechanical plug designs satisfy all applicable stress and retention acceptance criteria for operation at the 1.4% 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. The structural evaluation of the weld plug addressed the qualification of these plugs for these transient changes as discussed in Section 5.1.

Description of Evaluation

A structural evaluation was performed for the existing shop weld tube plugs for the 1.4% power uprate conditions. The evaluation was performed to the applicable requirements of ASME B&PV Code, 1965 Edition through Summer 1966 Addenda.

Acceptance Criteria

The installed Westinghouse shop weld plugs were evaluated for the effects of changes to the transients due to the 1.4% power uprate. The primary stresses due to Design, Normal, Upset, and Test conditions must remain within the respective ASME B&PV Code allowable values. The maximum primary-to-secondary stresses are to be less than the $3S_m$ limit. The CFU 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 and Conclusions

The evaluation of the weld plug first addressed the Design condition. A vertical failure plane around the perimeter (circumference) of the weld plug was considered. The design pressure differential of 1550 psi between the primary and secondary systems 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.

For Normal/Upset conditions, it was determined that the controlling transient for both Normal and Upset conditions was the Loss of Load transient. The differential pressure considered was 1737 psi. This was the controlling pressure condition for the baseline transient conditions. The governing differential pressure for the 1.4% power uprate was calculated at 1529 psi. It was found that the stress limits are acceptable for the 1737 psi differential pressure.

The last step in the evaluation process was an evaluation for fatigue. The approach was to investigate if the weld plug would be exempt from fatigue calculations based on the ASME B&PV Code requirements for fatigue exemption. The six required fatigue exemption conditions were determined to be satisfied. Therefore, the welded plug meets the ASME Code cycle load fatigue limits for the 1.4% power uprate.

In summary, all primary stresses are satisfied for the weld between the weld plug and the tube sheet cladding. The primary plus secondary stresses for the enveloping Loss of Load transient were determined to be acceptable. The maximum primary plus secondary stress intensity was found to be acceptable. The fatigue evaluation for the weld plug utilized the ASME fatigue exemption rules. It was found that the fatigue exemption rules were met and, therefore, fatigue conditions remain acceptable.

Tube Undercut Qualification

The field machining of steam generator tube ends is a possibility that may need to occur for modifications and repair of tubes (i.e., plugging, sleeving and tube end reopening). A portion of the tube and weld material could potentially be required to be removed by a machining process (drilling and reaming) in connection with a Westinghouse mechanical plug removal. This was addressed for the 1.4% power uprate to ensure the acceptability of a 0.020-inch of tube wall thickness undercut. The evaluation addressed situations that may result in the tube end being machined up to a 0.020-inch of tube wall thickness.

A structural evaluation was performed for the undercut of the steam generator tube ends for the 1.4% power uprate. The evaluation was performed to the applicable requirements of ASME B&PV Code, 1966 Edition through Summer 1966 Addenda.

Acceptance Criteria

The steam generator tube end undercut requires evaluation to address the design transients for the 1.4% power uprate (Section 5.1). The primary stresses due to design must remain within the respective ASME B&PV Code allowable values. The maximum range of stress intensities is to be less than the ASME $3S_m$ limit. The CFU factor must be less than or equal to 1.0, or the ASME B&PV Code fatigue exemption rules must apply, for a 40-year fatigue life for the tube.

Results and Conclusions

Past structural evaluations for steam generator tube end machining have taken place. The approach for the IP3 tube end evaluation was to utilize the results from a previous evaluation and adjust the stress values as appropriate for design transient changes. The adjustment value was conservatively based on the increase in a differential pressure across the tube sheet for the 1.4% power uprate. The results showed that all revised stresses for the 1.4% power uprate are within ASME Code allowable values.

A similar approach, using stress factors based on increased pressure differentials, was taken in the investigation of fatigue for the tube undercut machining. It was found that fatigue usage values, when adjusted for the 1.4% power uprate, remain acceptable.

In summary, the stress evaluation of the IP3 Model 44F steam generators determined that the stresses are all within ASME B&PV Code allowable values, and also that the fatigue usage factors were found to be less than 1.0.

Primary Side Loose Part Evaluation

During the September, 1990 refueling outage at IP3, it was determined that the hot leg channelhead of steam generator No. 34 had been subjected to physical impacting by a loose foreign object. The damage included denting of the steam generator tubes at the weld to the tube sheet. The damage was assessed and it was determined that the components remained structurally sound for continued operation at the current power level. This section evaluates the potentially affected welds of the hot leg tubes in the IP3 steam generator No. 34 based on the 1.4% power uprate design conditions (Table 2-1).

As a result of that loose part, the most-affected welds had significant indentations over approximately 20% of the weld circumference, and insignificant indentations over approximately 80% of the remaining circumference. Significant dents were identified as those in the analyzed portion of the tube-to-cladding fillet weld. Insignificant dents occurred in the superfluous portion of the weld, i.e., the crown, the portion extending downward from the tubesheet face, and the "rollover," the encroachment of the weld on the inside diameter of the tube at the weld. The superfluous portions of the weld are conservatively ignored in the structural evaluation of the weld.

The potential effect on the tube welds was determined by a combination of qualitative assessments of impact deformations and analytical assessments of the 1.4% power uprate on the limiting analysis section through the analyzed portion of the weld.

Acceptance Criteria

Fatigue usage at the limiting analysis section of the portion of the weld, analyzed at the current plant conditions was also evaluated for the 1.4% power uprate conditions. The fatigue usage limit, per the ASME B&PV Code, must be less than 1.0.

Results and Conclusions

The fatigue usage factor for the limiting analysis section of the non-dented tube-to-tubesheet weld for the current plant conditions was very small, i.e., 0.072. This increased slightly to 0.077 for the 1.4% power uprate conditions. Not only is the fatigue essentially negligible, but the increase due to 1.4% power uprate, is even more so.

With a low weld fatigue usage factor, fatigue would not be expected to initiate formation of a weld crack. Therefore, for the 1.4% power uprate, since the fatigue increase for uprate is negligible, a primary-to-secondary side leak would not occur due to fatigue.

Furthermore, the conclusions from the existing primary-side loose part evaluation of the impacted channelhead are unaffected and remain valid as follows:

- There was no indication that the primary-to-secondary pressure boundary at the tube-to-tubesheet weld was damaged by impacting of the loose part, to the extent that the sealing integrity of the weld was in question. This addressed the very low probability of minor leakage from the primary side to the secondary side.

- There was no concern that the tube weld was weakened by the impacting, to the extent that a large primary-to-secondary leak would be expected.

Structural Evaluation Conclusions

Results of the analyses performed on the IP3 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 limits for the 1.4% power uprate conditions. The primary-to-secondary pressure differential remains below the design value of 1550 psi. In addition, both the weld plugs and mechanical plugs remain qualified for use in the IP3 Model 44F steam generators.

7.7.3 Tube Wear

The potential effects of the 1.4% power uprate on steam generator tube wear was evaluated based on the current design basis analysis and consideration of the changes in the thermal-hydraulic properties of the secondary side of the steam generator resulting from the 1.4% power uprate. Steam generator tube wear due to flow effects is more likely to occur in the U-bend region of the tubes. An evaluation was performed to evaluate the potential effects of the 1.4% power uprate and the potential effect on tube wear.

Description of Analyses and Evaluation

The increased potential for wear of the steam generator tubes at the tube support plate (TSP)/tube intersections and at the AVB/tube intersections is evaluated based on the change in NSSS design operating parameters based on the proposed 1.4% power uprate. The flow-induced vibration and tube wear calculated for the IP3 Model 44F steam generator were reviewed. The potential for increased wear, based on changes in the secondary side fluid velocity and density associated with the 1.4% power uprate, was evaluated to determine the degree of wear that could be expected on the steam generator tubes.

The increase in the wear rate for the steam generator tubes was calculated to be 4%. The baseline tube vibration and wear analysis results for the IP3 Model 44F steam generator concluded that the predicted tube vibration amplitudes and fluid-elastic stability ratios are low, resulting in estimated tube wear of 0.0013-inch over a 40-year operating period. This is lower than the tube wear allowance used to determine the minimum tube wall thickness. Since this baseline wear is small the increase in wear due to the 1.4% power uprate is much less than 0.001-inch for a total projected wear of approximately 2 mils. This increase in wear continues to remain below the tube wear allowance of 0.003-inch.

Results and Conclusions

The baseline analysis for the IP3 Model 44F steam generators indicates that tube wear as a result of tube vibration is small over a steam generator design life of 40 years. The increase in the wear rate resulting from the 1.4% power uprate will not result in unacceptable wear, with the maximum wear projected to increase from 1.3 mils (current) to less than 2 mils (1.4% power uprate). With a typical plugging limit of 40% through-wall, (20 mils for a wall thickness of 0.050-inch), sufficient margin exists to account for anticipated wear due to vibration following the 1.4% power uprate. Therefore, tube wear following the 1.4% power uprate will remain acceptable.

7.7.4 Secondary Side Foreign Object Evaluation

Foreign object search and retrieval (FOSAR) operations performed during previous refueling outages at IP3 have determined that a number of un-retrievable objects may remain present inside the steam generators. Analyses were previously performed to address the 10 objects identified during the 1997 refueling outage, and the 28 objects identified during the 1999 outage. Although there was no indication of wear present on any of the tubes adjacent to the foreign objects, wear time analyses were performed assuming that 20% initial tube wear was present on the limiting tubes. These analyses determined that the amount of time required for the limiting foreign object to wear a tube down to a minimum allowable tube wall thickness was greater than 5 years for the objects, identified in 1997, and greater than 7.2 years for the objects identified in 1999.

Description of Analyses and Evaluations

These previous evaluations were assessed to determine the potential effects of the 1.4% power uprate on the projected wear times of the objects. This assessment has determined that, although the steam generator secondary side conditions will change slightly as a result of the 1.4% power uprate, these changes do not significantly affect the previously calculated wear times. For the cases considered, the worst resulted in an insignificant decrease in wear time. Calculations indicate that this reduction in wear time would be on the order of less than 1% of the previously calculated wear times. Changes of this magnitude are considered to be insignificant and will not affect the conclusions presented in the earlier analysis.

Results and Conclusions

In summary, the analysis determined that for the 1.4% power uprate operating condition, the changes in previously calculated secondary side foreign object wear times are not significant and do not affect the previously stated conclusions.

7.7.5 Regulatory Guide 1.121 Analysis

The heat transfer area of steam generators in a PWR NSSS comprises over 50% 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 to include an operational allowance for continued tube degradation until the next scheduled inspection.

NRC Regulatory Guide 1.121, "Bases for Plugged Degraded PWR Steam Generator Tubes," describes an acceptable method for establishing the limiting safe condition of degradation in the tubes beyond which tubes found defective by the established in-service inspection shall be removed from service. The level of acceptable degradation is referred to as the "repair limit."

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 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, 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 and AVB intersections.

A summary of the resulting tube structural limits is provided in Table 7-10.

7.7.6 Tube Fatigue

The IP3 Model 44F steam generators utilize stainless steel tube support plates. Generators with stainless steel support plates are not subject to steam generator tube fatigue concerns, as discussed in NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes," dated February 5, 1988.

TABLE 7-8 RESULTS OF THERMAL-HYDRAULIC EVALUATIONS		
Operating Conditions	**Case 0**	**Case 1***
Power, %	100.0	101.4
Reactor Power, MWt	3025	3068
NSSS Power, MWt	3039	3082
Power per SG, MWt	759.75	770.50
Primary Flow per Loop, GPM	89,700	89,700
Primary Fluid Pressure, psia	2250	2250
Primary Average Temperature, °F	571.5	571.5
Feedwater Temperature, °F	427.2	427.8
Water Level Above Tubesheet, in	459	459
Blowdown Flow Rate, lb/hr	33,150	33,150
Tube Plugging, % of Total Tubes	0	0
T/H Characteristics		
Steam Flow Rate per SG, 10^6 lb/hr	3.2569	3.3052
Steam Pressure at Nozzle, psia	771.59	766.18
Circulation Ratio	3.70	3.65
Separator Parameter	8.5971	8.8806
Moisture Carryover, %	0.0025	0.0031
Downcomer Fluid Velocity, ft/sec	12.15	12.14
Sec. Side Pressure Drop, psi	27.738	28.574
Sec. Fluid Liquid Mass, lb	81,644	81,282
Sec. Fluid Vapor Mass, lb	5,255	5,233
Sec. Fluid Heat Content, 10^6 Btu	46.344	46.062
Average Heat Flux, Btu/hr-ft^2	59,640	60,484
Damping Factor, 1/hr	-562.2	-565.9
SG Inlet, T_{hot} – Temperature, °F	600.54	600.93
SG Outlet, T_{cold} – Temperature, °F	542.46	542.07
Maximum (X/XDNB)	0.803	0.811

* Based on Set 1 from Table 2-1.

TABLE 7-9 IP3 1.4% POWER UPRATE EVALUATION SUMMARY PRIMARY AND SECONDARY-SIDE STEAM GENERATOR 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 Fatigue	notes 0.664	notes 0.683	69.90 1.00	Plastic analysis performed
Tubesheet & Shell Juncture	Normal/ Upset	Pm+Pb+Q Fatigue	79.9 0.356	80.46 0.365	80.10 1.00	
Tube-to-Tubesheet Weld[2]	Normal/ Upset	Pm+Pb+Q Fatigue	notes 0.072	notes 0.077	69.90 1.00	
Tubes	Normal/ Upset	Pm+Pb+Q Fatigue	59.77 0.142	60.61 0.146	79.8 1.00	
Secondary Side Components [1]						
Main Feedwater Nozzle	Normal/ Upset	Pm+Pb+Q Fatigue	76.02 0.994	77.08 1.000	80.10 1.00	
Secondary Manway Stud	Normal/ Upset	Pm+Pb+Q Fatigue	80.20 0.665	80.67 0.700	94.5 ($2.7S_m$) 1.00	
Steam Nozzle	Normal/ Upset					
Section A-A		Pm+Pb+Q Fatigue	57.96 0.020	58.11 0.021	80.10 1.00	
Nozzle Insert		Pm+Pb+Q [2] Fatigue	62.97 [2] 0.865	63.266 [2] 0.868	56.07 1.00	Simplified plastic analysis
Support Ring		Pm+Pb+Q [2] Fatigue	46.527 [2] 0.190	46.823 [2] 0.191	36.50 1.00	Simplified plastic analysis

Notes

1. Additional stress due to the reduction of pressure is considered to calculate the increase in stress range for secondary side components.
2. Exceeds $3S_m$ simplified plastic analysis was done in the reference analysis for fatigue evaluation.

TABLE 7-10

IP3 1.4% POWER UPRATE EVALUATION

SUMMARY OF TUBE STRUCTURAL LIMITS

Location / Wear Scar Length	Parameter	Value
Straight Leg	t_{min} (inch)	0.022
	Structural Limit (%)	56.0
Anti-Vibration Bar/0.5" (Tube Rows 16 – 45)	t_{min} (inch)	0.015
	Structural Limit (%)	71.0
Anti-Vibration Bar/0.75" (Tube Rows 1 – 15)	t_{min} (inch)	0.018
	Structural Limit (%)	64.6
Flow Distribution Baffle/0.75"	t_{min} (inch)	0.018
	Structural Limit (%)	64.6
Tube Support Plate/1.125"	t_{min} (inch)	0.021
	Structural Limit (%)	59.0

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

$t_{nom} = 0.050$ in

7.8 PRESSURIZER

7.8.1 Structural Analysis

An analysis was performed to assess the potential effect of the 1.4% power uprate on the pressurizer components. The conditions that affect the primary plus secondary stresses, and the primary plus secondary plus peak stresses, are the changes in the design T_{hot}, T_{cold}, and the pressurizer transients. A review of the revised temperature parameters showed that the changes in T_{hot} and T_{cold} are very small and are enveloped by the current stress analysis of record.

The operating temperature analyzed for the pressurizer is 653°F. The following is a summary of the temperature changes incurred (see Table 2-1 for T_{hot} and T_{cold} data):

Parameter	Value for 1.4% Uprate Conditions	$\Delta T_{\text{Current stress report}}$	ΔT_{Uprate}
T_{hot} °F	600.8	125.0	653-600.8 = 52.2
T_{cold} °F*	541.9	125.0	653-541.9 = 111.1

* steam generator outlet

For components potentially affected by T_{hot} (e.g., the surge nozzle) and T_{cold} (e.g., the spray nozzle), the temperature difference for the 1.4% power uprate parameters is bounded by the ΔT of 125°F listed in the current stress report.

No changes were made to the primary system design transients (see Section 5) and, therefore, the transients specified in the current pressurizer design specification remain applicable. For this reason, the 1.4% power uprate will not affect the current pressurizer stress analysis and fatigue analysis. These analyses demonstrate that the pressurizer components will continue to meet the stress/fatigue analysis requirement of the ASME Code, Section III for the plant operation at the 1.4% power uprate conditions.

7.9 NSSS AUXILIARY EQUIPMENT

NSSS auxiliary equipment includes the heat exchangers, pumps, valves, and tanks. An evaluation was performed to determine the potential effect that the 1.4% power uprate design conditions will have on theses components. Section 5.2 has already 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 NSSS auxiliary equipment are unaffected by the 1.4% power uprate.

7.10 FUEL EVALUATION

This section summarizes the evaluations performed to determine the potential effects of the 1.4% power uprate on the nuclear fuel at IP3. Fuel evaluations are performed for each specific IP3 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(7-6). The evaluations herein address fuel-related analyses that are not cycle-specific, and that are directly affected by the 1.4% 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% power uprate on these analyses were evaluated in terms of the IP3 fuel/core nuclear design, the fuel rod design, the core thermal-hydraulic design, and the fuel structural integrity. This was done based on both Westinghouse VANTAGE+ fuel (currently being used in Cycle 12 core at IP3) and VANTAGE5 fuel (could possibly be taken from the IP3 SFP and used in future cycle cores at IP3).

The potential effects of the 1.4% power uprate on the analyses that are cycle-specific will be addressed consistent with the Westinghouse Reload Methodology(7-6) 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% power uprate, and there is sufficient margin to applicable limits to accommodate a 1.4% 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% power uprate at IP3 during Cycle 12.

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(7-6). These analyses will be repeated prior to implementation of the 1.4% power uprate of IP3 during Cycle 12 as well as prior to all subsequent cycles.

7.10.2 Fuel Rod Design

The current fuel rod design analyses for IP3 were reviewed to assess the potential effect of the 1.4% power uprate. The design margin for rod internal pressure (gap re-opening) and cladding stress were re-evaluated based on the 1.4% power uprate conditions. The results indicate that these fuel rod design parameters continue to meet the acceptance criteria at the 1.4% power uprate conditions. Other 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% power uprate.

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(7-6). These analyses will be repeated prior to the implementation of the 1.4% power uprate of IP3 during Cycle 12, as well as all prior to subsequent cycles.

7.10.3 Core Thermal-Hydraulic Design

A core thermal-hydraulic evaluation was performed at the 1.4% power uprate nominal core power level of 3067.4 MWt. The evaluation was based on the 15x15 VANTAGE+ fuel design with the IFM grids that bounds the 15x15 VANTAGE5 fuel design for future reloads. The key input parameters are summarized in Table 7-11.

The current design methodology for the IP3 reload safety evaluation remains unchanged for the 1.4% power uprate evaluation. The WRB-1 DNB correlation and the 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 7-12.

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% power uprate analysis is in full compliance with the conditions specified in the NRC Safety Evaluation Report (SER) on WCAP-14565[7-7].

Based on the parameter values in Table 7-11, VIPRE DNBR calculations were performed for the 15x15 VANTAGE+ w/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% 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 steamline break events, all meet the revised DNBR Safety Analysis Limits in Table 7-12.

Because of the significant amount of burnup, the 15x15 VANTAGE 5 fuel is less limiting than the VANTAGE+ w/IFM fuel. The VANTAGE 5 fuel assemblies may be used for future reloads with a maximum $F_{\Delta H}$ 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 IP3 fuel designs in support of the 1.4% power uprate, based on the same methodology used for the IP3 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% power uprate nominal core power level of 3067.4 MWt.

7.10.4 Fuel Structural Evaluation

The VANTAGE 5 and VANTAGE+ assembly designs were evaluated to determine the potential effect of the 1.4% power uprate on the fuel assembly structural integrity. The original core plate motions remain applicable for the 1.4% power uprate. Therefore, there is no effect on the fuel assembly seismic / LOCA structural evaluation. The 1.4% 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 IP3, if justified by cycle-specific evaluations.

TABLE 7-11 SUMMARY OF KEY THERMAL HYDRAULIC INPUT PARAMETERS		
Parameter	**Cycle 12 (Current Power)**	**Power Uprate**
Core Power, MWt	3025	3067.4
Minimum Measured Flow[1], gpm	330,800	330,800
Thermal Design Flow[2], gpm	323,600	323,600
Best-Estimate Core Bypass Flow[1]	4.7	4.7
Design Core Bypass Flow[2]	5.2	5.2
Vessel Average Temperature, °F	574.4	574.4
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).

TABLE 7-12 **RTDP DNBR LIMITS AND MARGIN SUMMARY FOR 1.4% CORE POWER UPRATE**		
Parameter	**15x15 VANTAGE5 Fuel**	**15x15 VANTAGE+ Fuel**
DNB Correlation	WRB-1	WRB-1
Safety Analysis Limit DNBR	1.45	1.45[1]
Design Limit DNBR (Typical / Thimble Cells)	1.24 / 1.23	1.23 / 1.23
Margin Reserved (% DNBR)	14.5	15.0
DNBR Penalties (% DNBR):		
a) Rod Bow	3.0	3.0[2]
b) Transition Core	6.3	< 8.0
c) Locked Rotor (non-OTΔT)	2.6	N/A
d) Core Limits (OTΔT)	3.3	N/A
e) Core Bypass Flow Increase	1.6	N/A
Net Margin Available (% DNBR)	> 0.3 (OTΔT Events) > 1.0 (Non-OTΔT Events)	> 4.0

Notes:

1. Equivalent to the Safety Analysis Limit DNBR of 1.54 minus penalties for the 1.4% power uprate, and for core bypass flow increase due to VANTAGE+ fuel.
2. Applicable to the grid spans without IFM grids.

7.11 SECTION 7 REFERENCES

7-1 Emergency Response Guidelines - Revision 1B, Westinghouse Owners Group, February 28, 1992.

7-2 ASTM E185-82, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," E706 (IF), in ASTM Standards, Section 3, American Society for Testing and Materials, Philadelphia, PA, 1993.

7-3 USA Standard Code for Pressure Piping, Power Piping USAS B31.1.0 – 1967.

7-4 USNRC, Docket Nos. 50-286, Letter to John C. Brons, Senior Vice President – Nuclear Generation, Power Authority of the State of New York, By Steven A. Varga, Director, PWR Project Directorate No. 3, Division of PWR Licensing – A, March 10, 1986, Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Elimination of Large Primary Loop Ruptures as a Design Basis, Power Authority of the State of New York, Indian Point Nuclear Generating Unit No. 3.

7-5 WCAP-8228, Volume 1, Revision 1, "Structural Evaluation of the Reactor Coolant Loop/Support System for Indian Point Nuclear Generation Station Unit No. 3,"April 1997.

7-6 WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," Davidson, S., et al, July 1985.

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

8 UFSAR CHAPTER 14 ACCIDENT ANALYSES AND CALCULATIONS

Table 8-1 summarizes the UFSAR Chapter 14 accident analyses and calculations for IP3, and identifies whether each analysis is affected or unaffected (according to NRC RIS 2002-03) by the 1.4% power uprate. Details of how these analyses were evaluated or reanalyzed for the 1.4% power uprate follow in later subsections.

TABLE 8-1

ACCIDENT ANALYSIS DESIGN BASIS EVENTS

UFSAR Section	Event Description	Affected or Unaffected (RIS 2002-03)
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 Reanalyzed for 1.4% Power Uprate		
14.1.2	Uncontrolled Control Rod Assembly Withdrawal at Power	Affected
14.1.8	Loss of External Electrical Load – DNB Analysis	Affected
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		
14.1.3	Rod Assembly Misalignment	Affected
14.1.4	Rod Cluster Control Assembly (RCCA) Drop	Affected
14.1.6	Loss of Reactor Coolant Flow (Partial and Complete Loss of Flow)	Affected
14.1.6	Reactor coolant Pump Shaft Seizure (Locked Rotor) and Shaft Break	Affected

TABLE 8-1 (CONT.) ACCIDENT ANALYSIS DESIGN BASIS EVENTS		
UFSAR Section	**Event Description**	**Affected or Unaffected (RIS 2002-03)**
Non-LOCA Events (Cont.)		
Events Bounded by Current 102% Power Assumption		
14.1.6	Reactor Coolant Pump Shaft Seizure (Locked Rotor) and Shaft Break – Overpressure, Maximum Clad Temperature, and Maximum Zirconium-Water Reaction Analyses	Unaffected
14.1.8	Loss of External Electrical Load – Overpressure Analysis	Unaffected
14.1.9	Loss of Normal Feedwater	Unaffected
14.1.12	Loss of All AC Power to the Station Auxiliaries	Unaffected
14.2.6	Rupture of a Control Rod Drive Mechanism Housing (RCC Assembly Ejection) – Full Power Analysis	Unaffected
Non-Limiting / Bounded Events		
14.1.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical Condition	Unaffected
14.1.5	Chemical and Volume Control System Malfunction	Unaffected
14.1.7	Startup of an Inactive Reactor Coolant 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 Drive Mechanism Housing (RCC Assembly Ejection) – Zero Power Analysis	Unaffected
14.3.6	Steam Line Break Mass and Energy Releases and Containment Integrity Long Term Inside Containment	Unaffected
14.3.6	Steam Line Break Mass and Energy Releases and Containment Integrity Long-term Outside Containment/Equipment Qualification Input	Unaffected
14.3.7	Post-LOCA Hydrogen	Unaffected

TABLE 8-1 (CONT.) ACCIDENT ANALYSIS DESIGN BASIS EVENTS		
UFSAR Section	**Event Description**	**Affected or Unaffected (RIS 2002-03)**
	Non-LOCA Events (Cont.)	
14.3.6	LOCA Mass and Energy Releases and Containment Integrity Long-term	Unaffected
14.3.6	LOCA Mass and Energy Releases and Containment Integrity Short-term	Unaffected
	Radiological Dose Calculations	
14.2.1	Fuel Handling Accident Dose	Unaffected
14.2.2	Accidental Release of Waste Liquid Dose	Unaffected
14.2.3	Accidental Release of Waste Gas Dose	Unaffected
14.2.4	Steam Generator Tube Rupture and Radiological Dose	Unaffected
14.2.5.5	Rupture of a Steam Pipe Dose	Unaffected
14.3.5	LOCA Dose	Unaffected

8.1 LOCA HYDRAULIC FORCES

8.1.1 Analysis Results

LOCA hydraulic forces are used as input for faulted-condition structural qualification of the RCS, including the reactor vessel, internals, fuel, loop piping and primary system supports. LOCA forces are not directly affected by 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. LOCA forces are calculated at full power, minimum temperature, and 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 1.4% power uprate conditions were evaluated relative to the conditions assumed in the current LOCA forces calculations of record.

The current LOCA hydraulic forces calculations applicable to IP3 for qualification of reactor vessel and internals were developed using the NRC reviewed and approved MULTIFLEX computer code[8-1]. These LOCA forces were originally calculated for IP2 utilizing accumulator and pressurizer surge line breaks, and assumed a minimum cold leg operating temperature of 515.8°F. The IP2 forces were determined to be conservatively applicable to IP3. The minimum cold leg temperature for the IP3 1.4% power uprate program is 539.2°F. This may be reduced to 534.2°F by accounting for a 5°F rod control average temperature uncertainty. The IP2 calculation included an RCS pressurizer pressure of 2250 psia plus 30 psi uncertainty for a total of 2280 psia. The IP3 full-power operating pressure is 2250 psia with a 60 psi uncertainty on pressurizer pressure for a total of 2310 psia. Using established sensitivities to temperature and pressure, Westinghouse determined that the IP2 LOCA forces remain conservatively applicable and bound the IP3 1.4% power uprate conditions by more than 10%.

The LOCA forces calculations currently applied to IP3 for qualification of the loop piping and supports were also performed using the NRC reviewed and approved MULTIFLEX computer code[8-1]. These loop LOCA forces calculations were developed using accumulator and pressurizer surge line breaks and a minimum cold leg operating temperature of 515.7°F at an RCS pressure of 2310 psia. Since these conditions bound the IP3 1.4% power uprate conditions, the currently applicable loop LOCA forces remain bounding.

In addition to these analyses, which are part of the existing design basis, there were two generic LOCA forces analyses developed to bound IP3 such that the results may be applied to IP3 as needed. The first of these was an acceptable baffle-barrel bolting analysis[8-2], performed in accordance with methodology reviewed and approved by the NRC[8-3]. The second was a control rod insertion analysis for cold leg break LOCA[8-4]. This control rod insertion analysis was performed using methodology identical to that applied to D. C. Cook[8-5], which was reviewed and approved by the NRC[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 bound the IP3 1.4% power uprate conditions. Both generic analyses[8-2, 8-4] confirm that 15x15 VANTAGE+ fuel remains qualified under bounding LOCA and seismic loads for IP3.

8.2 LOCA AND LOCA-RELATED EVALUATIONS

8.2.1 Appendix K LBLOCA and SBLOCA

The current licensing basis large-break LOCA (LBLOCA) and small-break LOCA (SBLOCA) analyses employ a nominal core power of 3025 MWt. The current licensing basis methodology applies a 2% calorimetric power measurement uncertainty allowance resulting in an assumed core power of 3086 MWt in accordance with the original requirements 10 CFR 50, Appendix K. Consistent with the subject change to 10 CFR 50 Appendix K, the existing 2% uncertainty allowance in the LBLOCA and SBLOCA analyses is being reallocated. First, 1.4% is allocated to the increase in the licensed core power. Second, 0.6% is allocated to account for the actual power measurement uncertainty calculation based on the use of the Caldon LEFM system. Therefore, the 3086 MWt total core power (including uncertainties) assumed in the current analyses is unaffected.

8.2.2 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 ECCS water contained in the RCS/sump following a LOCA. Since credit for the control rods is not taken for a LBLOCA, the borated ECCS water provided by the RWST and accumulators must have a boron concentration that, when mixed with other sources of water, will result in the reactor core remaining subcritical, assuming all control rods out. The 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, ECCS/RHR piping, and the spray additive tank. The water volumes and associated boric acid concentrations are not directly affected by the 1.4% power uprate. The core reload licensing process will confirm that there are no required changes to these volumes and boron concentrations. Therefore, the current long-term core cooling analysis of record is unaffected by the 1.4% power uprate.

8.2.3 Hot Leg Switchover

For a post-LOCA cold leg break, some of the ECCS injection into the cold leg will circulate around the top of the full downcomer and out the broken cold leg. Flow stagnation in the core and the boiling off of nearly pure water will increase the boron concentration in the remaining water. As the boron concentration increases, the boron would eventually precipitate and potentially inhibit core cooling. To preclude this from happening, the ECCS configuration is switched to hot leg injection, at a designated time after a LOCA, to flush the core with water and keep the boron concentration below the precipitation point. The current licensing basis analysis methodology employs a 2% calorimetric power uncertainty allowance, in accordance with the original requirements of 10 CFR 50, Appendix K. This existing 2% uncertainty allowance in the current hot leg switchover analysis is being reallocated with 1.4% applied to the increase in the licensed core power and 0.6% allocated to account for the actual power measurement uncertainty based on the use of the Caldon LEFM system. The 3086 MWt total core power (including uncertainties) assumed in the analysis remains bounding, and the current analysis remains unaffected.

8.3 NON-LOCA ANALYSES AND EVALUATIONS

This section addresses the effects of the IP3 1.4% power uprate on the non-LOCA analyses presented in Chapter 14 of the IP3 UFSAR.

Non-LOCA design-basis events are documented in Sections 14.1 and 14.2 of the IP3 UFSAR. Three of the affected UFSAR non-LOCA events were reanalyzed, and the four other affected events were simply evaluated to address the potential effects of the 1.4% power uprate. The remaining analyses were determined to be unaffected, as described herein.

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

- Uncontrolled Control Rod Assembly 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% 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 RTDP methodology[8-7].

- Rod Assembly Misalignment (UFSAR Section 14.1.3)
- Rod Cluster Control Assembly (RCCA) Drop (UFSAR Section 14.1.4)
- Loss of Reactor Coolant Flow (UFSAR Sections 14.1.6) – Partial and Complete Loss of Flow Analyses
- Reactor Coolant Pump Shaft Seizure (Locked Rotor) and Shaft Break (UFSAR Section 14.1.6)

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

- Reactor Coolant Pump Shaft Seizure (Locked Rotor) and Shaft Break (UFSAR Section 14.1.6) – Overpressure, Maximum Clad Temperature, and Maximum Zirconium-Water Reaction Analyses
- 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 Drive 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% uprate were evaluated, and it was determined that the current analyses of record for these events remain valid for the 1.4% power uprate conditions (i.e., these analyses are unaffected).

The following group of 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%.

- Uncontrolled Control Rod Assembly Withdrawal from a Subcritical Condition (UFSAR Section 14.1.1)
- Chemical and Volume Control System 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 Drive Mechanism Housing (RCC Assembly Ejection) (UFSAR Section 14.2.6) – Zero Power Analysis
- Anticipated Transients Without Scram

The following event is prohibited by the IP3 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

Design operating parameters that were used as a basis for the evaluations and analyses performed to support the 1.4% 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 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% power uprate is the power measurement uncertainty, which now is ± 0.6%. 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 OTΔT and 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[8-8], and 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% 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% 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% 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 sections.

8.3.3 Affected Non-LOCA Events Reanalyzed for the 1.4% Power Uprate

As shown in Table 8-1, three of the IP3 non-LOCA events have been analyzed in support of the 1.4% power uprate. Each of the analyses specifically models the increased power level. These events and analyses are described in the following three subsections of this report.

8.3.3.1 Uncontrolled Control Rod Assembly 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 faulty operator action 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 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.

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

2. 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.

3. A reactor trip is initiated if any two-out-of-four OPΔT channels exceed an OPΔT setpoint.

4. 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.

5. 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 MSS pressure is maintained below 110% of the design pressure. The RCCA bank withdrawal at power analysis described in UFSAR Section 14.1.2 (that assumes a 4% pressurizer safety valve tolerance) was evaluated for the 1.4% 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 IP3 analysis.

Since all applicable acceptance criteria continue to be met for the 1.4% power uprate, this event will not adversely affect the IP3 core, RCS and MSS.

TABLE 8-2 ANALYSIS DETAILS FOR UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL AT POWER	
Key Inputs	• *Initiating event: RCCA bank withdrawal* • *A spectrum of reactivity insertion rates ranging from 0.6 pcm/sec to 100 pcm/sec were examined at 10%, 60%, and 100% of the 1.4% uprate 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 (1.4% Uprate) and 1.34 (VANTAGE+ Fuel Reload Transition Safety Report {RTSR})].* • *A conservatively high neutron flux reactor protection setpoint of 118% of the 1.4% uprated RTP was assumed.*
Methodology	*The applied methodology is consistent with the current licensing basis analysis presented in the UFSAR supporting the VANTAGE+ Fuel 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 safety analysis limit (SAL) is 1.54 for the 1.4% power uprate program, corresponding to the WRB-1 DNBR correlation. In comparison, the DNBR SAL for the VANTAGE+ Fuel RTSR program was 1.54, 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 (CONT.) **ANALYSIS DETAILS FOR UNCONTROLLED CONTROL ROD ASSEMBLY WITHDRAWAL AT POWER**	
Calculated Results	• *For the 1.4% power uprate, the minimum DNBR calculated using LOFTRAN is 1.7165 and corresponds to a case initiated from 100% power assuming minimum reactivity feedback conditions and a reactivity insertion rate of 2.0 pcm/sec.* • *For the VANTAGE+ Fuel RTSR, the minimum DNBR calculated using LOFTRAN is 1.828 and corresponds to a case initiated from 100% power assuming minimum reactivity feedback conditions and a reactivity insertion rate of 5.0 pcm/sec.* • *The peak secondary pressure calculated for the 1.4% power uprate is 1204 psia.* • *The peak core average heat flux calculated for the 1.4% power uprate is 117.1%.* • *The peak primary pressure that was calculated previously and remains limiting is 2746 psia.*

8.3.3.2 **Loss of External Electrical Load** (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 because this bounds both the loss of external electrical load event and the turbine trip event. 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 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:

- 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, electrical power is available for the continued operation of plant components, such as the RCPs.

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 RCS 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 could exceed the values at which the integrity of these 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 IP3 UFSAR 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 available (the case in which the DNB design basis is examined) was addressed to support of the 1.4% power uprate. The case in the licensing-basis analysis in which pressurizer pressure control to be unavailable (the case in RCS and MSS overpressure criteria are examined) is not affected by an increase in the nominal full-power because the power level assumed in the current analysis for this case (with 2% uncertainty) is equivalent to that based upon the uprated power of 3068 MWt, combined with the lower uncertainty of 0.6% and remain bounding for the 1.4% power uprate. Therefore, the results of the case analyzed without pressurizer pressure control available are unaffected and the turbine trip presents no hazard to the integrity of the RCS or the MSS pressure boundary.

As stated earlier in this section, the turbine trip analysis bounds the total loss of external electrical load event for IP3 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 limit value.

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

Since all applicable acceptance criteria continue to be met for the 1.4% power uprate, this event will not adversely affect the IP3 core, RCS and MSS.

TABLE 8-3 **DETAILS FOR LOSS OF EXTERNAL ELECTRICAL LOAD**	
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 1.34 (VANTAGE+ Fuel RTSR)].* • *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 VANTAGE+ Fuel 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 safety analysis limit (SAL) is 1.54 for the 1.4% power uprate program, corresponding to the WRB-1 DNBR correlation. In comparison, the DNBR SAL for the VANTAGE+ Fuel RTSR program was 1.54, corresponding to the WRB-1 DNBR correlation.*
Calculated Results	• *For the 1.4% power uprate, the minimum DNBR calculated using LOFTRAN is 2.2343.** • *For the VANTAGE+ Fuel RTSR, the minimum DNBR calculated using LOFTRAN is 2.176.*
** The DNBR value corresponding to nominal conditions is used in the calculation of DNBR values during transient conditions. For the VANTAGE+ Fuel RTSR, the nominal DNBR value that was used in calculating the transient DNBR values was that corresponding to VANTAGE5 fuel. This value was used because nominal DNBR values for VANTAGE5 fuel are lower than nominal DNBR values calculated for VANTAGE+ fuel at the same conditions and using the lower DNBR value corresponding to the VANTAGE5 fuel bounded the transition cores. However, since the transition to VANTAGE+ fuel at IP3 has since been completed, 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 existing RTSR calculations - which had to bound transition cores. Therefore, the minimum DNBR values that were calculated for this transient for the 1.4% power uprate improved relative to those values calculated for the VANTAGE5 / VANTAGE+ Fuel transition cores as documented in the RTSR.*	

8.3.3.3 Excessive Heat Removal Due to Feedwater System Malfunctions Full Power Analysis (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 an FCV. At power, this excess flow would cause 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, therefore, 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 (i.e., high neutron flux, OTΔT, overpressure delta temperature [OPΔT], etc.) 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 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% power uprate. The licensing-basis analysis also addresses cases that are initiated at 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 HZP conditions are unaffected and continue to remain bounding for the 1.4% power uprate.

The results of this analysis show that the minimum DNBR calculated for an excessive feedwater addition at power will remain above the safety analysis limit value. Therefore, the DNB design basis is met. With regard to the RCS and MSS over-pressure criteria, this event is bounded by the turbine trip analysis documented in Section 8.3.3.2.

Table 8-4 provides details concerning the key input assumptions, methodology, safety analyses limits, and calculated results for excessive heat removal due to feedwater system malfunctions analysis.

Since all applicable acceptance criteria continue to be met for the 1.4% power uprate, this event will not adversely affect the IP3 core, RCS and MSS.

TABLE 8-4 **ANALYSIS DETAILS FOR EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTIONS**	
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 250% 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 minimized at the value that corresponds to the nominal level minus instrument uncertainties [45% 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 VANTAGE+ Fuel 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 safety analysis limit (SAL) is 1.54 for the 1.4% power uprate program, corresponding to the WRB-1 DNBR correlation. In comparison, the DNBR SAL for the VANTAGE+ Fuel RTSR program was 1.54, 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 - 2.063 (1.4% uprate)** *- 1.975 (VANTAGE+ Fuel RTSR)* • *Manual rod control - 2.113 (1.4% uprate)* *- 1.998 (VANTAGE+ Fuel RTSR)*

TABLE 8-4 (CONT.) **ANALYSIS DETAILS FOR EXCESSIVE HEAT REMOVAL DUE TO FEEDWATER SYSTEM MALFUNCTIONS**
* *The DNBR value corresponding to nominal conditions is used in the calculation of DNBR values during transient conditions. For the VANTAGE+ Fuel RTSR, the nominal DNBR value that was used in calculating the transient DNBR values was that corresponding to VANTAGE5 fuel. This value was used because nominal DNBR values for VANTAGE5 fuel are lower than nominal DNBR values calculated for VANTAGE+ fuel at the same conditions and using the lower DNBR value corresponding to the VANTAGE5 fuel bounded the transition cores. However, since the transition to VANTAGE+ fuel at IP3 has since been completed, 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 existing RTSR calculations - which had to bound transition cores. Therefore, the minimum DNBR values that were calculated for this transient for the 1.4% power uprate improved relative to those values calculated for the VANTAGE5 / VANTAGE+ Fuel transition cores as documented in the RTSR.*

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

8.3.4.1 Rod Assembly Misalignment and Rod Cluster Control Assembly (RCCA) Drop
(UFSAR Sections 14.1.3 and 14.1.4)

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[8-9], and are examined to demonstrate that the DNB design basis is met.

The methodology described in WCAP-11394 involves the use of generic statepoints for the dropped rod event. Sensitivity studies on the effect of a power increase on the generic statepoints were previously performed for a four-loop plant. The studies quantified the effect of an ~5% increase in power on the four-loop generic statepoints, and found that the statepoints were still applicable for use at the 1.4% power uprate conditions. Therefore, since the IP3 1.4% power uprate is much smaller than the uprate used in the sensitivity studies (~5%), the generic statepoints also continue to apply to IP3.

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 heat core flux confirmed that the DNB design basis continues to be met. Therefore, all applicable acceptance criteria continue to be met for the 1.4% power uprate.

8.3.4.2 Loss of Reactor Coolant Flow (UFSAR Sections 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 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 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 limit value. The limiting results are obtained at full-power conditions and occur very quickly following initiation of the event.

Since the 1.4% 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% 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 will still be met for the 1.4% power uprate.

8.3.4.2.2 Reactor Coolant Pump Shaft Seizure (Locked Rotor) and Shaft Break

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% 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 case completed to confirm that the RCS pressure criterion is met was not reanalyzed, since it currently models a 2% power uncertainty. As such, the RCS pressure criterion continues to be met for the locked rotor event.

The analysis of the locked rotor event conservatively bounds the reactor coolant pump shaft break event presented in UFSAR section 14.1.6. Therefore, all applicable acceptance criteria will still be met for the 1.4% power uprate.

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

8.3.5.1 Reactor Coolant Pump Shaft Seizure and Shaft Break
(UFSAR Section 14.1.6)

The case completed to confirm that the RCS pressure criterion is met was not reanalyzed, since it currently models a 2% power uncertainty. As such, the RCS pressure criterion continues 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 available (the case in which the DNB design basis is examined) was addressed to support of the 1.4% power uprate. The case in the licensing-basis analysis in which pressurizer pressure control to be unavailable (the case in RCS and MSS overpressure criteria are examined) is not affected by an increase in the nominal full-power because the power level assumed in the current analysis for this case (with 2% uncertainty) is equivalent to that based upon the uprated power of 3068 MWt, combined with the lower uncertainty of 0.6% and remain bounding for the 1.4% power uprate. Therefore, the results of the case analyzed without pressurizer pressure control available are unaffected and the turbine trip presents no hazard to the integrity of the RCS or the 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% power uncertainty allowance. Therefore, these analyses are unaffected by the 1.4% power uprate, and the conclusions presented in Sections 14.1.9 and 14.1.12 of the UFSAR remain valid.

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

The rupture of a CRDM housing (RCCA ejection) event is the result of the assumed mechanical failure of a CRDM pressure housing such that the RCS would eject the a RCCA and drive shaft to the fully withdrawn position. The transient responses for the hypothetical RCCA ejection event are analyzed at BOL and EOL, for both full (hot full-power [HFP]) and zero (hot zero-power [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, in order to maximize fuel temperature response).

The current HFP analysis of record is performed at 102% 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% power uprate conditions. However, since these setpoints are fractions of the nominal full-power level, it was necessary to confirm that the event 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% for the HZP and HFP cases, respectively. Since the trip setpoints are not changing for this program, the power level at which the plant 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 effect, 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% (or 118%) of the 1.4% power uprate versus 35% (or 118%) 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. Therefore, this analysis of record is unaffected, and bounds the 1.4% power uprate.

8.3.6 Non-Limiting/Bounded Non-LOCA Events

8.3.6.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical 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 limit value.

By definition, 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 any increase in the reactor full-power level.

Since the power range neutron flux low setpoint of 35% is not changing for the 1.4% power uprate, the power level at which the plant 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 would be 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% of the 1.4% power uprate versus 35% 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.

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% 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% 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, and 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 an initial power level.

A sensitivity study performed by Westinghouse showed that a 1.4% power uprate would increase the reactor trip (due to an OTΔT condition) time by less than one 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. Westinghouse determined that this is applicable to IP3.

An evaluation of the Mode 1 analysis was performed that showed that the 1.4% power uprate does not significantly effect the automatic reactor trip time used in this 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% power uprate.

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

The analysis of record for this event conservatively assumes initial conditions representative of this event, with three loops in operation and a power level equal to 77% of hot full power. However, the IP3 Technical Specifications require that all four 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 only of historical interest and not applicable to current IP3 operation or operation with the 1.4% 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% power uprate. The licensing-basis analysis also addresses cases that are initiated at 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 HZP conditions are unaffected and continue to remain bounding for the 1.4% 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% power uprate, and the conclusions presented in Section 14.1.11 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 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 steamline 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 steamline 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 steamline 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 steamline 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% power uprate. Therefore, the licensing-basis zero-power steamline rupture analysis presented in Section 14.2.5 of the IP3 UFSAR remains valid and bounds the 1.4% 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 IP3 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 CRDM 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% power. A change in the 100% power value does not change the results.

8.3.6.8 **Anticipated Transients Without Scram** (UFSAR Section 14.1.2)

For Westinghouse-designed 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 AMSAC (ATWS Mitigation System Actuation Circuitry). In compliance with 10 CFR 50.62 (b), AMSAC has been installed and implemented at IP3.

As documented in NRC SECY-83-293 ("Amendments to 10 CFR 50 Related to Anticipated Transient Without Scram (ATWS) Events," dated July 19, 1983), the analytical bases for the Final ATWS rule are the generic ATWS analyses for Westinghouse PWRs generated by Westinghouse in 1979. Those generic ATWS analyses were performed based on the guidelines provided in NUREG-0460 ("Anticipated

Transient Without Scram for Light Water Reactors," dated December, 1978), and were formally transmitted to the NRC via letter NS-TMA-2182 ("ATWS Submittal," 12/30/79).

In the generic ATWS analyses documented in NS-TMA-2182, ATWS analyses were performed for the various ANS Condition II events (i.e., Anticipated Transients) considering various Westinghouse PWR configurations applicable at that time. These analyses included two-, three- and four-loop PWRs with various steam generator models. For IP3, the generic ATWS analyses applicable at that time are those for a four-loop PWR with Model 44 steam generators and with 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.

IP3 is currently licensed to a core power of 3025 MWt, and operates with replacement Westinghouse Model 44F steam generators. Hence, the generic ATWS analyses documented in NS-TMA-2182 continue to appropriately reflect the current plant configuration and licensed power level for IP3.

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[8-10]. For the purpose of these AMSAC designs, the generic ATWS analyses for the four-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 IP3, WCAP-10858P-A, AMSAC Logic 2, AMSAC Actuation on Low Main Feedwater Flow, has been employed.

The IP3 1.4% power uprate reflects a power increase of 1.4% above that considered in the generic ATWS analysis for the four-loop PWRs with Model 44 steam generators. As documented in NS-TMA-2182, an increase in core thermal power adversely affects the results of the ATWS analyses. As reported for the generic four-loop PWR with Model 51 steam generators, an increase in power of 2% increases peak RCS pressure by 44 psi in the limiting loss of load ATWS event. As demonstrated in NS-TMA-2182, the peak RCS pressure with the 2% increase in power remains below 3200 psig. This ATWS sensitivity analysis was performed assuming a 2% variation in power, consistent with the typical calorimetric measurement uncertainty allowance on power at the time of these analyses. The proposed increase in core power of 1.4% is within the applicable range of the 2% increase in power assumed in the sensitivity analysis.

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 -8 pcm/°F. A sensitivity analysis including the use of a MTC of -7 pcm/°F was also provided as prescribed by NUREG-0460. In 1979, the MTC values of -8 pcm/°F and -7 pcm/°F represented MTCs that Westinghouse PWRs would be more negative than for 95% and 99% of the cycle, respectively. The base case of 95% represents a 95% confidence limit on favorable MTC for the fuel cycle. For IP3, the Technical Specification requirement on MTC is limited to < 0 pcm/°F at all power levels and, therefore, remains the same as that which was applicable for most Westinghouse PWRs in 1979. Hence, the reactivity feedback for IP3 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 safety valve relief capacity and auxiliary feedwater capacity is unaffected by the proposed 1.4% power uprate as documented in Sections

6.1.1 and 6.2.4, respectively, of this report. The design capacity of each of the three IP3 pressurizer safety relief valves is 420,000 lbm/hr. This is greater than the 408,000 lbm/hr pressurizer safety valve relief capacity assumed in the 1979 generic ATWS analyses for the Westinghouse four-loop plant configuration with Model 44 steam generators as documented in NS-TMA-2182. Therefore, this is a benefit with respect to peak RCS pressure following the pressure limiting ATWS events.

Both of the IP3 pressurizer PORVs have a design relief capacity of 179,000 lbm/hr. Hence, the pressure relief capacities of the PORVs at IP3 are consistent with those modeled in the 1979 generic ATWS analyses for the Westinghouse four-loop plant configuration with Model 44 steam generators as documented in NS-TMA-2182.

For IP3, the design capacities of the AFW pumps are as follows:

- Motor-Driven AFW Pump - 400 gpm
- Turbine-Driven AFW Pump - 800 gpm

The IP3 AFW system has two motor-driven AFW pumps (each pump aligned to two steam generators) and one turbine-driven AFW pump (aligned to all four steam generators). Hence, the total design capacity of the IP3 AFWS is 1600 gpm. This is 160 gpm less than the total AFW system capacity of 1760 gpm assumed in the 1979 generic ATWS analyses for the Westinghouse four-loop plant configuration with Model 44 steam generators as documented in NS-TMA-2182. However, as reported for the generic four-loop PWR with Model 51 steam generators, reducing the auxiliary feedwater (AFW) by 50% (i.e., 880 gpm) would only increase the peak RCS pressure by 31 psi in the limiting loss of load ATWS event.

Based on the above, operation of IP3 with the 1.4% power uprate (core power 3071.4 MWt) remains bounded by the generic Westinghouse ATWS analysis documented in NS-TMA-2182[8-10] and, therefore, remains in compliance with the Final ATWS Rule, 10 CFR 50.62 (b).

8.4 CONTAINMENT INTEGRITY

8.4.1 Steam Line Break Mass and Energy Releases and Containment Integrity (UFSAR Section 14.3.6)

The licensing-basis safety analyses related to the steam line break mass and energy releases and containment integrity were evaluated to determine the effect of the 1.4% power uprate. The evaluation determined that the NSSS design parameters for IP3, as shown in Table 2-1, remain unchanged or bounded by the current safety analysis values for the IP3 mass and energy release calculations. As a result, containment integrity is not affected by steamline break at the 1.4% uprate conditions.

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, auxiliary feedwater 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% power uprate will be offset in this calculation by an equivalent reduction in the calorimetric uncertainty allowance. The current analysis of record applicable for the inside containment long-term steam line breaks already assumes a 2% power calorimetric uncertainty allowance added to the nominal full power. A minimum 0.6% power calorimetric uncertainty applied to a maximum 1.4% power increase, is equivalent to the power level used in the current licensing-basis safety analysis of record. Therefore, the current licensing basis long-term steam line break mass and energy release analysis of record and the steamline break containment integrity analysis of record are unaffected, and bound the 1.4% power uprate.

8.4.2 LOCA Mass and Energy Releases and Containment Integrity
(UFSAR Section 14.3.6)

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% 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 IP3 UFSAR Section 14.3.6 and has been supplemented by evaluations supporting various IP3 plant changes.

The analysis of record presently assumes a core thermal power of 3025 MWt, plus an additional 2% 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% power uprate with a 0.6% power measurement uncertainty allowance, the current analyses of record are unaffected, and bound the 1.4% power uprate.

8.4.3 Short-Term LOCA Mass and Energy Release Analysis

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.

IP3 has been approved for LBB 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% 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. For the 1.4% power uprate conditions, the minimum hot, full-power RCS cold leg temperature may be as much as 3.4°F lower than that for the design basis evaluation. The decrease in minimum RCS cold leg temperature is considered by Westinghouse to be more than offset by the reduction of mass and energy releases from the smaller primary system nozzle breaks relative to the original design basis. Therefore, the current licensing basis for the short-term LOCA subcompartment pressurization analysis is unaffected, and bounds the 1.4% power uprate.

8.5 LONG-TERM MASS AND ENERGY RELEASES OUTSIDE OF CONTAINMENT (EQUIPMENT QUALIFICATION INPUT)

The critical parameters for the long-term steam line break outside containment event include the following conditions on the primary and secondary sides: NSSS power level, RCS coolant temperature, and a conservatively low initial value for the steam generator water mass combined with minimized main and auxiliary feedwater addition. The input assumptions related to these critical parameters dictate the timing and magnitude of the superheated steam that is released from the break.

The 1.4% power uprate will be offset in this calculation by an equivalent reduction in the calorimetric uncertainty. The current analysis of record applicable for the long-term steam line breaks already assumes a 2% power calorimetric uncertainty added to the assumed NSSS power of 3045 MWt (3025 MWt core power plus 20 MWt conservatively assumed for reactor coolant pump heat). A minimum 0.6% power calorimetric uncertainty applied to a maximum 1.4% power increase, is equivalent to the licensing-basis safety analysis of record. Therefore, as long as the sum of the power increase and power calorimetric uncertainty does not exceed 2%, there is no effect on the long-term mass and energy releases outside containment.

8.6 POST-LOCA CONTAINMENT HYDROGEN GENERATION

An evaluation of the hydrogen generation in containment following a LOCA for IP3 has been performed based on parameters that bound the uprate plant conditions. This analysis concludes that without recombination and no contingency on corrodible inventories, a containment concentration of 3 volume percent (v/o) would be reached after 9 days, and a containment concentration of 4 v/o (lower flammable limit) would be reached at approximately 21 days. Assuming the operation of a single recombiner at a capacity of 100 standard cubic feet per minute (scfm) and recombination beginning at 9 days after LOCA, the peak concentration will not exceed 3 v/o.

The 1.4% uprate has no adverse effect on the post-LOCA containment hydrogen generation, since the hydrogen concentration remains below the lower flammability limit specified in Regulatory Guide 1.7 (4 v/o) and the UFSAR statement that "... 3% hydrogen occurs at approximately 10 days" is not compromised.

8.7 STEAM GENERATOR TUBE RUPTURE THERMAL-HYDRAULIC AND RADIOLOGICAL DOSE ANALYSES

The analysis for the steam generator tube rupture (SGTR) event is performed to demonstrate that the offsite radiological consequences remain below the guideline values. The analysis for the offsite dose considers the mass of primary coolant transferred to the ruptured steam generator and the mass of steam released as a result of the event, and the source term. As input to the radiological consequences analysis, a SGTR thermal-hydraulic analysis is performed. The thermal-hydraulic analysis calculates the primary-to-secondary break flow and steam released to the environment. 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 core power up to 3025 MWt, with a 10% increase applied to enable future evaluations without requiring reanalysis. Half of this steam release margin has previously been allocated by Entergy to accommodate past plant evaluations. Since break flow is not sensitive to power, and the steam releases are considered to be proportional to power, the remaining margin will bound the potential effects of the 1.4% power uprate. Furthermore, the current source term for this analysis was developed assuming an initial core power level of 3216 MWt. Therefore, the current analysis of record for the SGTR event is unaffected, and bounds the 1.4% power uprate.

As documented in the IP3 UFSAR, Westinghouse has completed a setpoint relaxation study for the postulated SGTR Emergency Operating Procedures (EOPs), increasing operator response time from 30 minutes to 60 minutes and allowing for the effect of charging flow and the potential for steam generator overfill. The Westinghouse evaluation did not change the formal design basis response time of 30 minutes. It does, however, justify extending the allowable time from 30 to 60 minutes, for operator response in the affected EOPs. Both the associated radiological consequences assessment and the margin to overfill assessment included a 2% power uncertainty allowance. Therefore, those assessments are also unaffected, and bound the 1.4% power uprate.

8.8 ACCIDENT ANALYSES RADIOLOGICAL CONSEQUENCES

An evaluation was performed to determine the potential effects of the 1.4% power uprate.

Analyses performed using a nominal core power of 3025 MWt would be unaffected since a 2% uncertainty allowance was also included in those calculations. All other IP3 radiological dose analyses were performed using a core power of 3216 MWt. Therefore, the current IP3 radiological analyses of record are unaffected, and bound the 1.4% power uprate.

The IP3 radiological dose analyses are:

- Fuel Handling Accident (UFSAR Chapter 14.2.1)
- Waste Liquid Tank Failure (UFSAR Chapter 14.2.2)
- Waste Gas System Failure (UFSAR Chapter 14.2.3)
- Steam Generator Tube Rupture Accident (UFSAR Chapter 14.2.4)
- Main Steamline Break (UFSAR Chapter 14.2.5)
- Loss-of-Coolant-Accident (LOCA) (UFSAR Chapter 14.3.5)

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 3067.4 MWt with a 0.6% power measurement uncertainty. Also, the RCS and secondary-side parameters used in the analyses bound those for the 1.4% power uprate (Table 2-1). The steam release rates to the environment were also evaluated and found to bound those for the 1.4% power uprate.

Therefore, the current IP3 radiological dose analyses of record for the UFSAR Chapter 14 accidents above are unaffected and bound the 1.4% power uprate.

8.9 SECTION 8 REFERENCES

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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.

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8-7. Westinghouse Topical Report WCAP-11397-A (proprietary) and WCAP-11398 (non-proprietary), "Revised Thermal Design Procedure," A. J. Friedland and S. Ray, April 1989.

8-8. Westinghouse Topical Report WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions," September 1986.

8-9. Westinghouse Topical Report WCAP-11394-P-A, "Methodology for the Analysis of the Dropped Rod Event," R. L. Haessler, et al., January 1990.

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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% 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% power uprate.

9.1 ELECTRICAL DISTRIBUTION SYSTEM

The station electrical distribution system is comprised of the connections to the Main Transformers (MTs), the Unit Auxiliary Transformer (UAT) and the Station Auxiliary Transformer (SAT), the medium voltage buses and low voltage buses and associated transformers and interconnections and the 120V Vital AC and the DC Distribution System.

9.1.1 DC Systems

The IP3 site electrical power supplies include the Cat. I battery system. This system consists of four independent and physically separated buses. One available battery charger and one battery energize each bus.

The DC system is not affected by the 1.4% power uprate since no loads were added to the system.

9.1.2 AC Systems

Voltage Profile

The electrical changes resulting from the 1.4% power uprate occur for the BOP equipment at the medium voltage level. The effects of these load changes are minimal, 0.2% or less for buses 1, 2, 3 and 4. The load increases are within the capability of the medium voltage buses. The changes in bus loading will reflect in revised bus voltages, either decreases or increases. The magnitude of the changes is so minimal that they will not affect the existing station voltage profile.

The loading changes occur at the medium voltage buses 1, 2, 3 and 4. No load changes occur on buses 5 and 6. The degraded voltage relays monitor voltage at the low voltage bus level. The relays monitor the voltage on safety buses 2A, 3A, 5A and 6A. There were no load increases on these low voltage buses. The load increase for the medium voltage buses feeding bus 2A, 3A, 5A and 6A is minimal at the 6.9 kV level, and will not have an effect at the low voltage level. Therefore, the station auxiliary electrical distribution systems will remain within acceptable limits.

Station Service Fault Analysis

The available fault current at the 6,900 V switchgear will not increase for the 1.4% power uprate condition. The medium voltage motors affected by a 1.4% power uprate will remain the same size, and no additional motor loads will be added. The 1.4% 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: 15 kV for the medium voltage switchgear, and; the 600 volt for the low voltage switchgear.

The 15 kV Non-Segregated Phase Bus Duct connects the UAT to 6,900 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 bus is composed of two 2000 amp buses). The segments that run from Switchgear Bus 4 to Switchgear Bus 2, and from Switchgear 5 to Switchgear Bus 6 have a continuous rating of 2000 amps. The remaining segments are 1200 amps per phase, 15 kV.

The 600 volt Non-Segregated Phase Bus Ducts connect the emergency diesel generators (EDG 31, EDG 32 and EDG 33) to respective 480 V switchgear buses 31 and 32. These buses are rated 3000 amps. The bus connecting 480 V switchgear buses 31 and 32 has a continuous rating of 3000 amps per phase, 600 V.

The non-segregated bus rated capacity exceeds the UAT and SAT capacity. The other non-segregated buses are sized to the same rating as the bus ratings they feed. A review of the Load Flow Calculation showed that the non-segregated bus currents are less then the rating and remain less with the additional load added by the 1.4% power uprate. Therefore, the medium voltage non-segregated buses remain acceptable.

No additional load will be added to the low voltage buses.

9.1.4 Station Service Transformers

Four Station Service Transformers (SSTs) (2, 3, 5 and 6) supply power to the 480 V safety 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) at 150°C, three-phase, 60 Hz.

The two SSTs (312 and 313) that supply power to the 480 V non-safety switchgear buses 312 and 313 each have a nameplate rating of 6,900 V primary and 480/277 kV secondary, and 2000/2666 kVA AA/FA (self-cooled/forced-air cooled) at 150°C, three-phase, 60 Hz.

The increase in loading resulting from the 1.4% power uprate occurs at the medium voltage level. There is no significant 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 Isophase Bus Duct, the MTs, the UAT and the SAT, 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 1,125.6 MVa (based on 75 psig hydrogen pressure), 0.90 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 current operating point of the main generator is 1,027.2 MWe, taken for the current station heat balance. This operating point is at a power factor of 91.26. At the 1.4% power uprate, the main generator is anticipated to operate at an output of 1,041.2 MWe, based on the baseline heat balance for 101.4% of the current core thermal power. The generator capability curves show this operation is possible at a power factor of 92.54%. This operating point will allow production of up to 427.6 MVAR lagging. This operating point also allows acceptance of 170 MVARs leading. The leading limit is a result of instability limits shown on the generator capability curve.

The cooling systems consist of stator water cooling and generator hydrogen cooling equipment, each sized to support the generator at nameplate rating. The generator lubrication systems provide lubrication to the bearings, and the seal oil systems maintain the generator atmosphere isolated from the outside ambient. These systems are supported by low voltage AC and DC components. These systems are sized to support the nameplate rating of the machine. Operation of the machine within the capability curve will not require additional capacity.

The exciter has the capability to support machine operation within its nameplate rating and within the capability curve of the machine for the leading and lagging case of VAR production.

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% 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% power uprate conditions because the machine will be operated within its rated capability. A Voltage Balance relay protects against a blown Potential Transformer (PT) fuse.

9.2.2 Isolated Phase Bus Duct

The Isolated Phase Bus Duct (Iso-Phase Bus) 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 two segments that run to the two MTs. This first segment has a forced air-cooled rating of 32 kA at 23 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 23 kV, 65C. The third segment runs from the main bus tap to the UAT. This segment has a self-cooled rating of 1.5 kA at 23 kV. This segment does not have a forced cooled rating.

The generator operating at its nameplate rating, 1125.6 MVa at 22 kV, produces 29.54 kA. Industry standards allow generator operation at nameplate rating with a voltage range of ± 5%. At full load and 0.95 per unit (PU) voltage, the generator can produce 31.09 kA. The two MTs have identical MVa ratings but different impedances. The current splits between the transformers unevenly in proportion to the impedance ratings. The currents through the segments of the Iso-Phase Bus between the split and the MTs are within the bus rating for the generator at full load for a generator voltage of 0.95 PU.

The tap to the UAT is rated at 1.5 kA. This is a self-cooled rating. The tap is capable of carrying the full load rating of the transformer, 1.13 kA or with the generator voltage at 0.95 PU, 1.188 kA.

9.2.3 Transformers

Main Transformers

The main generator delivers its power output to two MTs (31 and 32). MT 31 is a Westinghouse transformer, nameplate rating 20.3/345 kV, 542 MVa forced-oil-air cooled (FOA) at 55°C, three-phase, 60 Hz. MT 32 is a General Electric transformer, nameplate rating 20.3/345 kV, 325/433/542 MVa OA/FOA/FOA at 55°C, and 607 MVa FOA at 65°, three-phase, 60 Hz.

The impedance for transformer MT 31 is 16.37% and the impedance for MT 32 is 16.19% at the 55°C rating. Therefore, the load will divide between the transformers with MT 31 taking approximately 49.72% and MT 32 taking approximately 50.28%.

The total capacity of the MT bank is 1084 MVa at the 55°C rating. Allowing for the load of the UAT, the required capacity of the MTs is 1125.6 – 43 MVa = 1082.6 MVa. MT 32 will slightly exceed its 55°C rating based on the division of flow between the MTs. Therefore, MT 32 will operate within its 65°C rating. MT 31 will operate within its 55°C rating. Therefore, the Main Transformers remain acceptable.

Unit Auxiliary Transformer

The UAT nameplate rating is 22/6.9 kV, 43MVa FOA at 55°C, three-phase, 60 Hz. The transformer is equipped with a +10/-5% load tap changer. The UAT supplies power to BOP systems under normal operating conditions.

The BOP systems are those systems most affected by the 1.4% power uprate. The BOP systems most affected are the Feedwater System, the Condensate System and the Heater Drains System. The analysis of these systems at the 1.4% power uprate level produced new pump operating points. The feedwater pumps have non-electrical drivers so their new operating point does not effect the station electrical distribution system. The operating point of the heater drains pumps changed from 940 bhp to 930 bhp. The operating point of the condensate pumps also changed from 2575 bhp to 2600 bhp. This results in a net electrical load increase of approximately 55 hp based on operation with three condensate pumps and two heater drains pumps.

The station performance database, based on operation's logs, were reviewed to determine the house load with the unit operating at various power levels, including full power. The review encompasses the 1992 to 2001 time frame. This review showed that the house load historically is approximately 34 to 36 MWe with a maximum of 37.5 MWe. The increase in house load resulting from the change in operating points for the affected BOP pumps will increase house load by about 41 kWe, or about 0.11%. Based on the information on the house load in the performance database and the predicted increase in house loads resulting from the 1.4% power uprate, the UAT is acceptable.

Station Auxiliary Transformer

The SAT nameplate rating is 138/6.9 kV, 43MVa FOA at 55°C, three-phase, 60 Hz. The transformer is equipped with a +10/-5% load tap changer. The SAT provides power to NSSS and BOP systems under abnormal operating conditions.

The BOP systems most affected by power uprate 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 feedwater pumps have non-electrical drivers so their new operating point does not effect the station electrical distribution system. The operating point of the heater drains pumps changed from 940 bhp to 930 bhp. The operating point of the condensate pumps also changed from 2575 bhp to 2600 bhp. This results in a net electrical load increase of approximately 55 hp based on operation with three condensate pumps and two heater drain pumps.

The station performance database, based on operations logs, was reviewed to determine the house load with the unit operating at various power levels, including full power. The review encompasses the 1992 to 2001 time frame. This review showed that the house load historically is approximately 34 MWe to 36 MWe with a maximum of 37.5 MWe. The increase in house load resulting from the change in operating points for the BOP pumps will increase house load by about 41 kWe, or about 0.11%.

Should a fault condition occur that interrupts power to the plant auxiliary loads, these loads will be energized by the SAT. Based on the information on the house load in the performance database, and the predicted increase in house loads resulting from the 1.4% power uprate, the SAT is acceptable.

Transformer Protection

A review of One-Line Diagrams and Protective Relay Settings indicates that the Main Transformers, UAT and SAT protection essentially consists of high-speed phase fault protection and ground fault protection.

The MTs are protected by a differential relay scheme. Back-up protection is provided by the Unit Differential Relay and the MT 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 to the low voltage winding. Back-up protection is provided by instantaneous and time over-current relays on each phase

The SAT is protected by a differential relay scheme. Back-up protection is provided by single-phase instantaneous and time over-current relays and a neutral time over-current relay protection scheme.

The relay protection schemes for the transformers (MTs, UAT and SAT) are not affected by load changes. Therefore the relay protection schemes for these transformers remain acceptable for the 1.4% power uprate.

9.2.4 Switchyard

The 345 kV breakers and switches installed in the 345 kV switchyard are rated 3,000 amps which exceeds the main generator output current of approximately 1,700 amps at its nameplate rating of 1,026 MVa ([(1,026 MVa x 1000) / $\sqrt{3}$ x (345 kV)]). Therefore, the breakers and switches will accept the additional load without the need of any hardware modification.

IP3 receives shutdown power from two physically independent and redundant offsite power sources: one from the 138 kV switchyard, the other from the 13.8 kV gas turbine substation. Under the 1.4% power uprate, there is no change in the shutdown (ESF) loads, and bus voltage values at different levels of the station auxiliary distribution systems are bounded by the existing load flow and voltage profile analyses. The extra 1.4% increase in power generated into the 345 kV System has no significant effect on the 138 kV switchyard system and the ability of the units to safely shut down. Therefore, IP3 is expected to remain in compliance with GDC-17 for the 1.4% power uprate.

Overhead Lead Protection

MT overhead leads to the 345 kV switchyard are protected by primary and back-up 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 back-up 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% power uprate conditions.

9.2.5 Grid Stability

A grid stability study was performed in January 2001. This study showed the unit was stable. It is expected that this study will remain valid for the 1.4% 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% power uprate. The load changes only occur to medium voltage motors and, 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% 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 are typically motors, cables and circuit breakers.

System evaluations have determined that some medium voltage motors on non-safety-related 6,900 V switchgear have revised operating points. The condensate pumps, rated at 3000 hp each, experience a brake horsepower increase from 2575 to 2600 bhp. The heater drains pumps, rated at 1000 hp each, experience a brake horsepower decrease from 940 bhp to 930 bhp. The existing motor drives will operate at a brake horsepower less than the design rating during full load conditions at the 1.4% power uprate, so no motor replacements will be required at the 1.4% power uprate. Normal design practice is to size the motor cables to the nameplate rating of the motor and to also allow for motor service factor. The brake horsepower of the condensate and heater drains pumps remain below the nameplate rating. Therefore, the cables are acceptable.

A review of One-Line Diagrams and Protective Relay Settings indicates that several different schemes are used to provide medium-voltage (6,900 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, 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% 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% uprate conditions will not affect the existing medium-voltage motors and associated protection schemes.

10 BALANCE OF PLANT

The IP3 BOP systems were reviewed for potential effects due to the 1.4% power uprate to 3067.4 MWt reactor core power. The BOP systems that could potentially be affected due to the 1.4% power uprate are the:

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

The IP3 BOP systems were evaluated based on actual heat balance data at current power conditions, and updated to reflect operating conditions expected for the 1.4% power uprate.

10.1 MAIN STEAM AND STEAM DUMP SYSTEM

Based on the original design and the expected 1.4% uprate condition heat balance models, main steam flow at uprate conditions is bound by the original design maximum guaranteed steam flow. The steam generator outlet pressure is anticipated to drop from its current 783 psia pressure at 100% power to approximately 778 psia. The saturation steam temperature will be reduced from 515.7°F to approximately 515.0°F. These pressure and temperature changes are bounded by the design pressure and temperature values.

The major components of the MSS include the main steam safety valves, atmospheric dump valves, main steam isolation valves, high pressure steam dump valves, and turbine overspeed and low pressure steam dump valves. These components were evaluated for the proposed uprate conditions and are discussed below. The zero-power load conditions do not change and remain bounding for the main steam design pressure and temperature.

10.2 CONDENSATE AND MAIN FEEDWATER SYSTEMS

The C&FS automatically maintains steam generator water levels during steady state and transient operations. The Main Feedwater System also automatically isolates C&FS from the steam generators, when required, in order to mitigate the consequences of an accident. The revised 1.4% power uprate operating conditions will affect both the feedwater volumetric flow and system pressure drop. However, in all cases, the results of the evaluations are that the respective system design basis do not affect system operation and remain valid.

The C&FS include the condensate pumps, condensate booster 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. There are three condensate pumps normally running for full load, three condensate booster pumps (no pumps are normally running), two heater drain pumps normally running at full load, and two steam-powered MFPs normally running at full load. One condensate booster pump is normally in standby and started on low MFP suction pressure signal to ensure adequate MFP suction pressure. 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% of full-power load rejection transient. The result of the 1.4% power uprate will be to increase the amount of feedwater supplied to the steam generators at full-load by approximately 1.6%.

C&FS pressure and temperatures are evaluated for uprate conditions. This review is based on current calculations. These calculations were based on the condensate polishing facility being on line and condensate booster pumps running. The results indicate that the system pressure and temperatures continue to be bound by the current design.

Due to the flexibility of condensate polishing system, IP3 may operate with the condensate flow passing through empty resin vessels, post filters and condensate booster pumps secured. For this mode of operation, system pressures at 1.4% power uprate conditions would be slightly less than those at the current power due to the minor increase in frictional losses and reduction in condensate pump head. The MFP suction pressure was evaluated because reductions in this pressure could affect MFP NPSHa and possible instrument settings. Operational data was used to establish the current MFP suction pressure for this mode of operations. Current data indicated the MFP suction pressure is approximately 408 psia. This review indicated that at 1.4% power uprate conditions the MFP suction pressure would decrease approximately 8 psi to approximately 400 psia for the 1.4% power uprate conditions. This pressure is still above the minimum required for adequate NPSH.

The feedwater heater vents and relief valves are also included in this evaluation and all are acceptable for the 1.4% power uprate operating conditions.

An evaluation of the C&FS has not identified any limitations in the existing design that would preclude the 1.4% power uprate. Component parameters are bounded by original design equipment ratings, 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% 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% power uprate. Component parameters are bounded by original design equipment ratings, or by the original design considerations for off-normal operation.

10.5 FEEDWATER HEATERS AND DRAINS

The feedwater heaters were reviewed based on design information, current operating conditions, and expected uprate conditions. The heaters are either operating within design or have no significant parameter changes associated with the uprate. Therefore, all feedwater heaters are determined to be acceptable for the 1.4% power uprate. 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% power uprate operation.

For the reheater drains, the required control valve flow under 1.4% power uprate conditions is similar to the current flow. Based on the current operating positions, no normal control valve capacity limitations have been identified. The moisture separator drain tank control valves were also determined to be adequate for 1.4% power uprate operation. 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

The SWS provides cooling water directly to components, and to the following intermediate cooling water systems:

- Instrument Air Closed Cooling Water
- Turbine Hall Closed Cooling Water
- Component Cooling Water
- Diesel Generator Jacket Water Cooling

The increased decay heat and turbine auxiliaries cooling loads will have a small effect on the cooling water temperature increase, however does not impact the required cooling water flow rates. The evaluation of the SWS has not identified any limitations in the existing design that would preclude the 1.4% 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 CCWS has not identified any limitations in the existing design that would preclude the 1.4% power uprate. Because the accident analyses are performed assuming a core thermal power level of at least 102% of the rated value, the CCWS 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 Containment Cooling and Filtration Systems

CC&F systems are provided to remove heat from operating equipment and piping, remove energy from the containment atmosphere after an accident, remove fission products from the containment atmosphere after an accident, purge the containment, reduce pressure buildup in the containment during normal operation. The following CC&F systems are evaluated:

- Containment Air Recirculation Cooling and Filtration System
- CRDM Cooling System
- Containment Pressure Relief System
- Containment Purge System
- Containment Auxiliary Charcoal Filter System

An evaluation of the CC&F systems has not identified any limitations in the existing design that would preclude the 1.4% power uprate. The required post-accident functions are analyzed for at least 102% of the current RTP. These analyses bound the 1.4% power uprate conditions. Component parameters are bounded by original design equipment ratings, or by the original design considerations for off-normal operation.

10.7.2 Other Heating, Ventilation and Air Conditioning Systems

The following major plant areas, outside the Containment Building, are served by HVAC systems:

- Auxiliary Feedwater Pump Building
- Control Room
- Control Building
- Turbine Building
- Electrical Penetration Tunnels

- Emergency Diesel Generator Building
- Fuel Storage Building
- Power Conversion Equipment Building
- Shield Wall Enclosure
- Primary Auxiliary Building
- Hot Penetration Cooling

An evaluation of the above HVAC Systems has not identified any limitations in the existing design that would preclude the 1.4% power uprate. The required post-accident functions are analyzed for at least 102% of the current rated core thermal power. These analyses bound the 1.4% power uprate conditions. Operation at the 1.4% power uprate conditions will not affect the operation of any of the HVAC Systems.

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 I&C system design are required or recommended to support the 1.4% power uprate. The small increase (~1.6%) in BOP flows are within the plant instrumentation ranges or setpoints and no changes have been identified due to power uprate. The existing I&C system design will not be affected when the plant operates at the 1.4% power uprate conditions.

10.9 PIPING AND SUPPORT EVALUATION

The BOP 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% power uprate.

The piping system evaluations performed conclude that all piping systems remain acceptable and will continue to satisfy design basis requirements when considering the temperature, pressure and flow rate effects resulting from the 1.4% power uprate conditions.

The fluid systems that experienced a change in energy (pressures, temperatures, or flow rates) due to the 1.4% power uprate were reviewed to evaluate the effect of a pipe break or crack. These systems include the main steam system, the feedwater system, and the steam generator blowdown and cleanup system. In addition, the impact of the blowdown from the RCS through the piping attached to the RCS was reviewed to determine the impact of revised NSSS temperatures.

The 1.4% power uprate will have a negligible effect on the pipe stress. Therefore, no pipe stress re-analysis was required for the 1.4% power uprate.

The 1.4% power uprate does not result in any new or revised pipe break locations.

10.10 SPENT FUEL POOL COOLING SYSTEM

The design basis of the SFPCS includes the capability to maintain the SFP below 150°F following the discharge of 76 fuel assemblies, and below 200°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. Fuel movement is prohibited until the reactor has been sub-critical for at least 145 hours. In addition, no more than 76 assemblies may be discharged until the reactor has been sub-critical for at least 267 hours, unless acceptable movement with a shorter time delay is verified by a case-specific calculation.

10.10.1 Uprate Evaluations

Spent Fuel Pool Operating Temperatures

The Thermal-Hydraulic Report and the verification calculation were reviewed for current decay heat values, design information and to determine the calculations developed to support the current UFSAR Section 9.3 information. These values where then used to support uprate analysis and recommended UFSAR revisions.

The Thermal-Hydraulic Report determined the design case decay heat values for the 76 fuel assembly off load, 145 hours after reactor shutdown to be 17.48 MBtu/hr. Also this report determined the design case full core off-load decay heat, 267 hours after reactor shutdown to be 35 MBtu/hr.

Subsequent to the Thermal-Hydraulic Report, re-analysis of the decay heat values was performed to incorporate increased fuel enrichment data, 24-month fuel cycle and 95°F river water temperature. This calculation established the 76 fuel assembly, 145 hours after reactor shutdown decay heat value to 17.38 MBtu/hr. This is slightly less than the design value, stated in the Thermal Hydraulic Report, of 17.48 MBtu/hr. Also the full core, 267 hours after reactor shut down case was re-analyzed to determine a decay heat value of 30.32 MBtu/hr. This is also below the design value of 35 MBtu/hr.

As the calculated decay heat values are based on more recent analysis that more accurately represents current plant operation and conditions, these decay heat values were used as input for the uprate evaluations. The calculation that determines the SFP maximum temperatures and time to boil values for uprate conditions scaled the current values (17.38 MBtu/hr and 30.32 MBtu/hr for the two off-load cases) to determine expected uprate decay heat values. The results of this calculation formed the basis for the expected uprate conditions discussed below, as well as, determined the recommended value changes to the current UFSAR Section 9.3.

The design basis of the SFPCS includes the capability to maintain the SFP below 150°F following the discharge of 76 fuel assemblies, and below 200°F following the discharge of a full core. For a 1/3-core fuel assembly discharge after 145 hours, IP3 UFSAR Table 9.3-3 states that the SFP heat load is 17 MBtu/hr. This is below the current and expected uprate decay heat for the 76 fuel assembly discharge scenarios. (This difference in decay heat is do to differences in the "1/3 core" and "76 fuel assembly" conditions.) However, the analysis performed in support of the uprate determined that, at 1.4% power uprate conditions, the SFP will remain below 150°F for this off-load scenario.

The current analysis decay heat value (17.38 MBtu/hr) was used as input, and scaled to reflect uprate conditions, for the calculation that determined the 1.4% uprate maximum SFP temperature and time to boil for loss of cooling. This calculation determined that the expected uprate decay for the 76 fuel assembly off-load is 17.73 MBtu/hr. This resulted in a maximum SFP temperature of 148.9°F. This is below the 150°F limit, and therefore, acceptable for uprate. The expected uprate decay heat load for this case (17.73 MBtu/hr) is also greater then SFP design value (17.48 MBtu/hr). However, acceptable pool temperatures are maintained at the 1.4% power uprate conditions. Therefore, this higher value is acceptable.

For a full core discharge the current decay heat value of 30.32 MBtu/hr was scaled to reflect the 1.4% power uprate conditions. The uprate decay heat value for the full core off-load was determined to be 30.93 MBtu/hr. This is bounded by the design value of 35 MBtu/hr. This 1.4% power uprate condition calculation determined the maximum SFP temperature of 184.6°F for the full core off-load. This is below the design maximum temperature for this scenario of 200°F, and is therefore acceptable for the 1.4% power uprate.

Loss of Cooling SFP Heat-Up Times

The decay heat values for uprate conditions were evaluated to determine the time to boil for the SFP in the event of a loss of SFP cooling. In this scenario the SFP would be heated by the residual heat from the stored spent fuel elements. The design basis of the SFPCS includes the capability to maintain the SFP below 150°F following the discharge of 76 fuel assemblies, and below 200°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. The UFSAR Section 9.3 states that for the 76 fuel assemblies discharge case, it would take 8.5 hours for the SFP to heat from 150°F to boiling. For the full core discharge case, it would take 49.2 min. for the SFP to heat from 200°F to boiling.

To determine the SFP heat-up times for 1.4% power uprated operation the calculated maximum SFP heat loads for current and 1.4% power uprate conditions are compared with the UFSAR heat load and heat-up rate values to determine the expected SFP heat-up rate and the time to reach boiling.

The minimum time for the SFP to boil is calculated to be 55.7 min. for the 1.4% power uprate full-core discharge case. This value is bounded by the UFSAR value of 49.2 minutes which was based on a conservatively higher decay heat load. For the 76 fuel assemblies discharge case, the time for the SFP to boil is calculated to be 8.4 hours, which is a slight decrease from the UFSAR value of 8.5 hours. This decrease in time is not significant compared to the time available to take action. In addition, the heat loads used to develop this time are very conservative. Therefore, minimum times for the SFP to boil are acceptable for the 1.4% power uprate operating conditions.

10.10.2 Conclusions

An evaluation of the SFPCS has not identified any limitations in the existing design that would preclude the 1.4% power uprate. Component parameters are bounded by original design equipment ratings, or by the original design considerations for off-normal operation. However, UFSAR Section 9.3 includes specific heat load, heat-up rate and heat-up time values that are being revised as a result of the 1.4% power uprate.

10.11 MAIN TURBINE

The main turbine (ABB/ALSTOM low pressure units and Siemens-Westinghouse high pressure unit) have been evaluated and found acceptable for service at the 1.4% power uprate conditions. The high pressure unit will be at or near its maximum capability to pass the steam flow depending on the actual steam pressure achieved at the uprated power.

11 OTHER RADIOLOGICAL EVALUATIONS

The potential radiological effects of the 1.4% power uprate is evaluated for the following:

- Normal Operation Shielding and Personnel Exposure
- Normal Operation Annual Radwaste Effluent Releases
- Radiological Environmental Qualification (EQ)
- Post-LOCA Access to Vital Areas including Technical Support Center Habitability

Radiological evaluations for accident related issues are assessed at a core power level of 3086 MWt (100.6% of 3067.4 MWt) to include a margin of 0.6% 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% as recommended in Regulatory Guide 1.49, Revision 1, to 0.6%.

Except as noted, radiological evaluations for normal operation related issues are assessed, for the 1.4% power uprate, at a core power level of 3067.4 MWt. The normal operation radwaste effluent assessment is based on an assumed core power level of 3086 MWt to reflect current regulatory guidance relative to power levels to be used for 10 CFR 50 Appendix I assessments.

11.1 NORMAL OPERATION ANALYSES

11.1.1 Radiation Source Terms

The current licensed core power level for IP3 is 3025 MWt. The 1.4% power uprate will increase the isotopic inventory in the core by approximately the percentage of the uprate, i.e. 1.4%. 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 1.4% power uprate.

11.1.2 Normal Operation Shielding and Personnel Exposure

The 1.4% increase in expected radiation levels will not affect radiation zoning or shielding requirements in the various areas of the plant. As noted in UFSAR Section 11.2.2 and Tables 11.2-1 through 11.2-6, IP3 shielding design is based on a core power level of 3216 MWt and a design basis RCS with 1% fuel defects, and thus encompasses plant operation at the 1.4% power uprate.

Individual worker exposures will be maintained within acceptable limits by the site 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% 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.

- **Liquid Radioactive Waste**: As the activity levels in the reactor coolant or secondary fluids increase, the activity level 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 thereby minimizing 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 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% 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 NRC in support of IP3 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 3086 MWt (i.e. 3067.4 MWt plus a 0.6% power level uncertainty). Note that, the actual release concentrations and offsite doses are controlled by the IP3 Offsite Dose Calculation Manual, which assures compliance with the IP3 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 power uprate, the volume of solid waste would not be expected to increase proportionally, since the 1.4% 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% 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 3067.4 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.

The IP3 Containment and Primary Auxiliary Building are divided into various environmental zones. All other buildings are considered mild environments and are not included in the scope of the EQ Program. 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 one-year period following a LOCA. With the exception of equipment that fall under Division of Operating Reactors (DOR) Guidelines, a 10% margin is applied to the accident contribution in accordance with the requirements of IEEE-323.

The 1.4% 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 IP3 is based on survey data and will therefore also increase by the percentage of 1.4% power uprate.

The post-accident environmental dose estimates in the containment are based on a LOCA and a power level of 3280 MWt, which encompasses operation at the 1.4% power uprate conditions. As part of the 1.4% power uprate assessment, the analysis of record for the post-accident environmental dose estimates in the Primary Auxiliary Building will be updated to reflect a power level of 3086 MWt.

A comparison of the "specification" versus "qualification" doses associated with the safety-related components in the IP3 EQ Program indicates that there is sufficient available margin to accommodate a 1.4% 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 IP3 to demonstrate compliance with the requirements of NUREG 0578, Item 2.1.6.b and NUREG 0737, II.B.2, included radiation dose rate maps versus time that covered areas and access paths which may require occupancy during post-LOCA recovery operations. In addition, operator doses while performing vital functions post LOCA were estimated to be within the allowable limit of 5 rem.

Core power uprate will increase the activity level in the core by the percentage of the 1.4% power 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 1.4% power uprate assessment, the analyses of record, which are based on a power level of 3025 MWt, will be updated to reflect a core power level of ~3086 MWt (i.e. 3067.4 MWt, plus a 0.6% power level uncertainty) and to demonstrate continued compliance with the 5 rem operator exposure dose limits of NUREG 0737.

The original dose rate mapping will remain representative for the 1.4% power uprate. The simplifying/ conservative assumptions used to develop the radiation dose rate maps and the tolerance expected in zone boundaries, which are intended to cover dose rate in increments of a decade (e.g., zones are defined as within 1E2 R/hr - 1E3 R/hr, 1E3 R/hr – 1E4 R/hr, etc.), ensure that any uprate-related variance from existing calculated values will be insignificant.

The post-LOCA habitability of the Technical Support Center (TSC) is assessed based on a core power level of 3216 MWt, which encompasses plant operation at a core power level of ~3086 MWt (i.e. 3067.4 MWt, plus a 0.6% power level uncertainty).

12 MISCELLANEOUS EVALUATIONS

12.1 PLANT OPERATIONS

12.1.1 Procedures

Plant procedures will not require significant changes for the 1.4% power uprate. Procedural limitations on power operation due to BOP equipment unavailability will be revised as necessary to account for the increase in core power to 3067.4 MWt. Changes associated with the 1.4% power uprating will be treated in a manner consistent with any other plant modification.

Procedures required for the operation and maintenance of the Caldon LEFM system have been implemented and will be revised as necessary to reflect installation of the Check System.

Specific operator actions to be taken when the Caldon LEFM system is inoperable are discussed in Section 3.3 and will be addressed in procedural guidance.

12.1.2 Operator Actions and Training

Engineered Safety Feature System design and setpoints, and procedural requirements already bound the proposed uprating. The responses of the reactor operators to any event will be unaffected by a change in rated thermal power.

There will be minimal effect on alarms, controls and displays for a 1.4% power uprate. The Caldon LEFM system will have alarms 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% power uprate is implemented, the nuclear instrumentation system will be adjusted to indicate the new 100% RTP in accordance with Technical Specification requirements and plant administrative controls. Since the 1.4% power uprate is predicated on the availability of the Caldon LEFM system, procedural guidance will be implemented to facilitate operation when the Caldon LEFM system is unavailable. The reactor operators will be trained on the changes in a manner consistent with any other design modification.

The 1.4% power uprate will be reflected in the plant simulator.

12.1.3 Qualified Safety Parameter Display System

Process parameter scaling changes will be made, as required, to the Qualified Safety Parameter Display System (QSPDS). There are no other effects to the QSPDS from the 1.4% power uprate.

12.2 PLANT PROGRAMS

12.2.1 10 CFR 50, Appendix R

The Emergency Lighting and RCP Oil Collection sections are not affected by the 1.4% power uprate. The Safe Shutdown Capability is discussed below.

In accordance with the IP3 Appendix R Fire Protection Report, the RHRS must also be capable of achieving RCS cold shutdown (below 200°F) in less than 72 hours after reactor shutdown. This was addressed for the 1.4% power uprate in Section 6.1.3.

For a postulated fire with a loss of off-site power, the EDGs are the preferred source of AC power for the safe shutdown systems. An Appendix R diesel generator provides an additional permanently installed alternative AC power supply as part of an enhanced alternate shutdown capability. The Appendix R diesel generator loads are not affected by the 1.4% power uprate.

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% power uprate that would affect the Appendix R requirements. Therefore, the Appendix R program is not affected by the 1.4% power uprate.

12.2.2 Environmental Qualification

10 CFR Part 50.49 requires that nuclear power plants maintain an environmental qualification program that addresses all design basis events (DBEs). 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 IP3 UFSAR as follows:

- Loss-of-Coolant Accident
- High Energy Line Breaks (HELB)
- Main Steam Line Breaks (MSLB)

The current accident analysis for these conditions bound the 1.4% power uprate conditions. Therefore, the accident analysis component of the EQ program will not be affected by the 1.4% 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% power uprate will not significantly alter any normal or AOO conditions or environmental evaluations. Therefore, the 1.4% power uprate will not affect any normal operation aspects of the EQ program.

The 1.4% power uprate does not affect seismic aspects of the plant design and therefore there is no effect on the IP3 Seismic Qualification program.

See Section 11 for evaluation of the radiological aspects of Equipment Qualification.

12.2.3 Station Blackout

A Station Blackout (SBO) is defined as the complete loss of alternating current electric power to the essential and nonessential switchgear buses. (Loss of offsite electric power system, concurrent with a turbine trip and the unavailability of the onsite emergency AC power system.)

Additionally, the Appendix R diesel generator provides an additional permanently installed alternative AC power supply as part of an enhanced alternate shutdown capability. Refer to the Appendix R program discussion, Section 13.1, for addition details on the Appendix R diesel generator.

The methodology and assumptions associated with the SBO analysis with regard to equipment operability are unchanged with 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% power uprate.

Systems associated with SBO affected by uprate include Auxiliary Feedwater and Condensate Storage (Refer to Section 6.2.4).

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% power uprate.

12.2.4 Flow-Accelerated Corrosion

IP3 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% power uprate results in changes in the operating pressure, temperature and velocities in several of the BOP systems. Therefore, the FAC Program was determined to be affected by the 1.4% 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 1.4% power 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% power uprate.

Susceptible safety-related and non-safety-related systems are modeled at IP3 using EPRI's CHECWORKS software. IP3 CHECWORKS models will be revised to incorporate flow and process system conditions that are determined for the 1.4% 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 NRC Generic Letters 89-10 and 96-05 regarding motor operated valve thrust and torque requirements calculations and discussed in NRC Generic Letter 95-07 regarding motor-operated valve 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. RCS pressure and temperature limits during RHRS operations
6. Pressure/ temperature calculations for various accident scenarios

A review of (1) through (6) listed above concluded that the 1.4% 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% power uprate since these calculations used conservative inputs that bound the inputs for the uprate. Therefore, the 1.4% power uprate will not affect the motor operated valve calculations discussed in the NRC Generic Letters 89-10, 95-07 or 96-05.

The IP3 motor operated valve (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 1.4% power uprate affects 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% power uprate does not affect the MOV program.

12.2.6 Probabilistic Safety Assessment Results

The proposed power uprate has the potential to affect several areas in the IP3 Probabilistic Risk Analysis (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% 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., station black-out, room heat-up, steam generator boil-off, etc,) performed to support alternative criteria. Calculated system success criteria will be re-evaluated to ensure criteria assumed in the PRA are still valid. Based on available system margins, no changes in system success criteria are expected as a result of the 1.4% 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. As a result of the 1.4% power uprate, the calculations that support these operator actions will be re-evaluated. 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 Level II analysis included in the IP3 PRA is based on the fission product inventory of the present core. The containment release categories will be slightly affected by the 1.4% power uprate. Large Early Release fractions will be recalculated. However, based on the uncertainties in the Large Early Release results and the small change in RTP, no significant change in Large Early Release frequency is expected, and the change will remain below current NRC acceptance criteria.

12.3 ENVIRONMENTAL IMPACT CONSIDERATION

The proposed 1.4% power uprate will increase the current licensed core power level from 3025 MWt to 3067.4 MWt, and result in an upgrade of the NSSS thermal power from 3037 MWt to approximately 3082 MWt. The environmental review conducted for the 1.4% power uprate considered the need and the resulting environmental impact associated with it. This review included considering the IP3 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 could be expected for the proposed 1.4% power uprate as discussed herein.

Final Environmental Statement

Environmental issues associated with the issuance of an operating license for IP3 were originally evaluated in the IP3 FES that was approved by the NRC in February 1975.

The proposed 1.4% power uprate is projected to increase the plant's rejected heat by a similar percentage. However, the NRC approved FES related to operation of Indian Point Nuclear Generating Plant Unit

No. 3 (Volume 1, page I-2, Section I) has already addressed plant operation up to a maximum calculated thermal power of 3,216 MWt. Therefore, the slight increase in rejected heat has already been bounded and determined to not significantly affect the quality of the environment. Also, the 1.4% power uprate involves no significant change in 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 3,216 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 of this report.)

Circulating Water Discharge Limits

IP3 is required to maintain a 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% 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, IP3 varies the speed of the circulation water pumps based on plant operation and river water temperature. This will not be affected by the 1.4% power 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% 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% 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% power uprate may increase this temperature by 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, since the quality, quantity, and temperature of the plant discharges are not affected by the 1.4% power uprate, the IP3 SPDES permit requirements and limitations are not affected by the 1.4% power uprate.

Conclusion

The environmental review for the 1.4% thermal power uprate concludes that there are no significant associated radiological or non-radiological effects. The thermal power uprate will have no significant effect on the quality of human environment, and does not involve an unreviewed environmental question.

13 CONCLUSION

This report demonstrates that the 1.4% power uprate can be safely implemented at IP3. The analyses and evaluations described herein demonstrate that all applicable acceptance criteria will continue to be met based on operation at the 1.4% power uprate conditions at 3067.4 MWt, and that there are No Significant Hazards related to this power uprate according to the regulatory criteria of 10 CFR 50.92. Furthermore, the 1.4% power uprate will have no significant effect on the quality of human environment, and does not involve an unreviewed environmental question.