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Docket Number 50-346

License Number NPF-3

Serial Number 2692

October 12, 2001

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Subject: License Amendment Application to Increase Allowable Power (License Amendment Request No. 00-0006)

Ladies and Gentlemen:

Enclosed is an application for an amendment to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1 Operating License Number NPF-3, including the Appendix A, Technical Specifications. The proposed changes involve: Operating License (OL) paragraph 2.C(1), Maximum Power Level; OL paragraph 2.C(3)(d), Additional Conditions; Technical Specification (TS) 1.3, Definitions – Rated Thermal Power; TS 2.1.1, Safety Limits – Reactor Core, and associated Bases; TS 2.2.1, Limiting Safety System Settings – Reactor Protection System Setpoints, and associated Bases; TS 3/4.1.1.3, Reactivity Control Systems – Moderator Temperature Coefficient; TS 3/4.2.5, Power Distribution Limits – DNB Parameters; TS 3/4.4.9.1, Reactor Coolant System – Pressure/Temperature Limits, and associated Bases; and TS 6.9.1.7, Core Operating Limits Report.

The proposed amendment would make the necessary TS changes to allow an increase in the authorized rated thermal power from 2772 MWt to 2817 MWt (approximately 1.63%), based on the use of Caldon Inc. Leading Edge Flow Meter (LEFM) CheckPlus™ System instrumentation to improve the accuracy of the feedwater mass flow input to the plant power calorimetric measurement. It is noted that much of the analyses which were performed to support this power increase conservatively assumed an increased power level of 1.7% (2819 MWt). However, based on a calorimetric uncertainty analysis performed subsequent to the earlier analyses, a power increase of 1.63% is achievable.

The proposed power uprate is based on a redistribution of analytical margin originally required of Emergency Core Cooling System (ECCS) evaluation models performed in accordance with the requirements set forth in Title 10 of the Code of Federal Regulations

A001

(CFR), Section 50, Appendix K, "ECCS Evaluation Models." Appendix K mandated consideration of 102% of the licensed power level for ECCS evaluation models of light water reactors. The NRC approved a change to the requirements of 10CFR50, Appendix K on June 1, 2000 (Federal Register (FR), 65 FR 34913), providing licensees with the option of maintaining the 2% power margin between the licensed power level and the assumed power level for the ECCS evaluation, or applying a reduced margin for ECCS evaluation based on an accounting of uncertainties due to instrumentation error.

The technical basis for the use of the LEFM instrumentation is provided in Caldon Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, dated March, 1997, as supplemented by the latest revision of Caldon Inc. Engineering Report-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM√™ or LEFM CheckPlus™ System." Engineering Report-80P was approved in the NRC Safety Evaluation for the Comanche Peak Steam Electric Station Units 1 and 2 , dated March 8, 1999. Engineering Report-157P Revision 4 was previously submitted to the NRC via Entergy Operations, Inc. letter dated October 9, 2001.

The proposed amendment would also revise the moderator temperature coefficient requirements listed in TS Limiting Condition for Operation (LCO) 3.1.1.3, revise the Departure from Nucleate Boiling (DNB) parameters listed in TS Table 3.2-2 relating to reactor coolant pressure, and make a clarification to a note in the same table. The changes to LCO 3.1.1.3 make the TS consistent with the current Loss of Coolant Accident (LOCA) analyses, and the changes to TS Table 3.2-2 correct slightly non-conservative values in TS (thereby, eliminating the need for assessing a penalty in future fuel reload designs) and make an administrative clarification. These changes are unrelated to the power uprate changes.

In an effort to minimize the potential for NRC Requests for Additional Information (RAIs), the DBNPS has developed this license amendment application taking into consideration the RAIs made by the NRC staff in their review of similar power uprate license amendment applications for the Comanche Peak Steam Electric Station Units 1 and 2 (Amendment No. 72 to Facility Operating License No. NPF-87, and Amendment No. 72 to Facility Operating License No. NPF-89, dated September 30, 1999), the Watts Bar Nuclear Plant Unit 1 (Amendment No. 31 to Facility Operating License No. NPF-90, dated January 19, 2001), and the Beaver Valley Power Station Units 1 and 2 (Amendment No. 243 to Facility Operating License No. DPR-66, and Amendment No. 122 to Facility Operating License No. NPF-73, dated September 24, 2001). Attachments 9, 10, and 11 to Enclosure 1 describe how each of the applicable RAIs were addressed. Attachment 11 to Enclosure 1 also addresses the four criteria described in Section 3.0 of the March 8, 1999 NRC Safety Evaluation issued for the Comanche Peak dockets for review of the Caldon Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, dated March, 1997.

The DBNPS requests that the enclosed license amendment application be approved by the NRC by April 15, 2002. This will support the planned operation of the DBNPS at the proposed increased power level following startup from the Thirteenth Refueling Outage (13RFO).

Please note that as described in the attached Affidavit (Attachment 7 to Enclosure 1), the Framatome ANP Document 32-5012428-00, "Davis-Besse Heat Balance Uncertainty Calc. – App. K Uprate," dated June 8, 2001 (Attachment 8 to Enclosure 1) is considered proprietary in its entirety, and, pursuant to 10 CFR 2.790, it is requested that this information be withheld from public disclosure.

Should you have any questions or require additional information, please contact Mr. David H. Lockwood, Manager - Regulatory Affairs, at (419) 321-8450.

Very truly yours,

MKL

Enclosures

cc: J. E. Dyer, Regional Administrator, NRC Region III
S. P. Sands, NRC/NRR Project Manager
D. J. Shipley, Executive Director, Ohio Emergency Management Agency, State of Ohio (NRC Liaison)
D. S. Simpkins, NRC Region III, DB-1 Resident Inspector
Utility Radiological Safety Board

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Enclosure 1
Page 1

APPLICATION FOR AMENDMENT

TO

FACILITY OPERATING LICENSE NUMBER NPF-3

DAVIS-BESSE NUCLEAR POWER STATION

UNIT NUMBER 1

Attached are the requested changes to the Davis-Besse Nuclear Power Station, Unit Number 1 Facility Operating License Number NPF-3. The Safety Assessment and Significant Hazards Consideration is included as Attachment 1.

The proposed changes (submitted under cover letter Serial Number 2692) concern:

Operating License:

2.C(1)	Maximum Power Level
2.C(3)(d)	Additional Conditions

Appendix A, Technical Specifications (TS):

1.3	Definitions – Rated Thermal Power
2.1.1	Safety Limits – Reactor Core, and associated Bases
2.2.1	Limiting Safety System Settings – Reactor Protection System Setpoints, and associated Bases
3/4.1.1.3	Reactivity Control Systems – Moderator Temperature Coefficient
3/4.2.5	Power Distribution Limits – DNB Parameters
3/4.4.9.1	Reactor Coolant System – Pressure/Temperature Limits, and associated Bases
6.9.1.7	Core Operating Limits Report

I, Guy G. Campbell, state that (1) I am Vice President - Nuclear of the FirstEnergy Nuclear Operating Company, (2) I am duly authorized to execute and file this certification on behalf of the Toledo Edison Company and The Cleveland Electric Illuminating Company, and (3) the statements set forth herein are true and correct to the best of my knowledge, information and belief.

By: [signature]

Guy G. Campbell, Vice President - Nuclear

Affirmed and subscribed before me this 12th day of October, 2001.

Nora L. Flood

Notary Public, State of Ohio - Nora L. Flood
My commission expires September 4, 2002.

The following information is provided to support issuance of the requested changes to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1 Operating License Number NPF-3, including the Appendix A, Technical Specifications. The proposed changes involve: Operating License (OL) paragraph 2.C(1), Maximum Power Level; OL paragraph 2.C(3)(d), Additional Conditions; Technical Specification (TS) 1.3, Definitions – Rated Thermal Power; TS 2.1.1, Safety Limits – Reactor Core, and associated Bases; TS 2.2.1, Limiting Safety System Settings – Reactor Protection System Setpoints, and associated Bases; TS 3/4.1.1.3, Reactivity Control Systems – Moderator Temperature Coefficient; TS 3/4.2.5, Power Distribution Limits – DNB Parameters; TS 3/4.4.9.1, Reactor Coolant System – Pressure/Temperature Limits, and associated Bases; and TS 6.9.1.7, Core Operating Limits Report.

A. Time Required to Implement: The License Amendment associated with this license amendment application is to be implemented within 120 days following NRC issuance.

B. Reason for Change (License Amendment Request Number 00-0006):

The proposed amendment would make the necessary TS changes to allow an increase in the authorized rated thermal power from 2772 MWt to 2817 MWt (approximately 1.63%), based on the use of Caldon Inc. Leading Edge Flow Meter (LEFM) CheckPlus™ System instrumentation to improve the accuracy of the feedwater mass flow input to the plant power calorimetric measurement.

The proposed amendment would also revise the moderator temperature coefficient requirements listed in TS Limiting Condition for Operation (LCO) 3.1.1.3, revise the Departure from Nucleate Boiling (DNB) parameters listed in TS Table 3.2-2 relating to reactor coolant pressure, and make a clarification to a note in the same table. These changes are unrelated to the power uprate changes.

C. Attachments to Enclosure 1:

1. Safety Assessment and Significant Hazards Consideration
2. Environmental Consideration
3. Davis-Besse Nuclear Power Station, Licensing Report, Power Uprate Program
4. Not Used
5. Not Used
6. Not Used
7. Framatome ANP Affidavit Pursuant to 10 CFR 2.790

8. Framatome ANP Document 32-5012428-00, "Davis-Besse Heat Balance Uncertainty Calculation – Appendix K Uprate," dated June 8, 2001

9. Applicability of Comanche Peak RAI Questions to Proposed Davis-Besse Power Uprate

10. Applicability of Watts Bar RAI Questions to Proposed Davis-Besse Power Uprate

11. Applicability of Beaver Valley RAI Questions to Proposed Davis-Besse Power Uprate

Docket Number 50-346
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Serial Number 2692
Enclosure 1
Attachment 1

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION
FOR
LICENSE AMENDMENT REQUEST NUMBER 00-0006

(44 pages follow)

SAFETY ASSESSMENT AND SIGNIFICANT HAZARDS CONSIDERATION FOR LICENSE AMENDMENT REQUEST NUMBER 00-0006

TITLE:

Proposed Modifications to the Davis-Besse Nuclear Power Station (DBNPS) Unit Number 1, Facility Operating License NPF-3, Including the Appendix A Technical Specifications, to Allow a Power Uprate and to Revise Departure From Nucleate Boiling Parameters.

DESCRIPTION:

The proposed amendment would increase the authorized rated thermal power from 2772 MWt to 2817 MWt (approximately 1.63%), based on the use of Caldon Inc. Leading Edge Flow Meter (LEFM) CheckPlus™ System instrumentation to improve the accuracy of the main feedwater mass flow input to the plant power calorimetric measurement. The DBNPS plans to install the LEFM CheckPlus™ System in both feedwater trains in the upcoming Thirteenth Refueling Outage (13RFO).

The proposed power uprate is based on a redistribution of analytical margin originally required of Emergency Core Cooling System (ECCS) evaluation models performed in accordance with the requirements set forth in Title 10 of the Code of Federal Regulations (CFR), Section 50, Appendix K, "ECCS Evaluation Models." Appendix K mandated consideration of 102% of the licensed power level for ECCS evaluation models of light water reactors. The NRC approved a change to the requirements of 10CFR50, Appendix K on June 1, 2000 (Federal Register (FR), 65 FR 34913), providing licensees with the option of maintaining the 2% power margin between the licensed power level and the assumed power level for the ECCS evaluation, or applying a reduced margin for ECCS evaluation based on an accounting of uncertainties due to instrumentation error.

The LEFM CheckPlus™ System will provide on-line measurement of main feedwater flow and temperature, which is used, in turn, for determining the reactor thermal power. This system uses acoustic energy pulses to determine the main feedwater mass flow rate and temperature. The LEFM CheckPlus™ System flow meter consists of a measurement section (spool piece) in each of the two 18-inch main feedwater lines that holds sixteen ultrasonic transducer assemblies, and an electronic signal processing cabinet.

The LEFM CheckPlus™ System will be used in lieu of the present venturi-based flow indication and the resistance temperature detector (RTD) temperature data in performing the plant calorimetric measurement calculation for reactor thermal power. The improved accuracy of this system will result in less total uncertainty in determining the actual reactor thermal power, thereby allowing the reactor to be operated at an increased ("uprated") power level.

The proposed amendment would also revise the moderator temperature coefficient requirements listed in TS Limiting Condition for Operation (LCO) 3.1.1.3, revise the

Departure from Nucleate Boiling (DNB) parameters listed in TS Table 3.2-2 relating to reactor coolant pressure, and make a clarification to a note in the same table. These changes are unrelated to the power uprate changes.

Each of the proposed revisions is shown on the attached marked-up Operating License pages. The proposed changes are described in further detail as follows:

Operating License Paragraph 2.C(1), Maximum Power Level

It is proposed to revise the first sentence of this statement to increase the authorized power level by having it read as follows:

> FENOC is authorized to operate the facility at steady state reactor core power levels not in excess of 2817 megawatts (thermal).

Operating License Paragraph 2.C(3)(d), Additional Conditions

It is proposed to revise this paragraph to reduce the number of Effective Full Power Years before a reanalysis is performed for low temperature reactor coolant system overpressure events by having it read as follows:

> Prior to operation beyond 20 Effective Full Power Years, FENOC shall provide to the NRC a reanalysis and proposed modifications, as necessary, to ensure continued means of protection against low temperature reactor coolant system overpressure events.

TS 1.3, Definitions – Rated Thermal Power

It is proposed to revise TS 1.3 to reflect the authorized power level by having it read as follows:

> RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2817 MWt.

TS 2.1.1, Safety Limits – Reactor Core, and associated Bases

TS 2.1.1 states that the combination of reactor coolant outlet pressure and outlet temperature shall not exceed the safety limit shown in Figure 2.1-1, Reactor Core Safety Limit. It is proposed to revise Figure 2.1-1 to reflect updated data as a result of the proposed power uprate. Related changes to the associated TS Bases are also proposed, including a revised Bases Figure 2.1, Pressure/Temperature Limits at Maximum Allowable Power for Minimum DNBR.

TS 2.2.1, Limiting Safety System Settings – Reactor Protection System Setpoints, and associated Bases

TS 2.2.1 states that the Reactor Protection System (RPS) instrumentation setpoints shall be set consistent with the Allowable Values shown in Table 2.2-1, Reactor Protection System Instrumentation Trip Setpoints. It is proposed to revise the Allowable Values listed in

Table 2.2-1 for Functional Unit 2, High Flux, and Functional Unit 7, RC Pressure-Temperature. The proposed new values are a result of the proposed power uprate. Related changes to the associated TS Bases are also proposed.

TS 3/4.1.1.3, Reactivity Control Systems – Moderator Temperature Coefficient

Limiting Condition for Operation 3.1.1.3 presently states, in part:

> The moderator temperature coefficient (MTC) shall be:
>
> a. Less positive than 0.9 x 10^{-4} Δk/k/°F whenever THERMAL POWER is < 95% of RATED THERMAL POWER,
>
> b. Less positive than 0.0 x 10^{-4} Δk/k/°F whenever THERMAL POWER is ≥ 95% of RATED THERMAL POWER

It is proposed to revise the 95% RATED THERMAL POWER limit in both LCO 3.1.1.3.a and 3.1.1.3.b to 80%. This change reflects the current Loss of Coolant Accident (LOCA) analyses and is unrelated to the proposed power uprate.

TS 3/4.2.5, Power Distribution Limits – DNB Parameters

It is proposed to revise the required measured reactor coolant pressure parameters listed in Table 3.2-2, DNB Margin, from ≥ 2062.7 to ≥ 2064.8 psig, and from ≥ 2058.7 to ≥ 2060.8 psig, for four and three reactor coolant pump operation, respectively. In addition, it is proposed to modify note (3) in the table to delete the discussion of its basis. This note applies to the reactor coolant flow rate parameter. The revised note would read as follows:

> These minimum required measured flows include a flow rate uncertainty of 2.5%.

These changes correct slightly non-conservative values in TS and make an administrative clarification, and are unrelated to the proposed power uprate.

TS 3/4.4.9.1, Reactor Coolant System – Pressure/Temperature Limits, and associated Bases

The titles of TS Figures 3.4-2, 3.4-3, and 3.4-4 presently state that they apply for the first 21 Effective Full Power Years (EFPY) of operation. As a result of the proposed power uprate, it is proposed to revise the titles to reflect that these figures now apply to the first 20 EFPY of operation.

Similar changes are proposed to the associated TS Bases, which presently refer to the 21 EFPY basis for the TS Figures.

TS 6.9.1.7, Administrative Controls – Core Operating Limits Report

It is proposed to revise TS 6.9.1.7 to include the following as a result of the power uprate:

As described in reference documents listed in accordance with the instructions given above, when an initial assumed power level of 102% of rated thermal power is specified in a previously approved method, 100.37% of rated thermal power may be used when input for reactor thermal power measurement of feedwater mass flow is by the Leading Edge Flow Meter (LEFM) CheckPlus™ System.

SYSTEMS, COMPONENTS, AND ACTIVITIES AFFECTED:

The proposed amendment would allow an increase in the authorized reactor core power level from 2772 MWt to 2817 MWt. The systems, components, and activities affected by the proposed power uprate are described in the attached "Davis-Besse Nuclear Power Station, Licensing Report, Power Uprate Program" (Reference 3).

The proposed amendment would decrease the rated thermal power values at which the moderator temperature coefficient (MTC) limits are specified, so as to make the TS consistent with the current LOCA analyses.

The proposed amendment would also revise Departure from Nucleate Boiling (DNB) parameters listed in TS Table 3.2-2 relating to reactor coolant pressure, and make a clarification to a note in the same table. The parameters listed in Table 3.2-2 are based upon values utilized in the transient and accident analyses. In Mode 1 (Power Operation), surveillances are performed to verify that the plant is operating within the limits listed in the table.

FUNCTIONS OF THE AFFECTED SYSTEMS, COMPONENTS, AND ACTIVITIES:

The functions of the systems, components, and activities affected by the proposed power uprate are described in the attached "Davis-Besse Nuclear Power Station, Licensing Report, Power Uprate Program" (Reference 3).

The limits on the moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle.

The limits on the DNB related parameters listed in TS Table 3.2-2 assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses.

EFFECTS ON SAFETY:

The DBNPS is presently licensed for a core thermal power rating of 2,772 MWt. Through the use of more accurate feedwater flow measurement equipment, approval is sought to increase this core power by approximately 1.63% to 2817 MWt. The impact of the proposed power uprate on applicable systems, components, and activities has been evaluated, and is described in the attached "Davis-Besse Nuclear Power Station, Licensing Report, Power Uprate Program" (Reference 3).

The proposed power uprate is based on a redistribution of analytical margin originally required of Emergency Core Cooling System (ECCS) evaluation models performed in accordance with the requirements set forth in Title 10 of the Code of Federal Regulations (CFR), Section 50, Appendix K, "ECCS Evaluation Models." Appendix K mandated consideration of 102% of the licensed power level for ECCS evaluation models of light water reactors. The NRC approved a change to the requirements of 10CFR50, Appendix K on June 1, 2000 (Federal Register (FR), 65 FR 34913), providing licensees with the option of maintaining the 2% power margin between the licensed power level and the assumed power level for the ECCS evaluation, or applying a reduced margin for ECCS evaluation based on an accounting of uncertainties due to instrumentation error.

Based on the proposed use of the Caldon Inc. Leading Edge Flow Meter (LEFM) CheckPlus™ System instrumentation, the allowance for power measurement uncertainties can be reduced. Complete technical support for this conclusion is discussed in detail in Caldon Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, dated March, 1997, as supplemented by Caldon Inc. Engineering Report-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM√™ or LEFM CheckPlus™ System." Engineering Report-80P was approved in the NRC Safety Evaluation for the Comanche Peak Steam Electric Station Units 1 and 2 , dated March 8, 1999.

The installation and post-maintenance testing of the LEFM system will be completed prior to increasing power above the current limit of 2772 MWt. New procedures for maintenance and calibration of the LEFM system will be developed based on vendor recommendations. A requirement will be placed in the DBNPS Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM) to address LEFM unavailability. Should the LEFM system be unavailable, the current feedwater flow instrumentation will be used as input to the core power calorimetric, and the core power will be limited to the original licensed power level of 2772 MWt.

Summary of Technical Evaluation

As described above, the impact of the proposed power uprate on applicable systems, components, and activities has been evaluated, and is described in the attached "Davis-Besse Nuclear Power Station, Licensing Report, Power Uprate Program" (Reference 3). Sections 1.0 and 2.0 of the report provide general background information. Section 3.0 of the report provides the details on the safety analysis. Section 3.1 of the report describes the safety analysis approach, and Section 3.2 provides details on the LEFM instrumentation. Section 3.3 of the report discusses the revised NSSS design thermal and hydraulic parameters that were modified as a result of the power uprate and that serve as the basis for all of the NSSS analyses and evaluations. Section 3.4 of the report provides the details supporting the conclusion that no design transient modifications are required to accommodate the revised NSSS design conditions. Sections 3.5 through 3.7 of the report present the system and component evaluations completed for the revised design conditions. Section 3.8 of the report summarizes the effects of the uprate on the balance-of-plant (secondary) systems based upon a heat balance evaluation. Section 3.9 of the report provides an analysis of the effects on the

electrical power systems. Section 3.10 of the report provides the results of the accident analyses and evaluations performed for the steam generator tube rupture, loss-of-coolant-accident (LOCA), and non-LOCA areas. Sections 3.11 and 3.12 of the report summarize the containment accident analyses and evaluations and the radiological consequence evaluations. Section 3.13 of the report contains the results of the fuel-related analyses. Section 4.0 of the report addresses miscellaneous issues and programs. The results of all of the analyses and evaluations performed demonstrate that all acceptance criteria continue to be met.

Proposed Operating License and Technical Specification Changes (see attached)

Operating License Paragraph 2.C(1), Maximum Power Level

The proposed amendment reflects the proposed increase in the authorized core power level. As previously described, the acceptability of this approach is evaluated in Reference 3. Based on the Reference 3 evaluation, this proposed change will have no adverse effect on nuclear safety.

Operating License Paragraph 2.C(3)(d), Additional Conditions

Due to the proposed changes to TS 3/4.4.9.1, FENOC will now be required to update the pressure/temperature limit curves provided in TS 3/4.4.9.1 prior to 20 Effective Full Power Years (EFPY) of operation. The curves are presently applicable through the first 20 EFPY of operation. The proposed change to OL 2.C(3)(d) reflects this administrative requirement, and will have no adverse effect on nuclear safety.

TS 1.3, Definitions – Rated Thermal Power

The proposed amendment reflects the proposed increase in the authorized core power level. As previously described, the acceptability of this approach is evaluated in Reference 3. Based on the Reference 3 evaluation, this proposed change will have no adverse effect on nuclear safety.

TS 2.1.1, Safety Limits – Reactor Core, and associated Bases

The proposed revision to TS Figure 2.1-1 is in accordance with updated analyses in support of the proposed power uprate, as described in Reference 3. These analyses showed that more restrictive reactor core safety limits were appropriate. Based on the Reference 3 evaluation, this proposed change will have no adverse effect on nuclear safety.

The proposed changes to the associated TS Bases, including a revised Bases Figure 2.1 are related to the above changes, and, as such, are administrative changes that will have no adverse effect on nuclear safety. These changes are supported by the reviews and evaluations described in Reference 3, and reflect the use of the Framatome ANP evaluation method for Statistical Core Design, BAW-10187P-A (Reference 8), that is the basis for the power uprate program DNBR safety evaluations.

TS 2.2.1, Limiting Safety System Settings – Reactor Protection System Setpoints, and associated Bases

The proposed changes to the TS Table 2.2-1 Allowable Values for Functional Unit 2, High Flux, and Functional Unit 7, RC Pressure-Temperature, are in accordance with updated analyses in support of the proposed power uprate, as described in Reference 3.

The existing safety analysis models the nuclear over-power (high flux) limit as a function of total power (MWt) and not percent of rated thermal power (RTP). Since all of the safety analyses were not reanalyzed, the total power allowed must be preserved. Therefore, the high flux limit, expressed in terms of percent of RTP, must be reduced for the new uprated power level. With the proposed total power increase to 2817 MWt, the new high flux limit becomes:

(1.12 * 2772 MWt) / (2817 MWt) * 100% = 110.2% of RTP (rounded down for conservatism)

Since the high flux limit and heat balance error are being revised, the high flux trip setpoint Allowable Value must also be changed. The heat balance error is reduced to a minimum value of 0.37% from 2% RTP due to the proposed power uprate. However, the total steady state and transient induced errors, 4% of RTP, and the instrumentation error, 0.835% of RTP, remain unchanged. The new high flux Allowable Value, based on the above values, becomes:

110.2 – 0.37 – 4.0 – 0.835 = 104.9% of RTP (rounded down for conservatism)

Based on the more restrictive reactor core safety limits, as described above, a more restrictive variable low pressure (RC pressure-temperature) trip setpoint was calculated. The new trip setpoint Allowable Value for the Channel Functional Test is:

(16.25 T_{out} °F – 8034) psig

Based on the above evaluation, these proposed changes will have no adverse effect on nuclear safety. In addition, the proposed changes to the associated TS Bases are related to the above changes, and, as such, are administrative changes that will have no adverse effect on nuclear safety. These changes are supported by the reviews and evaluations described in Reference 3.

TS 3/4.1.1.3, Reactivity Control Systems – Moderator Temperature Coefficient

The proposed changes to this TS section are unrelated to the power uprate changes.

The proposed change to revise the 95% RATED THERMAL POWER limit in both LCO 3.1.1.3.a and 3.1.1.3.b from 95% to 80% will make the TS consistent with the current LOCA analyses, as currently described in Updated Safety Analysis Report (USAR) Section 15.4.6.8.2, "Positive Moderator Temperature Coefficient (MTC) Analysis." These analyses were based on a nominal core power level of 2966 MWt. Since the proposed values are more conservative than the current values, there is no

adverse effect on nuclear safety.

TS 3/4.2.5, Power Distribution Limits – DNB Parameters

The proposed changes to this TS section are unrelated to the power uprate changes.

The proposed amendment revises the required measured reactor coolant pressure parameters listed in Table 3.2-2, DNB Margin, from ≥ 2062.7 to ≥ 2064.8 psig, and from ≥ 2058.7 to ≥ 2060.8 psig, for four and three reactor coolant pump operation, respectively. The fuel vendor had identified that the calculated minimum pressure drop from the core outlet to the hot leg pressure tap, upon which the Table 3.2-2 minimum pressure criteria is based, was not correctly factored into the minimum pressure criteria. Therefore, the current reactor coolant pressure parameters listed in the table are slightly non-conservative. In order to offset this slight non-conservatism, a DNB penalty has been assessed in the past against the retained DNB margin in the reload licensing analyses. Once the proposed changes to Table 3.2-2 have been made, this offset will no longer be necessary for future core reload analyses. Since the proposed values are more conservative than the current values, there is no adverse effect on nuclear safety.

In addition, the proposed amendment modifies note (3) in Table 3.2-2 to delete the discussion that the reactor coolant flow rate parameters "are based on a minimum of 52 lumped burnable poison rod assemblies in place in the core." While historically accurate as the basis for the reactor coolant flow rate parameters, the note is potentially a source of confusion in that the current core design philosophy does not utilize lumped burnable poison rod assemblies (BPRAs). Previously, with the BPRAs positioned in control rod guide tubes, reactor coolant flow through the guide tubes was impeded, allowing more flow to the heated fuel rods in the core (i.e., reducing the core bypass flow). A greater core bypass flow would result in less flow through the heated fuel rods, resulting in less DNB margin. With the removal of the BPRAs, an orificed control rod guide tube feature was added to the fuel design to compensate for the undesirable increase in core bypass flow that would otherwise occur. The Table 3.2-2 reactor coolant flow rate values continue to reflect the minimum flow assumptions used in the DNB analyses. Since the proposed Table 3.2-2 note (3) change is an administrative clarification that eliminates a potential source of confusion, there is no adverse effect on nuclear safety.

TS 3/4.4.9.1, Reactor Coolant System – Pressure/Temperature Limits, and associated Bases

The proposed revisions to the titles of TS Figures 3.4-2, 3.4-3, and 3.4-4 is in accordance with updated analyses in support of the proposed power uprate, as described in Reference 3. These analyses showed that due to the power uprate, the pressure/temperature limit curves shown in the figures are now only applicable to the first 20 Effective Full Power Years (EFPY) of operation, rather than the current 21 EFPY. This change continues to ensure that the Reactor Coolant System pressure/temperature limits will be appropriately maintained and, accordingly, this change will have no adverse effect on nuclear safety.

The similar changes proposed to the associated TS Bases, which presently refer to the 21 EFPY basis for the TS Figures, are similar changes that will have no adverse effect on nuclear safety.

TS 6.9.1.7, Administrative Controls – Core Operating Limits Report

This TS Section describes where the analytical methods used to determine core operating limits are listed. Some of the analytical methods apply a 2% uncertainty to reactor power, consistent with the version of 10 CFR 50, Appendix K that was in effect at the time the analysis was performed.

In general, the proposed power uprate is accomplished by replacing the prescribed 2% power measurement uncertainty with a plant specific uncertainty value, in effect trading the increased accuracy associated with the LEFM for increased power. As previously described, the acceptability of this approach is evaluated in Reference 3.

The revision of each of the analyses used to determine core operating limits, specifically to accommodate the proposed power uprate, would be a substantial administrative burden. In lieu of this administrative burden, it is proposed to allow the present versions of the reports to apply to the proposed power uprate, conditioned upon the LEFM being used to measure feedwater mass flow as the input to the reactor thermal power measurement. Consistent with the approach taken by the Tennessee Valley Authority (Reference 6), a requirement will be placed in the DBNPS USAR requiring that future, plant-specific revisions of these reports, incorporate consideration of the 1.63% power uprate. The proposed addition to TS 6.9.1.7 reflects this approach, and is an administrative change that will have no adverse effect on nuclear safety.

Conclusion

Based on the technical basis described in Reference 3 (attached), as summarized above, and based on the above evaluation of each individually proposed OL/TS change, it is concluded that the proposed changes will have no adverse effect on nuclear safety.

SIGNIFICANT HAZARDS CONSIDERATION:

The Nuclear Regulatory Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazard exists due to a proposed amendment to an Operating License for a facility. A proposed amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed changes would: (1) Not involve a significant increase in the probability or consequences of an accident previously evaluated; (2) Not create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Not involve a significant reduction in a margin of safety. The Davis-Besse Nuclear Power Station (DBNPS) has reviewed the proposed changes and determined that a significant hazards consideration does not exist because operation of the Davis-Besse Nuclear Power Station, Unit No. 1, in accordance with these changes would:

1a. Not involve a significant increase in the probability of an accident previously evaluated based on the comprehensive analytical efforts that were performed to demonstrate the acceptability of the proposed power uprate changes. The proposed changes include: revision of the maximum power level limit stated in Operating License (OL) paragraph 2.C(1) and Technical Specification (TS) Section 1.3, increasing the allowable power level from 2772 MWt to 2817 MWt; revision of the reactor core safety limits specified in TS Section 2.1.1; revision of the Reactor Protection System (RPS) high flux and Reactor Coolant System (RCS) pressure-temperature setpoints provided in TS Section 2.2.1; revision of the RCS pressure-temperature limits in TS Section 3/4.4.9.1, and a related change to OL paragraph 2.C(3)(d); and revision of administrative controls associated with the Core Operating Limits Report, as described in TS Section 6.9.1.7. In addition, related changes to the TS Bases associated with these TS Sections are proposed. An evaluation has been performed that identified the systems and components that could be affected by these proposed changes. The evaluation determined that these systems and components will function as designed and that performance requirements remain acceptable.

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

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

All of the Nuclear Steam Supply System (NSSS) systems will continue to perform their intended design functions during normal and accident conditions. The pressurizer spray flow remains above its design value. Thus, the control system design analyses, which credit the flow, do not require any modification. The components continue to comply with applicable structural limits and will continue to perform their intended design functions. Thus, there is no increase in the probability of a structural failure of these components.

All of the NSSS/Balance of Plant (BOP) interface systems will continue to perform their intended design functions. The main steam safety valves will provide adequate relief capacity to maintain the main steam system within design limits.

The current loss of coolant accident (LOCA) hydraulic forcing functions remain bounding.

The reduction in the power measurement uncertainty through the use of the Caldon Leading Edge Flow Meter (LEFM) CheckPlus™ system, allows for certain safety analyses to continue to be used, without modification, at the 2827 MWt power level (102% of 2772 MWt). Other safety analyses performed at a nominal power level of 2772 MWt have been either re-performed or re-evaluated at the 2817 MWt power level, and continue to meet their applicable acceptance criteria. Some existing safety analyses

had been previously performed at a power level greater than 2827 MWt, and thus continue to bound the 2817 MWt power level.

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

In addition to the changes related to the proposed power uprate, unrelated changes are proposed to revise the moderator temperature coefficient requirements listed in TS Section 3.1.1.3, and to revise requirements relating to the Departure from Nucleate Boiling (DNB) parameters listed in TS Section 3.2.5. These proposed changes are conservative changes and clarifications that do not involve any physical change to systems or components, nor do they alter the typical manner in which the systems or components are operated. Therefore, these changes will not result in a significant increase in the probability of an accident.

lb. Not involve a significant increase in the consequences of an accident previously evaluated because the proposed power uprate changes do not alter any assumptions previously made in the radiological consequence evaluations, nor affect mitigation of the radiological consequences of an accident previously evaluated.

The accident radiation dose evaluation was performed at 2827 MWt and is bounding when operating at the proposed 2817 MWt using the LEFM CheckPlus™ flow instrumentation.

The proposed changes unrelated to the power uprate also do not alter any assumption previously made in the radiological consequence evaluations, nor do they affect mitigation of the radiological consequences of an accident previously evaluated. Therefore, these changes will not involve a significant increase in the consequences of an accident previously evaluated.

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

3. Not involve a significant reduction in a margin of safety because extensive analyses of the primary fission product barriers, conducted in support of the proposed power uprate, have concluded that all relevant design criteria remain satisfied, both from the standpoint of the integrity of the primary fission product barrier and from the standpoint

of compliance with the regulatory acceptance criteria. As appropriate, all evaluations have been performed using methods that have either been reviewed and approved by the Nuclear Regulatory Commission (NRC) or that are in compliance with applicable regulatory review guidance and standards.

The proposed changes unrelated to the power uprate do not involve a significant reduction in a margin of safety because they do not involve the potential for a significant increase in a failure rate of any system or component, and existing system and component redundancy is not affected. Also, these changes do not involve any new or significant changes to the initial conditions contributing to accident severity or consequences.

CONCLUSION:

On the basis of the above, the Davis-Besse Nuclear Power Station has determined that the License Amendment Request does not involve a significant hazards consideration.

ATTACHMENT:

Attached are the proposed marked-up changes to the Operating License.

REFERENCES:

1. DBNPS Operating License NPF-3, Appendix A Technical Specifications through Amendment 246.

2. DBNPS Updated Safety Analysis Report through Revision 22.

3. Davis-Besse Nuclear Power Station, Licensing Report, Power Uprate Program.

4. Caldon Inc. Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, dated March, 1997.

5. Caldon Inc. Engineering Report-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM√™ or LEFM CheckPlus™ System."

6. Tennessee Valley Authority letter to NRC dated June 7, 2000, "Watts Bar Nuclear Plant (WBN) Unit 1 – Technical Specification (TS) Change No. 00-06 – Increase Unit 1 Reactor Power to 3459 MWt" (Docket No. 50-390).

7. NRC Safety Evaluation for the Comanche Peak Steam Electric Station, Units 1 and 2 , Docket Nos. 50-445 and 50-446, dated March 8, 1999.

8. BAW-10187P-A, "Statistical Core Design for B&W-Designed 177-FA Plants," B&W Fuel Company, Lynchburg, Virginia, dated March, 1994.

2.C. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

FENOC is authorized to operate the facility at steady state reactor core power levels not in excess of ~~2772~~ 2817 megawatts (thermal). Prior to attaining the power level, Toledo Edison Company shall comply with the conditions identified in Paragraph (3)(o) below and complete the preoperational tests, startup tests and other items identified in Attachment 2 to this license in the sequence specified. Attachment 2 is an integral part of this license.

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(3)(d) Prior to operation beyond ~~21~~ 20 Effective Full Power Years, FENOC shall provide to the NRC a reanalysis and proposed modifications, as necessary, to ensure continued means of protection against low temperature reactor coolant system overpressure events.

1.0 DEFINITIONS

DEFINED TERMS

1.1 The DEFINED TERMS of this section appear in capitalized type and are applicable throughout these Technical Specifications.

THERMAL POWER

1.2 THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant.

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2817 ~~2772~~ MWt.

OPERATIONAL MODE

1.4 An OPERATIONAL MODE shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principal specification and shall be part of the specifications.

OPERABLE - OPERABILITY

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

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of the reactor coolant core outlet pressure and outlet temperature shall not exceed the safety limit shown in Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of reactor coolant core outlet pressure and outlet temperature has exceeded the safety limit, be in HOT STANDBY within one hour.

REACTOR CORE

2.1.2 The combination of reactor THERMAL POWER and AXIAL POWER IMBALANCE shall not exceed the protective limit shown in the CORE OPERATING LIMITS REPORT for the various combinations of three and four reactor coolant pump operation.

APPLICABILITY: MODE 1.

ACTION:

Whenever the point defined by the combination of Reactor Coolant System flow, AXIAL POWER IMBALANCE and THERMAL POWER has exceeded the appropriate protective limit, be in HOT STANDBY within one hour, and comply with the requirements of Specification 6.7.2.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The Reactor Coolant System pressure shall not exceed 2750 psig.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1 and 2 -	Whenever the Reactor Coolant System pressure has exceeded 2750 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within one hour.
MODES 3, 4 and 5 -	Whenever the Reactor Coolant System pressure has exceeded 2750 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

Figure 2.1-1 Reactor Core Safety Limit

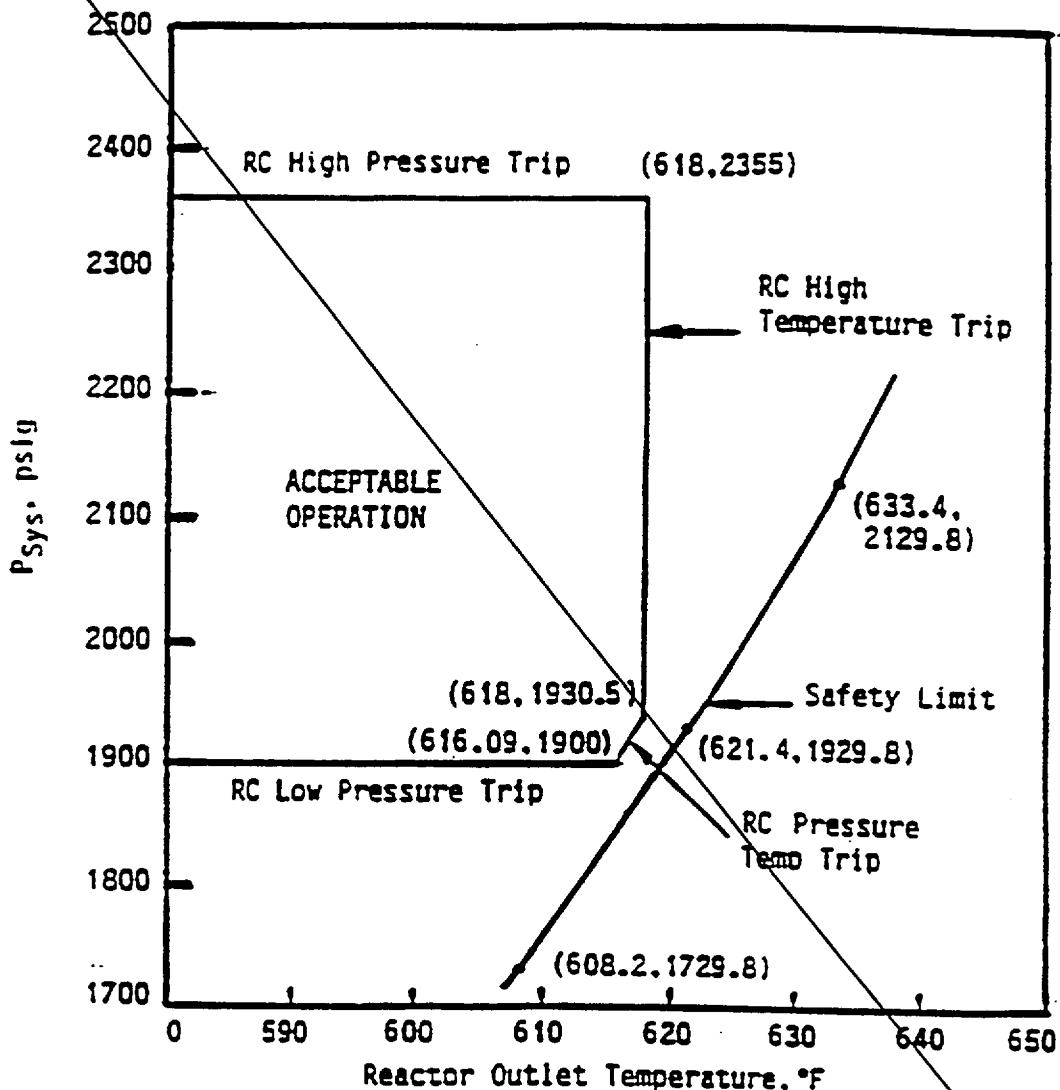

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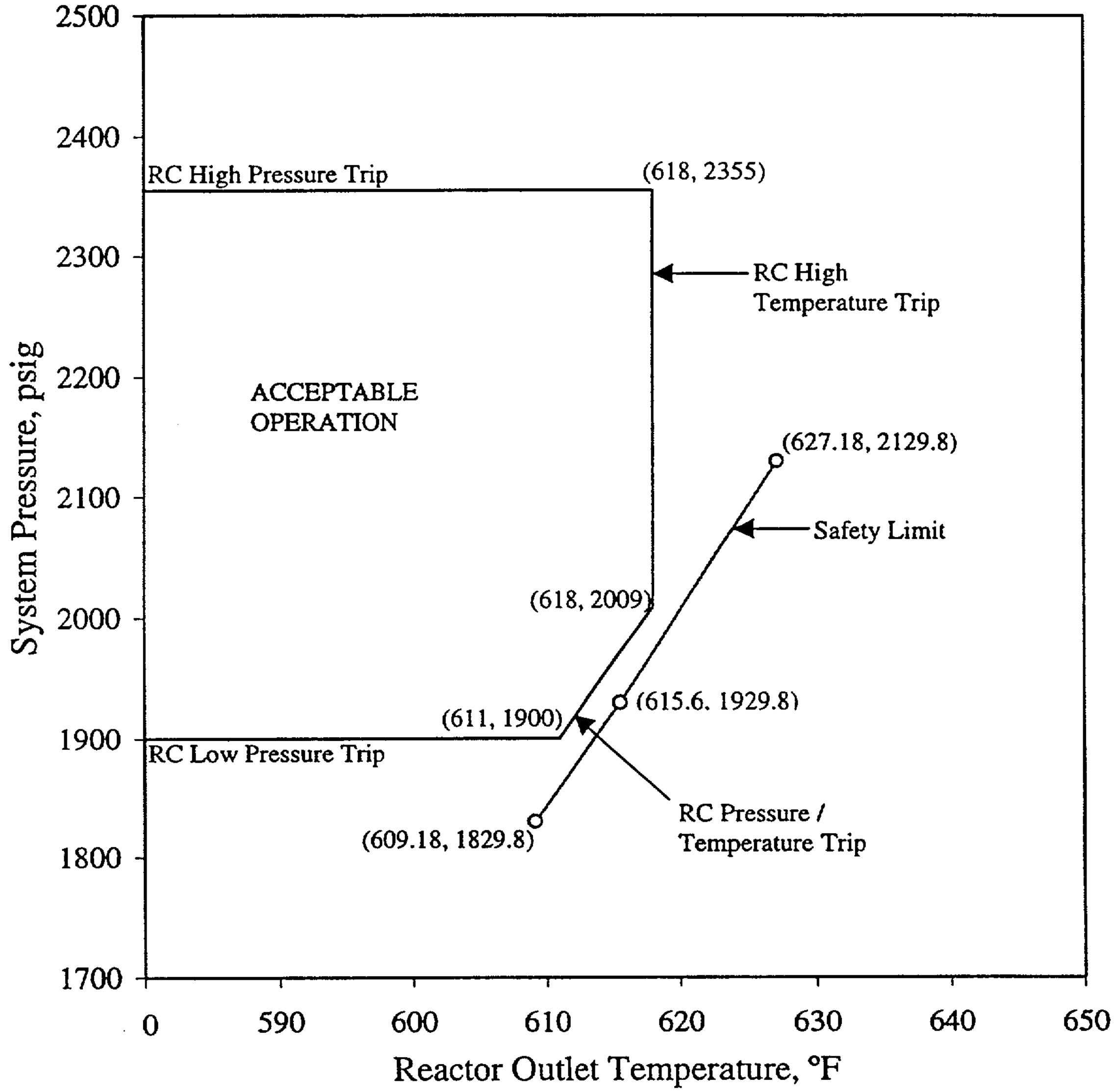
Figure 2.1-1 Reactor Core Safety Limit



2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM SETPOINTS

2.2.1 The Reactor Protection System instrumentation setpoints shall be set consistent with the Allowable Values shown in Table 2.2-1.

APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a Reactor Protection System instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Allowable Value.

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Table 2.2-1 Reactor Protection System Instrumentation Trip Setpoints

Functional unit	Allowable values
1. Manual reactor trip	Not applicable.
2. High flux	≤104.9 ~~105.1~~% of RATED THERMAL POWER with four pumps operating*
	≤80.6% of RATED THERMAL POWER with three pumps operating*
3. RC high temperature	≤618°F*
4. Flux -- Δflux/flow(1)	Pump allowable values not to exceed the limit lines shown in in the CORE OPERATING LIMITS REPORT for four and three pump operation.*
5. RC low pressure(1)	≥1900.0 psig*
6. RC high pressure	≤2355.0 psig*
7. RC pressure-temperature(1)	≥(16.25 ~~16.00~~ T_{out} °F - 8034 ~~7957.5~~) psig*
8. High flux/number of RC pumps on(1)	≤55.1% of RATED THERMAL POWER with one pump operating in each loop*
	≤0.0% of RATED THERMAL POWER with two pumps operating in one loop and no pumps operating in the other loop*
	≤0.0% of RATED THERMAL POWER with no pumps operating or only one pump operating*
9. Containment pressure high	≤4 psig*

Table 2.2-1. (Cont'd)

(1)Trip may be manually bypassed when RCS pressure ≤1820 psig by actuating shutdown bypass provided that:

a. The high flux trip setpoint is ≤5% of RATED THERMAL POWER.

b. The shutdown bypass high pressure trip setpoint of ≤1820 psig is imposed.

c. The shutdown bypass is removed when RCS pressure >1820 psig.

*Allowable value for CHANNEL FUNCTIONAL TEST.

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2.1 SAFETY LIMITS

BASES

2.1.1 AND 2.1.2 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime would result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB using critical heat flux (CHF) correlations. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The B&W-2 and BWC CHF correlations have been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The B&W-2 correlation applies to Mark-B fuel and the BWC correlation applies to all B&W fuel with zircaloy or M5 spacer grids. The minimum value of the DNBR, accounting only for DNBR correlation uncertainty, during steady state operation, normal operational transients, and anticipated transients is limited to 1.30 (B&W-2) and 1.18 (BWC). The minimum value of DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.313 (BWC) and accounts for all uncertainty values considered with the statistical core design methodology. The value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR equal to or greater than the correlation limit is predicted for the maximum possible thermal power of 110.2% of 2817 MWt ~~112%~~ when the reactor coolant flow is 380,000 GPM, which is approximately 108% of design flow rate for four operating reactor coolant pumps. (The minimum required measured flow is 389,500 GPM). This curve is based on the design hot channel factors with potential fuel densification and fuel rod bowing effects.

The design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to minimum allowable control rod withdrawal, and form the core DNBR design basis.

SAFETY LIMITS

BASES

The CORE OPERATING LIMITS REPORT includes curves for protective limits for AXIAL POWER IMBALANCE and for nuclear overpower based on reactor coolant system flow. A protective limit is a cycle-specific limit that ensures that a safety limit is not exceeded by requiring operation within both the cycle design (operating) limits and the Reactor Protection System setpoints. These protective limit curves reflect the more restrictive of two thermal limits and account for the effects of potential fuel densification and potential fuel rod bow:

1. The DNBR limit produced by a design nuclear power peaking factor as described in the CORE OPERATING LIMITS REPORT or the combination of the radial peak, axial peak, and position of the axial peak that yields no less than the DNBR limit.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limits for all fuel designs during the operating cycle are listed in the CORE OPERATING LIMITS REPORT.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the reactor power imbalance produced by the power peaking.

The specified flow rates for the CORE OPERATING LIMITS REPORT curves for protective limits for AXIAL POWER IMBALANCE and for nuclear overpower based on reactor coolant system flow correspond to the analyzed minimum flow rates with four pumps and three pumps, respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in BASES Figure 2.1. The curves of BASES Figure 2.1 represent the conditions at which a minimum DNBR equal to the DNBR limit is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to the corresponding DNB correlation quality limit (+22% (B&W-2) or +26% (BWC)), whichever condition is more restrictive.

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SAFETY LIMITS

BASES

For the curve of BASES Figure 2.1, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than the Statistical Design Limit (SDL) of 1.313 (BWC) ~~1.30 (B&W-2) or 1.18 (BWC)~~ and a local quality at the point of minimum DNBR less than +22% (B&W-2) or +26% (BWC) for that particular reactor coolant pump situation. The DNBR curve for three pump operation is less restrictive than the four pump curve.

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Boiler and Pressure Vessel Code which permits a maximum transient pressure of 110%, 2750 psig, of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, 1968 Edition, which permits a maximum transient pressure of 110%, 2750 psig, of component design pressure. The Safety Limit of 2750 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

The reactor protection system instrumentation Allowable Values specified in Table 2.2-1 have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits.

The shutdown bypass provides for bypassing certain functions of the reactor protection system in order to permit control rod drive tests, zero power PHYSICS TESTS and certain startup and shutdown procedures. The purpose of the shutdown bypass high pressure trip is to prevent normal operation with shut-down bypass activated. This high pressure setpoint is lower than the normal low pressure setpoint so that the reactor must be tripped before the bypass is initiated. The high flux setpoint of ≤5.0% prevents any significant reactor power from being produced. Sufficient natural circulation would be available to remove 5.0% of RATED THERMAL POWER if none of the reactor coolant pumps were operating.

Manual Reactor Trip

The manual reactor trip is a redundant channel to the automatic reactor protection system instrumentation channels and provides manual reactor trip capability.

High Flux

A high flux trip at high power level (neutron flux) provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry.

During normal station operation, reactor trip is initiated when the reactor power level reaches the Allowable Value ≤ 104.9 ~~105.1~~% of rated power. Due to transient overshoot, heat balance, and instrument errors, the maximum actual power at which a trip would be actuated could be at a thermal power of 110.2% of 2817 MWt, ~~112%,~~ which was used in the safety analysis.

LIMITING SAFETY SYSTEM SETTINGS

BASES

RC High Temperature

The RC high temperature trip ≤618°F prevents the reactor outlet temperature from exceeding the design limits and acts as a backup trip for all power excursion transients.

Flux -- ΔFlux/Flow

The power level Allowable Value produced by the reactor coolant system flow is based on a flux-to-flow ratio which has been established to accommodate flow decreasing transients from high power where protection is not provided by the high flux/number of reactor coolant pumps on trips.

The power level Allowable Value produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate.

For safety calculations the instrumentation errors for the power level were used. Full flow rate is defined as the flow calculated by the heat balance at 100% power. At the time of the calibration the RCS flow will be greater than or equal to the value in Table 3.2-2.

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LIMITING SAFETY SYSTEM SETTINGS

BASES

The AXIAL POWER IMBALANCE boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kW/ft limits or DNBR limits. The AXIAL POWER IMBALANCE reduces the power level trip produced by a flux-to-flow ratio such that the boundaries of the figure in the CORE OPERATING LIMITS REPORT are produced.

RC Pressure - Low, High, and Pressure Temperature

The high and low trips are provided to limit the pressure range in which reactor operation is permitted.

During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RC high pressure setpoint is reached before the high flux setpoint. The Allowable Value for RC high pressure, 2355 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RC high pressure trip is backed up by the pressurizer code safety valves for RCS over pressure protection. The RC high pressure trip is, therefore, set lower than the set pressure for these valves, 2500 psig (nominal), even when accounting for the RPS RC pressure instrument string uncertainty. The RC high pressure trip also backs up the high flux trip.

The RC low pressure, 1900.0 psig, and RC pressure-temperature (16.25~~16.00~~T_{out}-8034~~7957.5~~) psig, Allowable Values have been established to maintain the DNB ratio greater than or equal to the minimum allowable DNB ratio for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protecting against DNB.

High Flux/Number of Reactor Coolant Pumps On

In conjunction with the flux - Δflux/flow trip the high flux/number of reactor coolant pumps on trip prevents the minimum core DNBR from decreasing below the minimum allowable DNB ratio by tripping the reactor due to the loss of reactor coolant pump(s). The pump monitors also restrict the power level for the number of pumps in operation.

Containment High Pressure

The Containment High Pressure Allowable Value $\leq$ 4 psig, provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the containment vessel or a loss-of-coolant accident, even in the absence of a RC Low Pressure trip.

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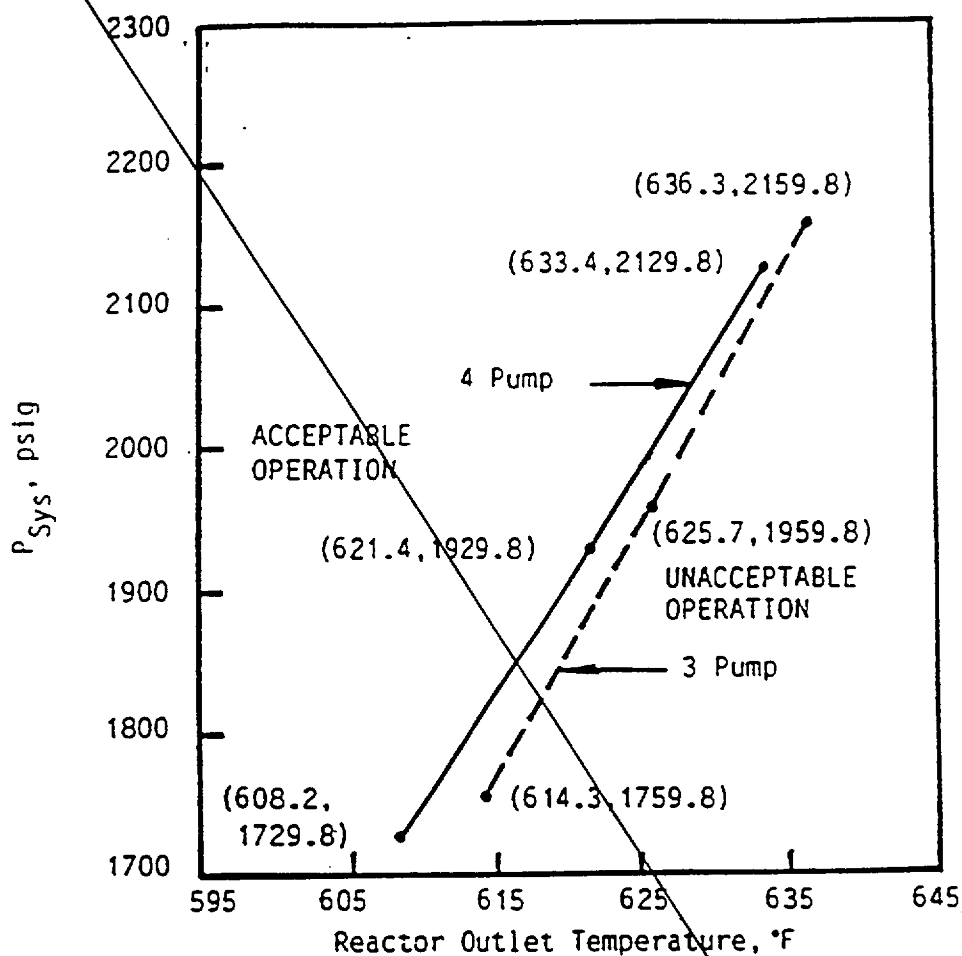
Bases Figure 2.1 Pressure/Temperature Limits at Maximum Allowable Power for Minimum DNBR



Pumps	Flow, gpm	Power	Required Measured Flow to ensure Compliance, gpm
4	380,000	112%	389,500
3	283,860	90.5%	290,957

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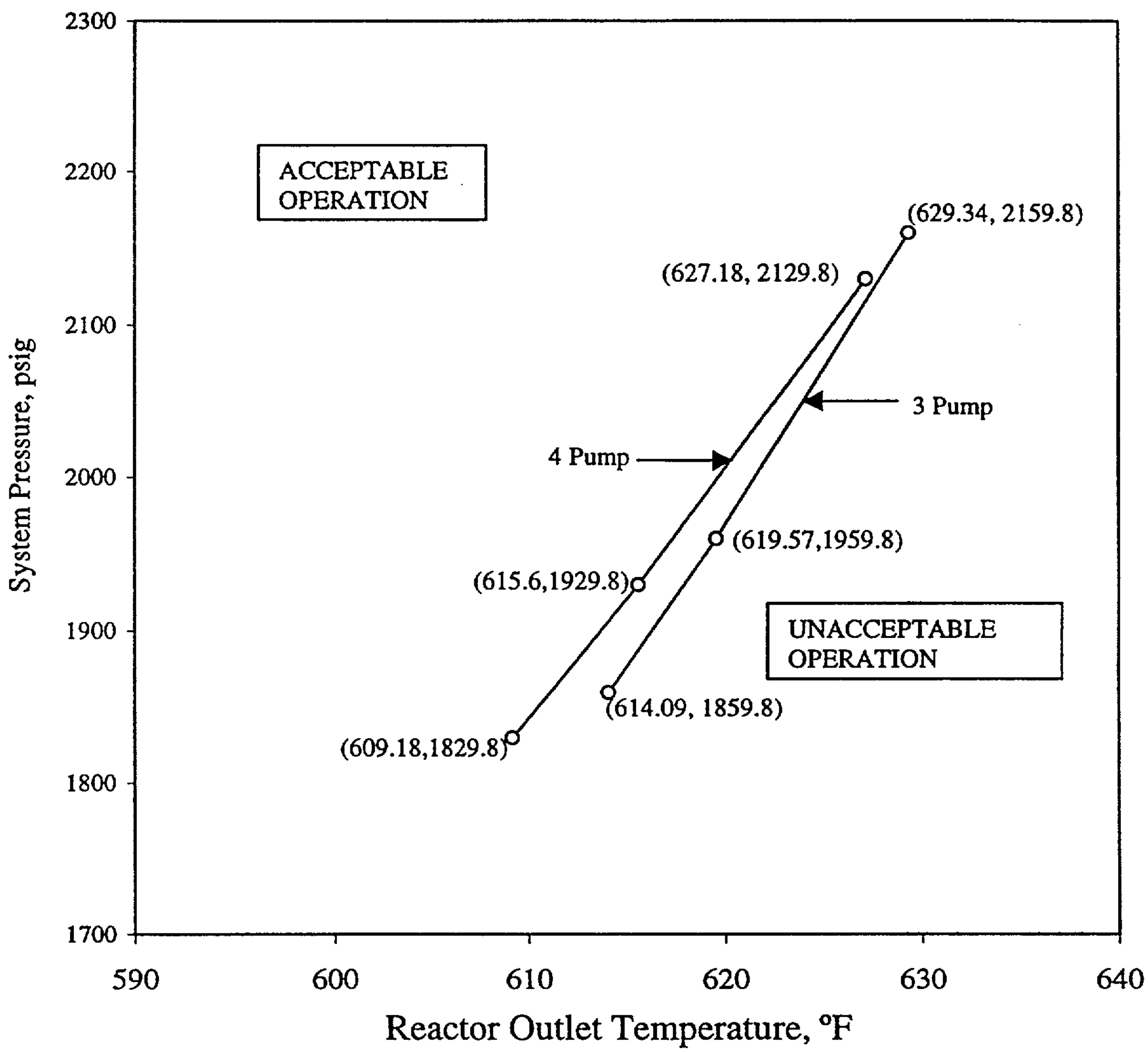
Bases Figure 2.1 Pressure/Temperature Limits at Maximum Allowable Power for Minimum DNBR



Pumps	Flow, gpm	Power	Required Measured Flow to ensure Compliance, gpm
4	380,000	110.2 %	389,500
3	283,860	89 %	290,957

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.9×10^{-4} Δk/k/°F whenever THERMAL POWER is < 80 ~~95~~% of RATED THERMAL POWER,

b. Less positive than 0.0×10^{-4} Δk/k/°F whenever THERMAL POWER is ≥ 80 ~~95~~% of RATED THERMAL POWER, and

c. Equal to or less negative than the limit provided in the CORE OPERATING LIMITS REPORT at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With the moderator temperature coefficient outside any of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading.

b. At any THERMAL POWER, within 7 days after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

---

*With $k_{eff} \geq 1.0$.

#See Special Test Exception 3.10.2.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-2.

a. Reactor Coolant Hot Leg Temperature

b. Reactor Coolant Pressure

c. Reactor Coolant Flow Rate

APPLICABILITY: MODE 1

ACTION:

If any parameter above exceeds its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-2 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

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TABLE 3.2-2

DNB MARGIN

Parameter	Required Measured Parameters with Four Reactor Coolant Pumps Operating	Required Measured Parameters with Three Reactor Coolant Pumps Operating
Reactor Coolant Hot Leg Temperature T_H°F	≤610	≤610[(1)]
Reactor Coolant Pressure, psig.[(2)]	≥2064.8 ~~2062.7~~	≥2060.8 ~~2058.7~~[(1)]
Reactor Coolant Flow Rate, gpm[(3)]	≥389,500	≥290,957

(1) Applicable to the loop with 2 Reactor Coolant Pumps Operating.

(2) Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

(3) These minimum required measured flows include a flow rate uncertainty of 2.5%. ~~, and are based on a minimum of 52 lumped burnable poison rod assemblies in place in the core.~~

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

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REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant system (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2, 3.4-3 and 3.4-4 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 50°F in any one hour period, and

b. A maximum cooldown of 100°F in any one hour period with cold leg temperature $\geq$ 270°F and a maximum cooldown of 50°F in any one hour period with cold leg temperature <270°F.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and be in COLD SHUTDOWN within the following 30 hours.

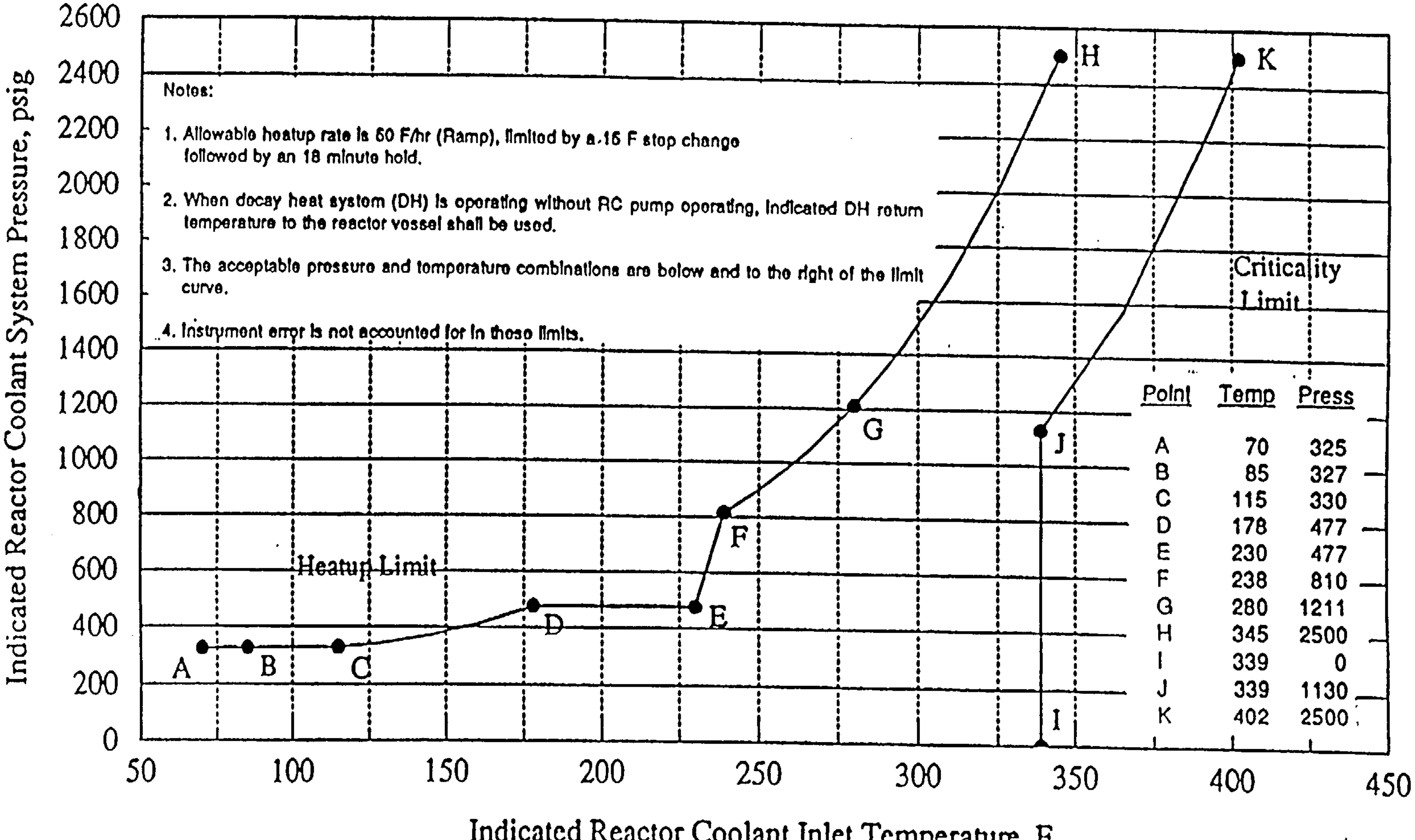
SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens representative of the vessel materials shall be removed and examined, to determine changes in material properties, at the intervals defined in BAW 1543A. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3 and 3.4-4.

Figure 3.4-2

Reactor Coolant System Pressure-Temperature Limits
For Heatup and Core Criticality for the First <u>20</u> ~~21~~ EFPY



Point	Temp	Press
A	70	325
B	85	327
C	115	330
D	178	477
E	230	477
F	238	810
G	280	1211
H	345	2500
I	339	0
J	339	1130
K	402	2500

Figure 3.4-3

Reactor Coolant System Pressure-Temperature Limits
For Cooldown for the First 20 ~~21~~ EFPY

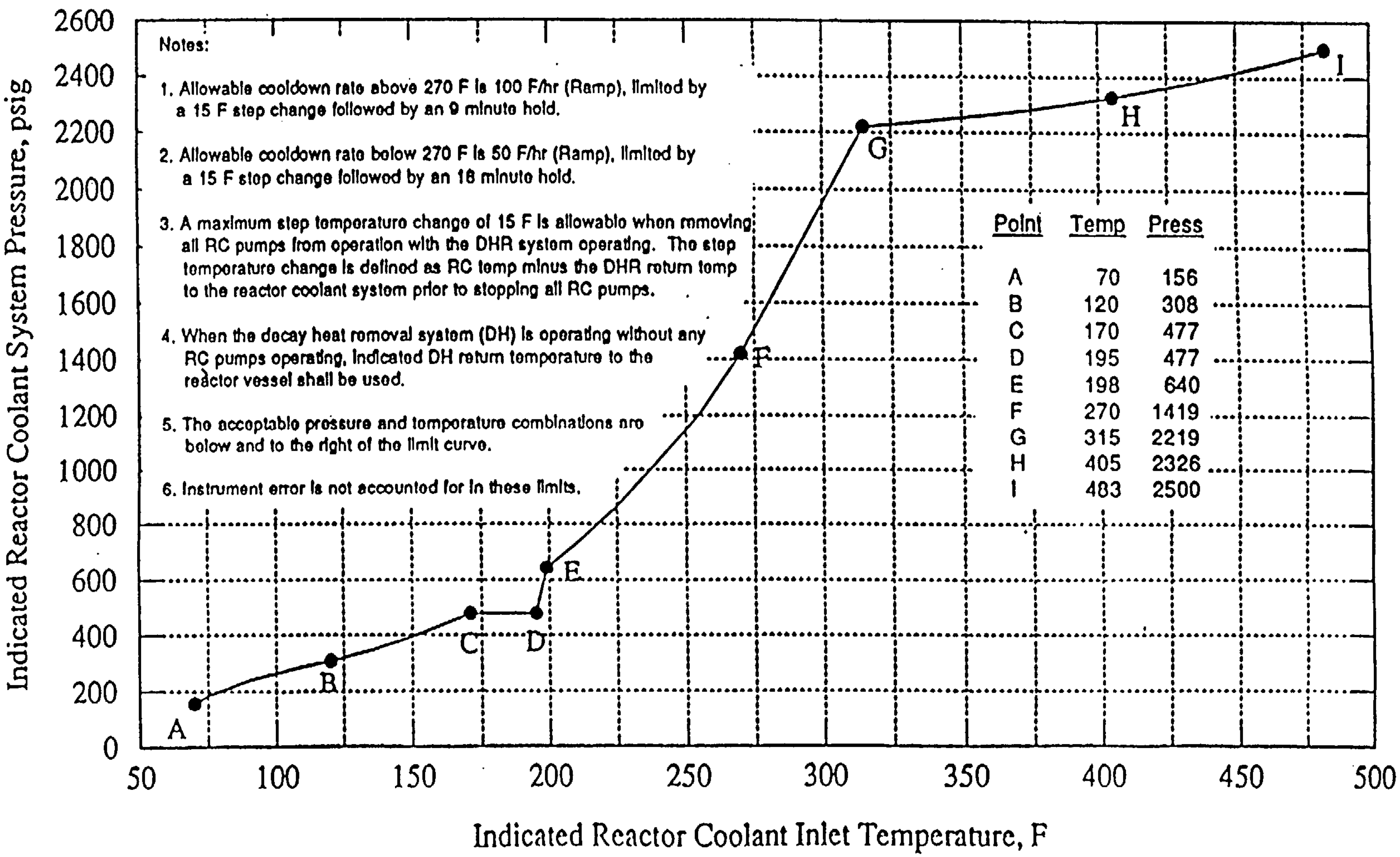
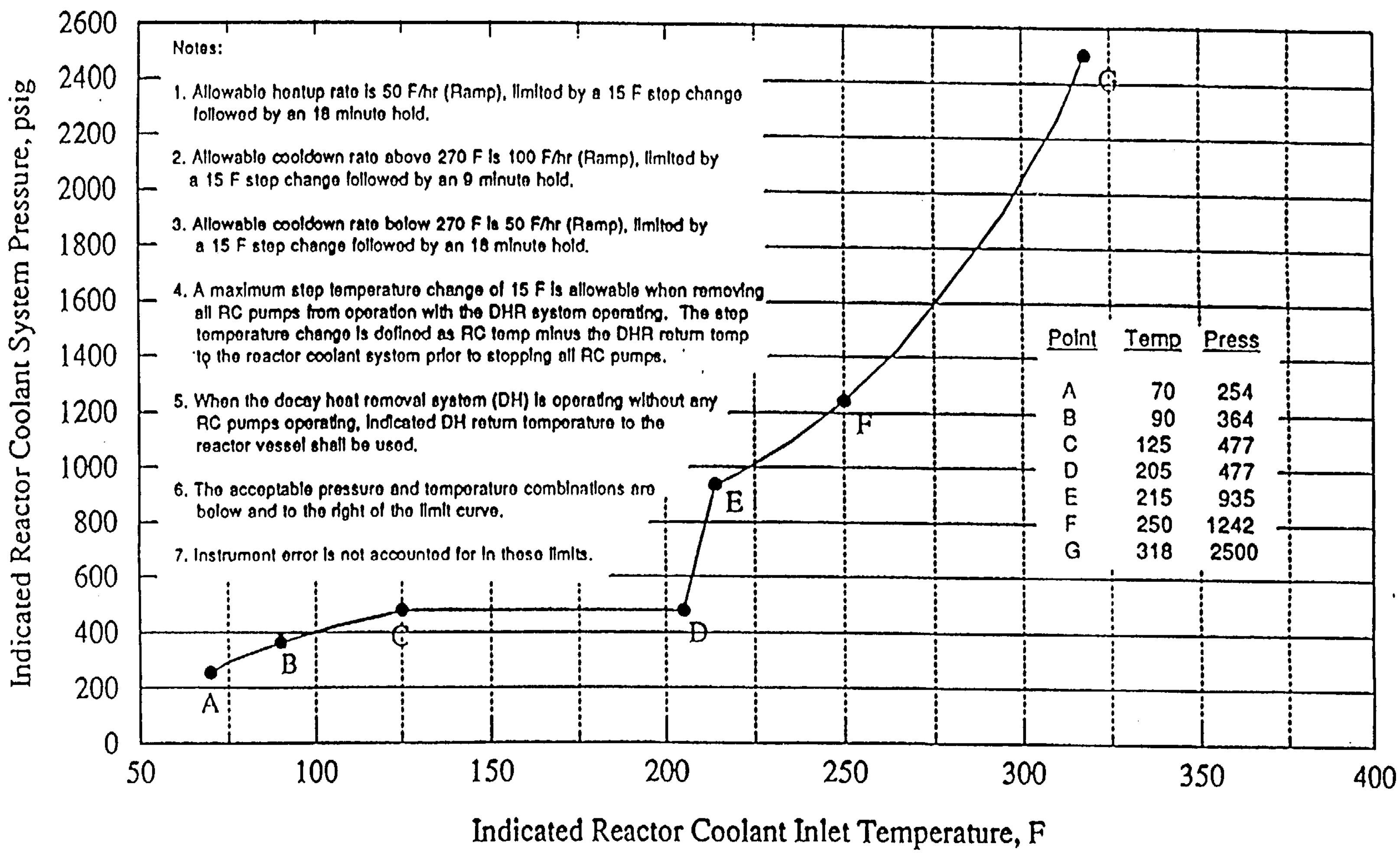


Figure 3.4-4

Reactor Coolant System Pressure-Temperature Heatup and Cooldown Limits for Inservice Leak and Hydrostatic Tests for the First 20 ~~21~~ EFPY



3/4.1.1 BORATION CONTROL

3/4.1.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. During Modes 1 and 2 the SHUTDOWN MARGIN is known to be within limits if all control rods are OPERABLE and withdrawn to or beyond the insertion limit.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature. The SHUTDOWN MARGIN required is consistent with FSAR safety analysis assumptions.

3/4.1.1.2 BORON DILUTION

A minimum flow rate of at least 2800 gpm provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual through the Reactor Coolant System in the core during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 2800 gpm will circulate an equivalent Reactor Coolant System volume of 12,110 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron concentration reduction will be within the capability for operator recognition and control.

In MODE 5 or MODE 6, the RCS boron concentration is typically somewhat higher than the boron concentration required by Specification 3.1.1.1 (MODE 5) or Specification 3.9.1 (MODE 6), and could be higher than the boron concentration of normal sources of water addition. At reduced inventory conditions in the RCS, in order to reduce the possibility of vortexing, the flowrate through the decay heat system may be procedurally restricted to somewhat less than 2800 gpm. In this situation, if water with a boron concentration equal to or greater than the boron concentration associated with the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 (MODE 5), or the boron concentration corresponding to the more restrictive reactivity condition specified in Specification 3.9.1 (MODE 6), is added to the RCS, the RCS boron concentration is assured to remain above the minimum boron concentration associated with the Specification 3.1.1.1 or Specification 3.9.1 requirement, and a flowrate of less than 2800 gpm is not of concern.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirement for measurement of the MTC each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurance that the coefficient will be maintained within acceptable values throughout each fuel cycle.

b. The measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by 5 percent to account for measurement error.

For Condition II events, the core is protected from exceeding the values given in the bases to specification 2.1 locally, and from going below the minimum allowable DNB ratio by automatic protection on power, AXIAL POWER IMBALANCE pressure and temperature. Only conditions 1 through 3, above, are mandatory since the AXIAL POWER IMBALANCE is an explicit input to the reactor protection system.

The QUADRANT POWER TILT limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The QUADRANT POWER TILT limit at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. In the event the tilt is not corrected, the margin for uncertainty on F_Q is reinstated by reducing the power by 2 percent for each percent of tilt in excess of the limit.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the FSAR initial assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR greater than the minimum allowable DNB ratio throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate using delta P instrumentation is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

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The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 μCi/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation during one continuous time interval with specific activity levels exceeding 1.0 μCi/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 48 hours since the activity levels allowed by Figure 3.4-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture.

Reducing T_{avg} to < 530°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves.

The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The pressure-temperature limits of the reactor coolant pressure boundary are established in accordance with the requirements of Appendix G to 10 CFR 50 and with the thermal and loading cycles used for design purposes.

The limitations prevent non-ductile failure during normal operation, including anticipated operational occurrences and system hydrostatic tests. The limits also prevent exceeding stress limits during cyclic operation. The loading conditions of interest include:

1. Normal operations, including heatup and cooldown,

2. Inservice leak and hydrostatic tests, and

3. Reactor core operation.

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10 CFR 50. The closure head region, reactor vessel outlet nozzles and the beltline region have been identified to be the only regions of the reactor vessel, and consequently of the reactor coolant pressure boundary, that determine the pressure-temperature limitations concerning non-ductile failure.

REACTOR COOLANT SYSTEM

BASES

The closure head region is significantly stressed at relatively low temperatures (due to mechanical loads resulting from bolt pre-load). This region largely controls the pressure-temperature limitations of the first several service periods. The outlet nozzles of the reactor vessel also affect the pressure-temperature limit curves of the first several service periods. This is due to the high local stresses at the inside corner of the nozzle which can be two to three times the membrane stresses of the shell. After the first several years of neutron radiation exposure, the RT_{NDT} temperature of the beltline region materials will be high enough so that the beltline region of the reactor vessel will start to control the pressure-temperature limitations of the reactor coolant pressure boundary. For the service period for which the limit curves, are established, the maximum allowable pressure as a function of fluid temperature is obtained through a point-by-point comparison of the limits imposed by the closure head region, outlet nozzles, and beltline region. The maximum allowable pressure is taken to be the lower pressure of the three calculated pressures. The pressure limit is adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all operating reactor coolant pump combinations. The limit curves were prepared based upon the most limiting adjusted reference temperature of all the beltline region materials at the end of twenty-~~one~~ effective full power years.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10 CFR 50, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside the radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The limit curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

REACTOR COOLANT SYSTEM

BASES

The unirradiated transverse impact properties of the beltline region materials, required by Appendices G and H to 10 CFR 50, were determined for those materials for which sufficient amounts of material were available. The adjusted reference temperatures are calculated by adding the predicted radiation-induced ΔRT_{NDT} and the unirradiated RT_{NDT}. The procedures described in Regulatory Guide 1.99, Rev. 2, were used for predicting the radiation induced ΔRT_{NDT} as a function of the material's copper and nickel content and neutron fluence.

Figure 3.4-2 presents the pressure-temperature limit curve for normal heatup. This figure also presents the core criticality limits as required by Appendix G to 10 CFR 50. Figure 3.4-3 presents the pressure-temperature limit curve for normal cooldown. Figure 3.4-4 presents the pressure-temperature limit curves for heatup and cooldown for inservice leak and hydrostatic testing.

All pressure-temperature limit curve are applicable up to twenty-~~one~~ effective full power years. The protection against non-ductile failure is assured by maintaining the coolant pressure below the upper limits of Figures 3.4-2, 3.4-3 and 3.4-4.

REACTOR COOLANT SYSTEM

BASES

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in BAW 1543A. The withdrawal schedule is based on four considerations: (a) uncover possible technical anomalies as early in life as they can be detected (end of first fuel cycle), (b) define the material properties needed to perform the analysis required by Appendix G to 10 CFR 50, (c) reserve two capsules for evaluation of the effectiveness of thermal annealing in the event inplace annealing becomes necessary, (d) provide material property data corresponding to the reactor vessel beltline conditions at the end of service. This withdrawal schedule is specified to assure compliance with the requirements of Appendix H to 10 CFR 50. Appendix H references the requirements of ASTM E185 for surveillance program criteria.

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microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics, shutdown experience and challenges to the Pressurizer Pilot Operated Relief Valve (PORV) and the Pressurizer Code Safety Valves shall be submitted on a monthly basis to arrive no later than the 15th of each month following the calendar month covered by the report, as follows: The signed original to the Nuclear Regulatory Commission, Document Control Desk, Washington, D. C. 20555, and one copy each to the Region III Administrator and the Davis-Besse Resident Inspector.

CORE OPERATING LIMITS REPORT

6.9.1.7 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle and any remaining part of a reload cycle for the following:

2.1.2	AXIAL POWER IMBALANCE Protective Limits for Reactor Core Specification 2.1.2
2.2.1	Trip Setpoint for Flux --ΔFlux/Flow for Reactor Protection System Setpoints Specification 2.2.1
3.1.1.3c	Negative Moderator Temperature Coefficient Limit
3.1.3.6	Regulating Rod Insertion Limits
3.1.3.7	Rod Program
3.1.3.8	Xenon Reactivity
3.1.3.9	Axial Power Shaping Rod Insertion Limits
3.2.1	AXIAL POWER IMBALANCE
3.2.2	Nuclear Heat Flux Hot Channel Factor, F_Q
3.2.3	Nuclear Enthalpy Rise Hot Channel Factor, $F^N_{\Delta H}$
3.2.4	QUADRANT POWER TILT

The analytical methods used to determine the core operating limits addressed by the individual Technical Specifications shall be: those previously reviewed and approved by the NRC, as described in BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses", or any other new NRC-approved analytical methods used to determine core operating limits that are not yet referenced in the applicable approved revision of BAW-10179P-A. The applicable approved revision number for BAW-10179P-A at the time the reload analyses are performed shall be identified in the CORE OPERATING LIMITS REPORT. The CORE OPERATING LIMITS REPORT shall also list any new NRC-approved analytical methods used to determine core operating limits that are not yet referenced in the applicable approved revision of BAW-10179P-A.

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CORE OPERATING LIMITS REPORT (Continued)

As described in reference documents listed in accordance with the instructions given above, when an initial assumed power level of 102% of rated thermal power is specified in a previously approved method, 100.37% of rated thermal power may be used when input for reactor thermal power measurement of feedwater mass flow is by the Leading Edge Flow Meter (LEFM) CheckPlus™ System.

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

The CORE OPERATING LIMITS REPORT, including any mid-cycle revision or supplements thereto, shall be provided upon issuance for each reload cycle to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

ENVIRONMENTAL CONSIDERATION FOR LICENSE AMENDMENT REQUEST NUMBER 00-0006

Identification of Proposed Action

This proposed action involves the Davis-Besse Nuclear Power Station (DBNPS), Unit 1, Operating License Number NPF-3, including the attached Appendix A, Technical Specifications (TS). Specifically, the proposed license amendment application involves: Operating License (OL) paragraph 2.C(1), Maximum Power Level; OL paragraph 2.C(3)(d), Additional Conditions; Technical Specification (TS) 1.3, Definitions – Rated Thermal Power; TS 2.1.1, Safety Limits – Reactor Core, and associated Bases; TS 2.2.1, Limiting Safety System Settings – Reactor Protection System Setpoints, and associated Bases; TS 3/4.1.1.3, Reactivity Control Systems – Moderator Temperature Coefficient; TS 3/4.2.5, Power Distribution Limits – DNB Parameters; TS 3/4.4.9.1, Reactor Coolant System – Pressure/Temperature Limits, and associated Bases; and TS 6.9.1.7, Core Operating Limits Report.

The proposed amendment would increase the authorized rated thermal power from 2772 MWt to 2817 MWt (approximately 1.63%), based on the use of Caldon Inc. Leading Edge Flow Meter (LEFM) instrumentation to improve the accuracy of the feedwater mass flow input to the plant power calorimetric measurement. The DBNPS plans to install the LEFM CheckPlus™ System in both feedwater trains in the upcoming Thirteenth Refueling Outage (13RFO).

The proposed amendment would also revise the moderator temperature coefficient requirements listed in TS Limiting Condition for Operation (LCO) 3.1.1.3, revise the Departure from Nucleate Boiling (DNB) parameters listed in TS Table 3.2-2 relating to reactor coolant pressure, and make a clarification to a note in the same table. These changes are unrelated to the power uprate changes.

Need for the Proposed Action

The proposed amendment would increase the authorized rated thermal power, and, correspondingly, increase the electrical generation capability of the DBNPS. This increased capability will allow the DBNPS to better meet the needs of its customers. In addition, the proposed amendment would update TS parameters consistent with the latest Loss of Coolant Accident (LOCA) analyses.

Environmental Impact of the Proposed Action

As described in the Safety Assessment and Significant Hazards Consideration (SASHC) for the proposed license amendment application, the DBNPS has determined that the structures,

systems, and components which could be affected by the proposed license amendment, will continue to be capable of performing their safety functions.

The proposed license amendment application involves a change to a requirement with respect to the use of plant components located within the restricted area as defined in 10 CFR Part 20. As concluded in the SASHC, this proposed license amendment does not involve a significant hazards consideration. In addition, as described in further detail below, the proposed changes do not involve a significant change in the types or a significant increase in the amounts of any radiological effluents that may be allowed to be released offsite. Furthermore, as also described in further detail below, there is no significant increase in the individual or cumulative occupational radiation exposure.

The solid waste generation volume is not expected to change significantly as a result of the proposed changes. The proposed changes do not appreciably impact installed equipment performance and do not require drastic changes in system operation. In addition, the reactor coolant activity will not change appreciably, and maintenance and operational practices are not expected to change, therefore, the specific activity of solid waste is not expected to change.

The gaseous and liquid effluent releases are expected to increase from current values by no more than the percentage increase in power level. Effluent releases will continue to be controlled in accordance with the DBNPS Offsite Dose Calculation Manual (ODCM), ensuring that the resultant offsite doses are in compliance with current regulatory requirements. The ODCM describes the methodology and parameters used in: determining the radioactive material release rates and cumulative releases; calculating the radioactive liquid and gaseous effluent monitoring instrumentation alarm/trip setpoints; and calculating the corresponding dose rates and cumulative quarterly and yearly doses. The ODCM also describes and provides requirements for the Radiological Environmental Monitoring Program. Sampling locations, media and collection frequencies, and analytical requirements are specified in the ODCM.

The specific activity of the primary and secondary coolant during normal operation will increase in approximately the same proportion as the proposed power increase, but are bounded by the current design source terms, which are based on a power level of 2772 MWt and a 12-month operating cycle. In addition, the specific activity of the primary and secondary coolant will still be subject to the existing Technical Specification limits. The proposed changes will not cause radiological exposure in excess of the dose criteria (for restricted and unrestricted access) provided in the current 10 CFR 20. Radiation levels in the plant are expected to increase by no more than the percentage increase in power level. Individual worker exposures will be maintained within acceptable limits by the site as-low-as reasonably-achievable (ALARA) program.

The radiological accident analyses presented in the DBNPS USAR are based on a power level of 2827 MWt (102% of the current licensed power level), and therefore bound the

proposed power level increase. These analyses demonstrate that the dose limits set by 10 CFR 100 and 10 CFR 50, Appendix A, General Design Criterion 19 for the site boundary and control room, respectively, are met.

With regard to potential non-radiological impacts, as described in further detail below, the proposed license amendment involves no significant increase in the amounts or change in the types of any non-radiological effluents that may be released offsite, and has no other environmental impact.

A closed-cycle cooling system, the Circulating Water System (CWS), which includes a natural draft cooling tower, limits the thermal discharge to Lake Erie during normal plant operation. Waste heat is transferred to the atmosphere via evaporation of some of the hot cooling water entering the cooling tower and via sensible heating of the ambient air flowing up through the tower. Except for a portion of flow (cooling tower blowdown) which is returned to Lake Erie in order to maintain the CWS chemistry, the cooled water is recirculated back through the plant. Makeup water to replace water lost through evaporation and cooling tower blowdown comes from Lake Erie via the Service Water System.

The waste heat load to the CWS will increase in approximately the same proportion as the proposed power increase. The maximum CWS outlet temperature increase due to the proposed changes will be approximately 0.5 °F. This increase can be accommodated by the cooling tower. As a result, the peak difference between the Lake Erie water temperature and the cooling tower blowdown temperature is not affected by the power uprate. In addition, due to the increased heat load, a slight increase in cooling tower evaporation rate will occur, requiring an increase in makeup flow rates. Less than a 2% increase is expected. The DBNPS is subject to the monitoring requirements of the Environmental Protection Agency as delineated under the National Pollutant Discharge Elimination System (NPDES) program, which does not place any absolute operating limits on either flow or temperature for the discharge into Lake Erie.

Based on the above, the DBNPS concludes that there are no significant radiological or non-radiological environmental impacts associated with the proposed license amendment.

Alternatives to the Proposed Action

Since the DBNPS has concluded that the environmental effects of the proposed action are not significant, any alternatives will have only similar or greater environmental impacts. The principal alternative would be to not grant the license amendment. Since the environmental impacts of the proposed action are not significant, denial of the proposed license amendment would not significantly reduce the environmental impacts attributable to the plant.

Alternative Use of Resources

This action does not involve the use of resources not previously considered in the Final Environmental Statement Related to the Operation of the Davis-Besse Nuclear Power Station, Unit Number 1 (NUREG 75/097).

Finding of No Significant Impact

The DBNPS has reviewed the proposed license amendment against the categorical exclusion criteria of 10 CFR 51.22(c)(9) for an environmental assessment. As demonstrated in the proposed license amendment's SASHC, the proposed changes do not involve a significant hazards consideration. In addition, the proposed changes do not significantly change the types or significantly increase the amounts of effluents that may be released offsite, and do not significantly increase individual or cumulative occupational radiation exposures. Accordingly, the DBNPS finds that the proposed license amendment, if approved by the Nuclear Regulatory Commission, will have no significant impact on the environment and that no environmental assessment is required.

Docket Number 50-346
License Number NPF-3
Serial Number 2692
Enclosure 1
Attachment 3

DAVIS-BESSE NUCLEAR POWER STATION
LICENSING REPORT
POWER UPRATE PROGRAM

(82 pages follow)

FENOC

FirstEnergy Nuclear Operating Company

DAVIS-BESSE NUCLEAR POWER STATION

LICENSING REPORT

POWER UPRATE PROGRAM

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AC	Alternating Current
AFW	Auxiliary Feedwater
AFWS	Auxiliary Feedwater System
ALARA	As Low As Reasonably Achievable
AMSAC	ATWS Mitigation System Actuation Circuitry
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARTS	Anticipatory Reactor Trip System
ASME	American Society of Mechanical Engineers
ATSI	American Transmission System Inc.
ATWS	Anticipated Transient Without Scram
BHP	Brake Horsepower
B&PV	Boiler and Pressure Vessel
BOC	Beginning Of Cycle
BOP	Balance Of Plant
B&W	Babcock & Wilcox
BWC	Babcock & Wilcox Correlation
B&WOG	Babcock & Wilcox Owners Group
BWST	Borated Water Storage Tank
C&FS	Condensate and Feedwater System
CAC	Containment Air Coolers
CCW	Component Cooling Water
CEMAP	Corrosion/Erosion Monitoring and Analysis Program
CF	Core Flooding System
CFR	Code of Federal Regulations
CHF	Critical Heat Flux
CRA	Control Rod Assembly
CRDCS	Control Rod Drive Control System
CRDM	Control Rod Drive Mechanism
CST	Condensate Storage Tank
CT	Circulating Water System
CWA	Cold Water Accident
DBNPS	Davis-Besse Nuclear Power Station
DC	Direct Current
degF	degree Fahrenheit
DFG	Diode Function Generator
DHR	Decay Heat Removal System
DHR/LPI	Decay Heat Removal/Low Pressure Injection
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DSS	Diverse Scram System
ECCS	Emergency Core Cooling System
EFPY	Effective Full-Power Year
EOC	End Of Cycle
EPRI	Electric Power Research Institute
EQ	Equipment Qualification

LIST OF ABBREVIATIONS

ES	Engineered Safeguards
ESF	Engineered Safety Feature
FAC	Flow Accelerated Corrosion
FENOC	FirstEnergy Nuclear Operating Company
FIV	Flow Induced Vibration
FP	Full Power
FRA-ANP	Framatome-ANP
GDC	General Design Criterion
GL	Generic Letter
GPM	Gallons Per Hour
HDS	Heater Drain System
HELB	High Energy Line Break
HP	High Pressure
HPI	High Pressure Injection System
ICS	Integrated Control System
IEEE	Institute of Electrical and Electronic Engineers
IPE	Independent Plant Examination
KV	Kilovolt
KVA	Kilovolt-ampere
KW	Kilowatt
kpph	Thousand Pounds Per Hour
LBB	Leak Before Break
LBLOCA	Large-Break Loss-Of-Coolant Accident
LEFM	Leading Edge Flow Meter
LOCA	Loss-Of-Coolant Accident
LOCF	Loss of Coolant Flow
LOFW	Loss of Feedwater
LOOP	Loss of Offsite Power
LP	Low Pressure
LPI	Low Pressure Injection
LTCC	Long-Term Core Cooling
LTOP	Low Temperature Overpressure Protection System
M_{steam}	Steam Mass Flow Rate
MAAP	Modular Accident Analysis Program
MCR	Maximum Continuous Rating
MDA	Moderator Dilution Accident
MFCV	Main Feedwater Control Valve
MFLB	Main Feedwater Line Break
MFP	Main Feedwater Pump
MFSV	Main Feedwater Stop Valve
MOL	Middle of Life
MOV	Motor Operated Valve
MPT	Main Power Transformer
MSAVV	Main Steam Atmospheric Vent Valve
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break

LIST OF ABBREVIATIONS

MSNRV	Main Steam Non-Return Valve
MSR	Moisture Separator Reheater
MSS	Main Steam System
MSSV	Main Steam Safety Valve
MSTBV	Main Steam Turbine Bypass Valve
MTC	Moderator Temperature Coefficient
MUT	Makeup Tank
MU&P	Make-up & Purification
MVA	Megavolt-ampere
MWe	Megawatt Electric
MWt	Megawatt Thermal
NERC	National Electric Reliability Council
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
ODCM	Offsite Dose Calculation Manual
OL	Operating License
OTSG	Once through Steam Generator
P_{OTSG}	OTSG Pressure
P_{steam}	Steam Pressure
PORV	Pilot-Operated Relief Valve
PRA	Probabilistic Risk Assessment
psia	pounds per square inch absolute
psig	pounds per square inch gauge
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
RAI	Request for Additional Information
RC	Reactor Coolant
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPS	Reactor Protection System
RTD	Resistance Temperature Detector
RTP	Rated Thermal Power
RT_{PTS}	Pressurized Thermal Shock Reference Temperature
RV	Reactor Vessel
SBLOCA	Small-Break Loss-Of-Coolant Accident
SBO	Station Blackout
SBODG	Station Blackout Diesel Generator
SCC	Stress Corrosion Cracking
SCD	Statistical Core Design
SFAS	Safety Features Actuation System
SFRCS	Steam and Feedwater Rupture Control System
SG	Steam Generator
SGTR	Steam Generator Tube Rupture

LIST OF ABBREVIATIONS

SJAE	Steam Jet Air Ejector
SPDS	Safety Parameter Display System
SSE	Safe Shutdown Earthquake
T_{avg}	Vessel Average Temperature
T_{cold}	Vessel/Core Inlet Temperature
T_{fw}	Feedwater Temperature
T_{hot}	Vessel Outlet Temperature
T_{sat}	Saturation Temperature
T_{steam}	Steam Temperature
TPCW	Turbine Plant Cooling Water
TRM	Technical Requirements Manual
TS	Technical Specification
UHS	Ultimate Heat Sink
USAR	Updated Safety Analysis Report
VLPT	Variable Low Pressure Trip
V&V	Verification and Validation
W_{steam}	Steam Flow
Zr-4	Zircaloy-4
#/hr	pounds per hour
°F	degree Fahrenheit

1.0 BACKGROUND AND REASON FOR THE PROPOSED CHANGE

The FirstEnergy Nuclear Operating Company (FENOC) Davis-Besse Nuclear Power Station (DBNPS) has plans to install two Caldon Inc. Leading Edge Flow Meter (LEFM) CheckPlus™ feedwater flow meters. Analysis demonstrates that the uncertainty on feedwater flow and temperature attained via use of these flow meters is reduced such that the core thermal power uncertainty is reduced to 0.37%. This uncertainty improvement translates directly into a 1.63% thermal power increase from 2772 MWt to 2817 MWt. However, much of the safety analyses and evaluations to support this power increase conservatively assumed an increased bounding allowable reactor thermal power of 1.7% (2819 MWt).

The 1.63-percent core power uprate for the DBNPS is based on eliminating unnecessary analytical margin originally required of emergency core cooling system (ECCS) evaluation models performed in accordance with the requirements set forth in the Code of Federal Regulations (CFR) 10CFR50, Appendix K, ECCS Evaluation Models.

The Nuclear Regulatory Commission (NRC) approved a change to the requirements of 10CFR50, Appendix K on June 1, 2000 (Federal Register (FR) 65 FR 34913). The change provides licensees with the option of maintaining the 2-percent power margin between the licensed power level and the assumed power level for the ECCS evaluation, or applying a reduced margin for ECCS evaluation. For the reduced margin for ECCS evaluation case, the proposed alternative reduced margin has been demonstrated to account for uncertainties due to power level instrumentation error. Based on the proposed use of the LEFM CheckPlus™ instrumentation to determine core power level with a power measurement uncertainty of less than 0.37 percent, it is proposed to reduce the licensed power uncertainty required by 10CFR50, Appendix K, for modest increases of up to 1.63 percent in the license power level using current NRC-approved methodologies.

The basis for the amendment request is that the Caldon LEFM CheckPlus™ instrumentation provides a more accurate indication of feedwater flow and temperature, and, correspondingly, reactor thermal power, than assumed during the development of Appendix K requirements. Complete technical support for this conclusion is discussed in detail in Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0, March 1997, as approved in the NRC's Safety Evaluation for the Comanche Peak Steam Electric Station Units 1 and 2, dated March 8, 1999. Topical Report ER-80P is supplemented by Caldon Engineering Report ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM√™ or LEFM CheckPlus™ System." The improved thermal power measurement accuracy obviates the need for the full 2-percent power margin assumed in Appendix K, thereby increasing the thermal power available for electrical generation.

It should be noted that the proposed power increase is supported, in part, by the use of Statistical Core Design (SCD) methodology, to demonstrate adequate departure from nucleate boiling (DNB) margin. The SCD methodology is described in BAW-10187P-A, "Statistical Core Design for B&W-Designed 177-FA Plants," B&W Fuel Company, Lynchburg, Virginia, March, 1994.

The desired power increase of 1.63 percent will be accomplished by increasing the electrical demand on the turbine generator. As a result of this demand increase, steam flow will increase and the resultant steam temperature will decrease. The reactor coolant system (RCS) average temperature will be maintained. The RCS hot leg temperature will increase and the cold leg temperature will decrease in response to the increased steam flow demand.

Procedures for maintenance and calibration of the LEFM CheckPlus™ system will be developed for the DBNPS based on the vendor's recommendations. In addition, the DBNPS Updated Safety Analysis Report (USAR) Technical Requirements Manual (TRM) will be updated to address the requirements to be followed should the LEFM Check Plus system become unavailable. Additional detail is provided by separate attachment (Enclosure 1 Attachment 9), Response to Question 2 (TXX-99203).

2.0 DESCRIPTION OF THE PROPOSED CHANGE

This Licensing Report supports the proposed license amendment to revise the DBNPS Operating License (OL) and Technical Specifications (TSs) to allow an increase in the core power level by 1.63 percent to 2817 MWt. The proposed power uprate is based on the use of the Caldon Leading Edge Flow Meter (LEFM) CheckPlus™ system for determination of main feedwater flow and the associated determination of reactor power through the performance of the power calorimetric. Markups of the proposed OL and TS changes are provided by separate attachment to the license amendment application.

3.0 SAFETY ANALYSIS

3.1 APPROACH

The uncertainty analysis of the Caldon LEFM CheckPlus™ system demonstrates that a power uprate of 1.63% to 2817 MWt can be achieved. The supporting analyses have been conservatively performed for a 1.7% power uprate (2819 MWt). The methodology used addressed the following categories: Nuclear Steam Supply System (NSSS) performance parameters, design transients, systems, components, accidents, and nuclear fuel as well as interfaces between the NSSS and balance-of-plant (BOP) systems. The methodologies use well-defined analysis input assumptions/parameter values and currently approved analytical techniques, and take into consideration applicable licensing criteria and standards.

Generally, no new analytical techniques were used to support the power uprate project. In a couple of areas, Once-Through Steam Generator (OTSG) flow induced vibration (FIV), and the development of mass and energy release following a LOCA and MSLB events, different methods from what is reported in the USAR were used. For the OTSG tubes, the integrity of the tubes, virgin, sleeved or stabilized were re-assessed using the latest techniques. The mass and energy (M&E) release rates for LOCA and MSLB were generated entirely using the RELAP5/MOD2-B&W computer code. RELAP5 has been approved for developing the system response to blowdown for the various postulated transients. The break flow models, while different from the calculations presented in the USAR, have also been approved. RELAP5 has not specifically been approved for generating the M&E data, but the same models and

conservative inputs that were used to generate the system response were applied to the M&E calculation. In addition, the M&E release data also included the effects of the Core Flood Tank noncondensible gas (LOCA) and the break flow stagnation energy (LOCA and MSLB). Therefore, a conservative calculation was performed.

Section 3.2 of this report provides details on the LEFM instrumentation. Section 3.3 of this report discusses the revised NSSS design thermal and hydraulic parameters that were modified as a result of the power uprate and that serve as the basis for all of the NSSS analyses and evaluations. Section 3.4 concludes that no design transient modifications are required to accommodate the revised NSSS design conditions. Sections 3.5 through 3.7 present the systems (e.g., safety injection, decay heat removal (DHR), and control systems) and components (e.g., reactor vessel, pressurizer, reactor coolant pumps (RCPs), steam generators, and NSSS auxiliary equipment) evaluations completed for the revised design conditions. Section 3.8 summarizes the effects of the uprate on the BOP (secondary) systems based upon a heat balance evaluation. Section 3.9 provides an analysis of the effects of the power uprate on the DBNPS electrical power systems. Section 3.10 provides the results of the accident analyses and evaluations performed for the steam generator tube rupture, loss-of-coolant-accident (LOCA), and non-LOCA areas. Sections 3.11 and 3.12 summarize the containment accident analyses and evaluations and the radiological consequence evaluations. Section 3.13 contains the results of the fuel-related analyses. The results of all of the analyses and evaluations performed demonstrate that all acceptance criteria continue to be met.

3.1.1 General Licensing Approach for Plant Analyses Using Plant Power Level

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

First, some analyses apply an explicit 2-percent increase to the initial condition power level to account solely for the power measurement uncertainty. These analyses have not been re-performed for the proposed power uprate because the sum of the increased core power level and the decreased power measurement uncertainty via use of the LEFM system falls within the previously analyzed conditions. The power calorimetric uncertainty calculation described in Section 3.10.4 indicates that with the LEFM CheckPlus™ devices installed, the power measurement uncertainty (based on a 95-percent probability at a 95-percent confidence interval) is less than 0.37 percent. Therefore, these analyses only need to reflect a 0.37-percent power measurement uncertainty. Accordingly, the existing 2-percent uncertainty can be allocated such that 1.63 percent is applied to provide sufficient margin to address the uprate to 2816.6 MWt (rounded to 2817 MWt throughout this report), and 0.37 percent is retained in the analysis to still account for the power measurement uncertainty.

Second, some analyses employ a nominal initial condition power level. These analyses have either been evaluated or re-performed for the proposed increased power level. The results demonstrate that the applicable analysis acceptance criteria continue to be met.

Third, some of the analyses already employ an initial condition power level in excess of the proposed 2,817 MWt. These analyses were previously performed at a higher power level as part of prior plant programs. For these analyses, some of this available margin has been used to offset the proposed power uprate. Consequently, the analyses have been evaluated to confirm that sufficient analysis margin exists to envelope the proposed power uprate.

Fourth, some of the analyses are performed at zero-percent initial condition power conditions or do not actually model the core power level. Consequently, these analyses have not been re-performed since they are unaffected by the core power level.

3.2 LEADING EDGE FLOW METER

The proposed power uprate is based on the use of the Caldon LEFM CheckPlus™ equipment for determination of main feedwater flow and the associated determination of reactor power through the performance of a routine calorimetric. The LEFM CheckPlus™ is an improved system for use in determining and monitoring feedwater flow and temperature. The LEFM CheckPlus™ provides on-line verification of the accuracy of the feedwater flow and temperature measurements upon which NSSS thermal power determinations are based. In addition, the LEFM CheckPlus™ provides a significant improvement in accuracy and an increase in reliability of flow and temperature measurements.

The LEFM CheckPlus™ ultrasonic flow meter consists of an electronic cabinet and a measurement section (spool piece) located in each of the two 18-inch main feedwater lines. The measurement section holds sixteen ultrasonic transducer assemblies that are secured in their own transducer housing, which forms the pressure boundary. Each transducer may be removed at full-power conditions without disturbing the pressure boundary. The LEFM CheckPlus™ uses acoustic energy pulses to determine the final feedwater mass flow rate. Transducers that transmit and receive the pulses are mounted in the LEFM CheckPlus™ spool piece at an angle of 45 degrees to the flow. The sound will travel faster when the pulse traverses the pipe with the flow and slower when the pulse traverses the pipe against the flow. The LEFM CheckPlus™ uses these transit times and time differences between pulses to determine the fluid velocity and temperature. The system uses a single digital system controlled by software to employ the ultrasonic transit time method to measure four-line integral velocities in each of two orthogonal planes at precise locations with respect to the pipe centerline. The system numerically integrates the four velocities in each plane measured, according to the method described in Caldon Topical Report ER-80P, as supplemented by Caldon Engineering Report ER-157P. Although its use for calorimetric input is not nuclear safety related, the CheckPlus™ system's software has been developed and will be maintained under a verification and validation (V&V) program. The V&V program has been applied to all system software and hardware, and includes a detailed code review. The mass flow rate is displayed on the local display panel and transmitted to the plant process computer for use in the calorimetric measurement. The feedwater mass flow rate is used to determine the reactor thermal output based on an energy balance of the secondary system.

The improved accuracy of measurements of feedwater mass flow and temperature results in a total uncertainty of less than ± 0.37 percent of reactor thermal power. This is substantially more

accurate than the typical ± 2 percent rated thermal power (RTP) assumed in the accident analyses, or that uncertainty currently obtainable with precision, venturi-based flow instrumentation.

The LEFM CheckPlus™ indications of feedwater mass flow will be directly substituted for the venturi-based flow indication and the resistance temperature detector (RTD) temperature indications currently used in the plant calorimetric measurement calculation performed with the plant computer. The plant computer will then calculate enthalpy and thermal power as it does now. The venturi-based feedwater flow measurement will continue to be used for feedwater control and other functions that it currently fulfills.

The DBNPS LEFM CheckPlus™ systems to be installed at the DBNPS will be extensively tested and calibrated at Alden Research Laboratories, in site-specific piping configurations, prior to their installation.

3.3 NUCLEAR STEAM SUPPLY SYSTEM (NSSS) DESIGN PARAMETERS

3.3.1 Introduction

The NSSS design parameters are the fundamental parameters used as input in all the NSSS analyses. They provide the Reactor Coolant System (RCS) and secondary system conditions (temperatures, pressures, flow) that are used as the basis for the design transient, system, component and accident evaluations.

The parameters are established using conservative assumptions in order to provide bounding conditions to be used in the NSSS analyses. For example, the RCS flow assumed is the RCS bounding best estimate flow, which is a conservatively low flow that accounts for flow measurement uncertainty.

3.3.2 Input Parameters and Assumptions

The total thermal power for the uprate analysis was set at 2836 MWt (2819 MWt core). As previously described, the 2819 MWt value is slightly more conservative that the proposed power uprate. The 2836 MWt (2819 MWt core) value is approximately 1.7% higher than the current total thermal power rating of 2789 MWt (2772 MWt core). Feedwater/steam flow, T_{hot}, and T_{cold} were allowed to change as a result of this power uprate. All other input parameters remained the same as those used for the current licensing basis.

3.3.3 Discussion of Parameter Cases

Table 3-1 provides the NSSS parameter cases, which were generated and used as the basis for the uprate project. Parameters were calculated at 0% and 20% OTSG tube plugging to bound the range of RCS temperatures and steam conditions (flow rate and temperature). It is important to note that while conditions were calculated for 20% OTSG tube plugging, this document is not inclusive in terms of justifying 20% tube plugging. This document provides RCS flow and steam temperature values at 20% plugging and demonstrates the insensitivity of OTSG tube flow

induced vibration to tube plugging in the upper span. It does not address the Safety/LOCA analysis and fuel design aspects of tube plugging. The DBNPS tube plugging limit is currently 1300 tubes (8.4%) per steam generator. There are currently 104 tubes plugged and 212 tubes sleeved for a total equivalent plugging of 122 tubes, or 0.8%, in OTSG 1-1, and 436 tubes plugged and 199 tubes sleeved for a total equivalent plugging of 466 tubes, or 3.0%, in OTSG 1-2. The average equivalent plugging is therefore 294 tubes per OTSG, or 1.9%.

A review of Table 3-1 reveals the following changes for the proposed power uprate:

T_{cold} decreased by 0.4 °F for 0% tube plugging

T_{cold} decreased by 1.3 °F for 20% tube plugging

T_{hot} increased by 0.4 °F for 0% tube plugging

T_{hot} increased by 1.3 °F for 20% tube plugging

T_{steam} decreased by 0.1 °F for 0% tube plugging

T_{steam} decreased by 11.9 °F for 20% tube plugging

P_{steam} was held constant at 930 psia

W_{steam} (flow) increased by 1.6% for 0% tube plugging

W_{steam} (flow) increased by 2.9% for 20% tube plugging

T_{fw} was held constant at 455 °F

3.3.4 Conclusions

New plant operating conditions were defined at the analysis power level of 2819 MWt to support the proposed power uprate, Table 3-1. Values were provided for 0% and 20% plugging. The new operating conditions were compared with design conditions for the RCS. The power uprate by itself will not result in operation outside the design conditions. Operation at 8.4% tube plugging will result in operation outside the current design conditions and may require future attention. Specifically, RCS flow is expected to decrease below the current technical specification value at some plugging amount beyond the current 8.4% tube plugging limit.

Table 3-1 NSSS Performance Parameters

	Current Design (1)	Current Operation (2)	1.7% Uprate: No OTSG Tube Plugging	1.7% Uprate: 20% OTSG Tube Plugging
Core Thermal Power (MWt)	2772	2772	2819	2819
Other RCS Power (3) (MWt)	17	17	17	17
Total Thermal Power (MWt)	2789	2789	2836	2836
T_{hot} (°F)	607.5	606.1	606.5	607.4
T_{cold} (°F)	556.5	557.9	557.5	556.6
T_{avg} (°F)	582	582	582	582
RCS Mass Flow Rate (kpph)	137,900	146,020	146,077	140,752
RCS Volumetric Flow Rate (gpm)	369,600	393,060	392,990	378,240
Steam Temperature (°F)	600.0	596.2	596.1	584.3
Terminal Temperature Difference ($T_{hot} - T_{steam}$)	N/A	9.9	10.4	23.1
Feedwater/Steam Flow Rate (4) (kpph)	12,240	11,650	11,840	11,990
Steam Pressure (Input) (psia)	1050	930	930	930
Feedwater Temperature (Input) (°F)	470	455	455	455

Notes:

(1) "Current Design" refers to those values provided in FRA-ANP Document No. 18-1149327, "RCS Functional Specification (DB)," Revision 1, May 27, 1993, and used as input for the Class 1 component fatigue evaluations.

(2) "Current Operation" RCS flow and primary temperatures are based on a core resistance that includes a core debris filter plate for each fuel assembly. Actual, current plant RCS flow should be ≈ 0.3% greater than the values indicated, based on 76 fuel assemblies with debris filter plates.

(3) Other RCS Power corresponds to RCP heat less makeup/letdown heat loss and ambient heat loss.

(4) Slight differences were identified between the feedwater/steam flow rate for the NSSS Performance Parameters in this Table and the BOP Parameters in Table 3-3. These differences are due to the differences in the performance codes. The larger value was used in the evaluation of the effects of the Main Steam and Feedwater flow.

3.4 DESIGN TRANSIENTS

3.4.1 Nuclear Steam Supply System Design Transients

The uprate conditions, as summarized in Table 3-1, were shown to be within the design conditions of FRA-ANP Document No. 18-1149327, "RCS Functional Specification (DB)," Revision 1, May 27, 1993 (hereafter referred to as the "RCS Functional Specification"). These design condition values serve as the final conditions for the power escalation transient, and initial conditions for full power transients such as reactor trip, load rejection, turbine trip, rapid depressurization, loss of flow, power change, and loss of main feedwater transients. Thus, these transients are not changed by the uprate. In addition, the injection transients, such as HPI and AFW, are unchanged since the uprate conditions are bounded by the design transient conditions. Also, since hot standby conditions are unaffected by the power uprate, plant heatups and cooldowns, the most fatigue significant transients, are unchanged. Therefore, it was concluded that the design transients are not adversely affected by the power uprate.

3.5 NUCLEAR STEAM SUPPLY SYSTEM (NSSS) FLUID SYSTEMS

This section presents the results of the evaluations and analyses performed for the NSSS fluid systems and control systems. The results and conclusions of each evaluation and analysis are presented within each subsection.

3.5.1 Reactor Coolant System (RCS)

The RCS consists of two heat transfer loops connected in parallel to the reactor vessel. Each loop contains two reactor coolant pumps, which circulate the water through the loops and reactor vessel, and a once through steam generator (OTSG), where heat is transferred to the main steam system (MSS). In addition, the RCS contains a pressurizer which controls the RCS pressure through electrical heaters, water sprays, a pilot-operated relief valve (PORV) and spring loaded safety/relief valves. The steam discharged from the PORV and safety/relief valves flows through interconnecting piping to the pressurizer quench tank.

Various assessments were performed to help confirm that the RCS design basis functions could still be met at the uprated conditions.

Primary pressure control will not change for the power uprate. The RCP discharge to pressurizer differential pressure does not change appreciably. It was assured that the minimum required pressurizer main spray flow of 190 gpm and the bypass spray flow of 0.8 gpm can be achieved for the uprate conditions defined in Table 3-1. The 20% tube plugging case causes a slight decrease in the pressure differential, resulting in a negligible decrease in main and bypass spray flow. Auxiliary spray flow will be unaffected because the pressurizer pressure does not change with either the power uprate or tube plugging.

The maximum expected T_{hot} at uprated conditions is 607.4°F for the 20% tube plugging case. This temperature is less than the RCS Functional Specification T_{hot} value of 607.5°F and well below the RCS loop design temperature of 650°F.

With respect to the Pressurizer Quench Tank discharge analysis, the nominal full load pressurizer steam volume is unaffected by the uprate since the pressurizer level and pressure (saturation temperature) have not changed. Thus, the existing discharge analysis is unaffected.

3.5.2 Emergency Core Cooling System (ECCS)

The ECCS is used to mitigate the effects of postulated design basis events. The basic functions of this system include providing short and long term core cooling, and maintaining core shutdown reactivity margin. The uprated conditions have no direct effect on the overall performance capability of the ECCS. No changes to the TS limits for the Core Flood Tanks or the Borated Water Storage Tank are required as a result of the power uprate. These systems will continue to deliver flow at the design basis RCS and containment pressures and are therefore unaffected by the proposed power uprate. The ECCS consists of three subsystems.

3.5.2.1 Core Flood System (CF)

The passive portion of the system is the two Core Flood Tanks (CFT) which are connected to each of the Low Pressure Injection (LPI) lines entering the Reactor Vessel. Each CFT contains borated water under pressure (nitrogen cover gas). The borated water automatically injects into the RCS when the pressure within the RCS drops below the operating pressure of each of the accumulators.

3.5.2.2 High Pressure Injection (HPI)

The active part of the ECCS injects borated water into the reactor following a break in either the reactor or steam systems in order to cool the core and prevent an uncontrolled return to criticality. Two High Pressure Injection (HPI) pumps take suction from the Borated Water Storage Tank (BWST) and deliver borated water to the reactor vessel via four cold leg connections.

3.5.2.3 Decay Heat Removal/Low Pressure Injection (DHR/LPI)

The primary ECCS function of the DHR/LPI system is to provide pumped injection of low pressure water directly into the reactor vessel for long term cooling of the reactor core during a loss-of-coolant accident. The primary normal operation function of the DHR/LPI system is to remove sensible and decay heat from the core and reduce the temperature of the RCS during the second phase of plant cooldown (see Section 3.5.4).

3.5.3 Makeup & Purification System (MU&P)

The MU&P system provides for boric acid addition, chemical additions for corrosion control, reactor coolant cleanup and degasification, reactor coolant makeup, reprocessing of letdown

water from the RCS, and RCP seal water injection. During plant operation, reactor coolant flows through the tube side of the Letdown Cooler and then through a letdown orifice. The Letdown Cooler reduces the temperature of the reactor coolant and the letdown orifice reduces the pressure. The cooled water leaves the reactor containment and enters the auxiliary building. After passing through one of the mixed bed purification demineralizers, where ionic impurities are removed, coolant flows through the Makeup filter and enters the Makeup tank (MUT).

In the assessment of MU&P system operation at revised RCS operating temperatures, the maximum expected RCS T_{cold} must be less than or equal to the applicable Makeup system design temperature and less than or equal to the letdown cooler design inlet operating temperature. The former criterion supports the functional operability of the system and its components. The latter criterion confirms that the letdown cooler design operating conditions remain bounding.

With regards to the MU&P system thermal performance, the current T_{cold} of 557.9°F is greater than the 0% tube plugging T_{cold} of 557.5°F or the 20% tube plugging T_{cold} of 556.6°F. Also, it is much less than the tube side design temperature of 600°F for the Letdown Cooler. The letdown path is used to process effluents associated with fluid expansion during plant heatup and thus, is unaffected by the revised T_{cold} at full power conditions. Therefore, operation of the MU&P System is unaffected by the revised RCS temperatures.

3.5.4 Decay Heat Removal/Low Pressure Injection System (DHR/LPI)

The DHR/LPI system is designed to remove sensible and decay heat from the core and reduces the temperature of the RCS during the second phase of plant cooldown. As a secondary function, the DHR/LPI system is used to transfer refueling water between the BWST and the refueling canal at the beginning and end of refueling operations.

The DHR/LPI system consists of two decay heat coolers, two DHR/LPI system pumps and associated piping, valves and instrumentation. During system operation, reactor coolant flows from one hot leg of the RCS to the DHR/LPI system pumps, through the tube side of the decay heat coolers and back to the Reactor Vessel downcomer region via the Core Flood nozzles. Component cooling water circulates through the shell. The decay heat coolers are of the shell and U-tube type.

A single train cooldown analysis and a normal cooldown analysis were performed to address the uprated reactor power. A single train cooldown is defined as cooling the RCS from 280°F at six hours after plant shutdown to 140°F by employing one DHR Pump, one DHR cooler and one train of component cooling. The overall single train cooldown should be achieved within 175 hours after plant shutdown based on system design requirements. An analysis determined that the cooldown would be extended by about 7 hours from 168 hours at 2772 MWt to 175 hours at 2819 MWt as a result of the power uprate.

A normal cooldown is defined as cooldown assuming all equipment available. The system design basis is that the normal cooldown can be achieved within approximately 24 hours following plant shutdown. Normal cooldown is defined as cooling the RCS from 280°F at six hours after plant shutdown to 140°F using two trains of cooling equipment (2 trains of DHR and

component cooling water). At the power uprate conditions, the cooldown can be achieved within 26 hours, a two hour increase.

The decay heat rate used in the cooldown analyses was based on the guidance of American Nuclear Society (ANS) Standard 5.1, "Decay Energy Release Rates Following Shutdown of Uranium-Fuel Thermal Reactors (DRAFT)," October 1971. A reactor operating time of 16,000 hours was assumed, consistent with Standard Review Plan 9.2.5 Branch Technical Position (BTP) ASB 9-2, "Residual Decay Energy Release Rate for Light-Water Reactors for Long-Term Cooling." As recommended in the BTP, an uncertainty factor of 10% was applied to the fission product decay for cooling times between 10^6 and 10^7 seconds.

3.5.5 Spent Fuel Pool Cooling System

The expected decay heat load increase will be proportional to the power increase. Spent fuel pool cooling calculations, including those for the proposed spent fuel pool re-racking project (license amendment application submitted via DBNPS letter to the NRC dated December 2, 2000, DBNPS Serial Number 2726) were reviewed and found to have adequate margin.

3.6 NUCLEAR STEAM SUPPLY SYSTEM COMPONENTS

The reactor vessel (RV) was evaluated at the uprated conditions for the structural acceptability of the vessel, and for the reactor vessel integrity in terms of the impact due to neutron fluence.

3.6.1 Reactor Vessel Structural Evaluation

No changes in RCS design or operating pressure were made as part of the power uprate. The conditions analyzed in the existing design basis analyses for the reactor vessel are based on the RCS Functional Specification. As noted in Section 3.3.4, the uprated conditions (RCS temperatures) are bounded by the design conditions in the RCS Functional Specification. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. Thus, the existing stress reports for the reactor vessel remain applicable for the uprated power conditions.

3.6.2 Reactor Vessel Integrity – Neutron Irradiation

The uprated conditions in Table 3-1 can affect the analyses generally in two ways. One way is that changes in T_{cold} may affect the value used in the various analysis methods. The second way is that the increase in core power can increase the neutron fluences experienced by the vessel.

The current analyses assume that the T_{cold} is maintained at 557.9°F. The T_{cold} of 556.6°F for the power uprate (see Table 3-1) with 20% tube plugging, is bounded by the assumed 5% increase in fluence (see the following discussion).

3.6.2.1 Neutron Fluence

The current 32 EFPY fluences for the DBNPS reactor vessel beltline materials are reported in Topical Report BAW-2108, "Fluence Tracking System," Revision 1. The reactor vessel fluence will increase with the power uprate. To bound the increase in fluence, the current 32 EFPY inside surface fluence values for the DBNPS reactor vessel beltline materials were increased by 5 percent (i.e., fluence = 1.05 x current fluence).

3.6.2.2 Surveillance Capsule Withdrawal Schedule

A withdrawal schedule is developed to periodically remove surveillance capsules from the reactor vessel to effectively monitor the condition of the reactor vessel materials under actual operating conditions. Since the revised fluence projections do not appreciably exceed the fluence projections used in development of the current withdrawal schedules, then the current withdrawal schedules remain valid.

3.6.2.3 Heatup and Cooldown Pressure / Temperature Limit Curves

The current P-T limit curves are valid through 21 effective full power years (EFPY) and are based on adjusted reference temperatures at the ¼-thickness (¼T) and ¾-thickness (¾T) wall locations for the limiting reactor vessel beltline material, the upper to lower shell circumferential weld WF-182-1. The adjusted reference temperature values of 155°F and 114°F for the ¼T and ¾T wall locations respectively were calculated in accordance with Regulatory Guide 1.99, Revision 2. With the implementation of the power uprate, these calculations were re-evaluated based on a 5% fluence increase added to the 21 EFPY peak inside surface fluence (E > 1.0 MeV), also using the guidelines of Regulatory Guide 1.99, Revision 2. With all inputs used in the re-evaluation unchanged (except fluence), the limiting reactor vessel beltline material adjusted reference temperatures at the ¼T and ¾T wall locations increased to 157°F and 115°F respectively. The increase in the ¼T and ¾T adjusted reference temperature values results in a decrease in the P-T limit curve validity time (i.e., 21 EFPY). To determine the reduction in the validity of the existing P-T limit curves, a validity time period adjustment ratio is performed by multiplying the current validity time (i.e., 21 EFPY) by the ratio of the 5% increased fluence to the fluence used in the existing DBNPS adjusted reference temperature calculation. Based on this ratio, the validity time of the current DBNPS P-T limit curves is reduced from 21 EFPY to 20 EFPY (i.e., the existing P-T limit curves are valid through 20 EFPY). As described in the license amendment application, TS Figures 3.4-2, 3.4-3, and 3.4-4 will require revision accordingly.

3.6.2.4 Low Temperature Overpressure Protection System (LTOP)

LTOP is designed to protect the RCS from overpressure events when the RCS temperature is below 280°F. Changes to full power operating parameters, such as NSSS power, do not impact LTOP. Thus, the existing LTOP analysis is unaffected.

3.6.2.5 Pressurized Thermal Shock (PTS)

The RT_{PTS} values in support of a power uprate applicable to the projected end-of-life period (32 EFPY) for the reactor vessel beltline materials were re-evaluated. These values were calculated in accordance with the requirements in the Code of Federal Regulations, Title 10, Part 50.61 (10 CFR 50.61). The controlling beltline material for the reactor vessel is the upper shell to lower shell circumferential weld, WF-182-1, with a RT_{PTS} value of 193.5°F. The screening criterion for this weld metal is 300°F. Therefore, the reactor vessel remains within its limits for PTS at the uprated condition.

3.6.2.6 Alloy 600 Primary Water Stress Corrosion Cracking (PWSCC)

The effects of a temperature increase resulting from the power uprate on Alloy 600 PWSCC has been evaluated. For the limiting case of 20% OTSG tube plugging, it is estimated that the increase of T_{hot} from 606.1°F to 607.4°F decreases the time to PWSCC initiation by 5% and increases the crack growth rate by 4%. Because the power uprate does not increase the T_{cold} and T_{avg}, or the pressure and T_{sat}, the impact is limited to Alloy 600 components and welds operating near T_{hot}. Examination of the FRA-ANP Alloy 600 ranking model shows that the current relative PWSCC ranking of Alloy 600 components will not change after the power uprate. The current top three most PWSCC susceptible components are all in the pressurizer, and therefore not affected by the power uprate. These components in the pressurizer continue to be the most susceptible after the power uprate. Hence, the impact of the power uprate on Alloy 600 PWSCC is considered very limited and bounded by current B&WOG aging management programs for Alloy 600.

On August 3, 2001, the NRC issued Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." The issues identified in this bulletin are being addressed separately.

3.6.3 Reactor Internals

The reactor internals support and orient 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, reported in Table 3-1, 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 former plate region. Several analyses have been performed to demonstrate that the reactor internals can perform their intended design functions at the uprated conditions.

3.6.3.1 Thermal-Hydraulic Systems Evaluations

Core Support Structures and Vessel Internals

The uprated conditions were reviewed for impact on the existing design basis analyses for the core support structure and reactor vessel internals. No change in RCS design or operating pressure was made as a part of the power uprate. The conditions analyzed in the existing analyses are based on the RCS Functional Specification. As noted in Section 3.3.4, the uprated conditions are bounded by the conditions in the RCS Functional Specification. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses, and fatigue values remain valid.

Reactor Internals Flow Induced Vibration (FIV)

The uprate by itself does not result in an appreciable change in RCS flow compared to the current operation value ($<< 0.1\%$ -- see Table 3-1). Thus, core DNBR, core lift, and flow induced vibration on reactor vessel internals are not affected by the power uprate.

OTSG tube plugging does reduce the RCS flow. Thus, as tube plugging increases, future core designs will need to accommodate the reduction in RCS flow.

Steam generator tube plugging does not adversely affect primary component flow induced vibration. Flow induced vibration is a result of the dynamic pressure, or the density-velocity-squared product of the flow. In terms of mass flow rate, the dynamic pressure is proportional to mass flow rate squared divided by the density. Since the uprated power RCS volumetric flow decreases with additional steam generator tube plugging (Table 3-1) and the uprated RCS volumetric flow rate is less than the current operation value, the existing reactor internals flow induced vibration analyses remain bounding.

Control Rod Assembly (CRA) Drop Time Analyses

The DBNPS Technical Specifications require that the CRA drop time be less than or equal to 1.58 seconds with all four reactor coolant pumps operating and the RCS temperature ≥ 525 °F. The actual test is performed at or near the hot zero power temperature of 532 °F. There is typically a large margin between measured CRA drop time and the Technical Specification requirement.

For the power uprate, the hot zero power temperature and the average temperature during heatup to the hot full power conditions (582 °F) will not be changed. The full power hot leg temperature will increase slightly by approximately 0.5 °F, and the cold leg temperature will be slightly cooler by approximately 0.5 °F, but this will have no significant effect on the actual CRA insertion time. Therefore, there should be no noticeable increase in the measured CRA drop times due to the RCS temperature change

associated with the power uprate, and there will be no change to the Technical Specification maximum allowable CRA drop time.

3.6.3.2 Mechanical Evaluations

The uprated conditions do not affect the current design bases for seismic and LOCA loads. Thus, it was not necessary to re-evaluate the structural affects from seismic OBE and SSE loads, and the LOCA hydraulic and dynamic loads. With regards to flow and pump induced vibration, the current analysis uses a mass design flow rate, which did not change for the uprated conditions. The uprated conditions will slightly alter the T_{cold} and T_{hot} fluid densities, which will slightly change the forces induced by flow. However, these changes are insignificant when compared to the current design temperature ranges. Thus, the uprated conditions do not affect the mechanical loads.

3.6.3.3 Structural Evaluations

The uprated conditions were reviewed for impact on the existing design basis analyses for the reactor vessel internals. No changes in RCS design or operating pressure were made as part of the power uprate. The design conditions in the existing analyses are based on the RCS Functional Specification. As noted in Section 3.3.4, the uprated conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. Thus, the existing stress reports for the reactor vessel remain applicable for the uprated power conditions.

3.6.4 Control Rod Drive Mechanisms (CRDMs) Structural Evaluation

The power uprate, even with 20% tube plugging, will result in core outlet temperatures and, thus, reactor vessel head temperatures within the design temperature (see Table 3-1). Also, primary pressure control will not be affected by the power uprate. Therefore, since the control rod drive mechanisms will operate within their design considerations, no change to the design is required.

3.6.5 Reactor Coolant Loop Piping and Supports Structural Evaluation

The uprated conditions were reviewed for impact on the existing design basis analyses for the reactor coolant piping and supports. No changes in RCS design or operating pressure were made as part of the power uprate. The design conditions in the existing analyses are based on the RCS Functional Specification. As noted in Section 3.3.4, the uprated conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. Thus, the existing stress reports for the reactor loop piping and supports remain applicable for the uprated power conditions.

3.6.6 Reactor Coolant Pumps (RCPs) and Motors

3.6.6.1 Reactor Coolant Pump Structural Evaluation

The uprated conditions were reviewed for impact on the existing design basis analyses for the reactor coolant pump. No changes in RCS design or operating pressure were made as part of the power uprate. The design conditions in the existing analyses are based on the RCS Functional Specification. As noted in Section 3.3.4, the uprated conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. Thus, the existing stress reports for the reactor coolant pumps remain applicable for the uprated power conditions.

3.6.6.2 Reactor Coolant Pump and Motor Evaluation

The power uprate changes the RCS flow only slightly. Thus, the pump head capacity performance and NPSH requirements are unchanged. With tube plugging, RCS flow decreases and the developed head increases. NPSH requirements will decrease slightly with the decreased flow.

Since the uprate will not cause a significant RCS flow change, the pump power requirements will not change. With 20% tube plugging, the pump flow will change from approximately 98,250 gpm per pump to 94,560 gpm per pump. The brake horsepower requirements do not perceptibly change over this flow change.

Pressure-temperature related limits (RCP NPSH, RCP Seal Staging) use a location adjustment between the pump suction and the hot leg to ensure compliance with the limits when sensing hot leg pressure. These limits will not be affected by the power uprate but the location adjustments will need to be recalculated with additional tube plugging.

The RCP motors were evaluated based on the revised design conditions for continuous operation at the revised rated conditions. The revised rated conditions will have no effect upon motor operation during pump start and loop operation. Since the power uprate changes the RCS flow very little, the pump head capacity performance and NPSH requirements are unchanged. Therefore, this evaluation concludes that the uprated conditions will have a negligible impact on the pump for all conditions of operation. The RCP motors will be able to continue operation at their revised operating ratings, and accelerate at their design basis starting conditions. The RCP's thrust bearings will not exceed their load ratings.

3.6.7 Once Through Steam Generators (OTSG)

During normal operation, the Steam Generators provide a means of transferring heat from the RCS to the secondary side systems. The Steam Generators are also required to be capable of the removal of decay heat, Reactor Coolant pump heat, and sensible heat at a rate sufficient to cool the RCS from hot shutdown to 280°F in 6 hours.

3.6.7.1 OTSG Thermal-Hydraulic Performance

The following evaluations and analyses were performed to assess the impact of the uprated conditions on the thermal-hydraulic performance of the steam generators.

Steam Generator Inventory

Within the tube bundle, the water level will not appreciably change due to the power uprate as evidenced by the similar steam temperature calculations. The downcomer inventory, however, will increase due to the increased feedwater flow. This is caused by the increase in tube region unrecoverable pressure drop, which must be offset by an increased downcomer water column (inventory). Additional tube plugging will cause further increases in water level due to the increased boiling lengths required due to the reduction in heat transfer area. Inventory limits on the steam generator are based on safety analysis. The safety analyses were performed using the maximum inventory possible without flooding the aspirator ports. Thus, plant operation will continue to be limited to the current 96% operate level value to comply with these inventory limits.

Due to the power uprate itself, the startup level is expected to increase about 8 inches. The combined effect of the power uprate and 20% tube plugging would result in an expected increase of about 15 inches. These estimates are based on the change in startup level versus feedwater flow data during power escalation for another B&W plant. Based on nearly identical steam generator designs, these estimates are valid for the DBNPS.

The trending of steam generator water levels has shown that tube support plate fouling has a dominant effect on measured water levels. So while some increase in level will occur due to the power uprate and additional tube plugging, the effects of fouling will dominate such that the power uprate and tube plugging effects (assuming a gradual progression in plugging) will not likely be observed.

Steam Generator Temperature

For the uprate alone (i.e., without tube plugging considerations), the change in steam temperature does not cause a change in the steady-state tube-to-shell ΔT. Additional tube plugging, however, can cause a significant decrease in steam temperature. As a result, the steam annulus and adjacent shell temperature will decrease as well. The lower shell temperature will remain constant as long as steam pressure remains unchanged. Relative to tube-to-shell ΔT limits, tube plugging improves the tensile (shell hotter than tubes) conditions and exacerbates the compressive (tubes hotter than shell) conditions.

During zero-power situations, the steam temperature will be the saturation temperature corresponding to the steam pressure, which is governed by operator actions (in the cases of heatups and cooldowns, for example) or by secondary relief valve setpoints (in the case of plant transients). Thus, tube plugging should not affect the shell temperature for zero-power conditions.

For power conditions, superheated steam is produced when the steam generator pressure reaches its normal power operating value. The full power (no plugging) tube-to-shell temperature difference can be approximated as:

Tube-to-shell ΔT = Average Tube Temperature – Average Shell Temperature

$$\text{Average Tube Temperature} = (T_{hot} + T_{cold})/2$$

$$\text{Average Shell Temperature} = 3/5 * T_{sat} + 2/5 * T_{steam}$$

The 3/5, 2/5 ratios are based on the relative lengths of the steam and feedwater downcomers. For the uprated power condition:

At $P_{OTSG} = 930$ psia, $T_{sat} = 536.1°F$

At no plugging, $T_{steam} \approx 596°F$

$$\text{Average Shell Temperature} = 3/5*536.1 + 2/5*596 = 560.1°F$$

$$\text{Tube-to-shell } \Delta T = 582 - 560.1 \approx 22°F$$

At large tube plugging (i.e., >> 20%) $T_{steam} = 570°F$

$$\text{Average Shell Temperature} = 3/5*536.1 + 2/5*570 = 549.7°F$$

$$\text{Tube-to-shell } \Delta T = 582 - 549.7 \approx 32°F$$

While the tube-to-shell ΔT will increase with plugging, it is still within the 60°F compressive limit. In summary, the power uprate and 20% tube plugging will not cause the design specification values to be exceeded.

3.6.7.2 OTSG Structural Integrity Evaluation

The uprated conditions were reviewed for impact on the existing design basis analyses for the steam generator. No changes in RCS design or operating pressure were made as part of the power uprate. The design conditions in the existing analyses are based on the RCS Functional Specification. As noted in Section 3.3.4, the uprated conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing

loads, stresses and fatigue values remain valid. Thus, the existing stress reports for the OTSG remain applicable for the uprated power conditions.

OTSG Tube Integrity

The uprated conditions were reviewed for impact on the existing design basis analysis for the steam generator tubes. An evaluation was performed to demonstrate that the existing structural and fatigue analyses of the steam generator tubes continue to comply with the ASME Code limits for the uprated conditions. This evaluation considered the steam generator tubes with regard to stress and fatigue usage. The evaluation demonstrated that the steam generator tubes continue to comply with the requirements of the ASME Code for the uprated conditions.

OTSG Flow Induced Vibration

Note: The following discussion is specific to hardware supplied by B&W/FTI. A review of FIV analysis for plugs and stabilizers supplied by ABB/CE is ongoing and will be completed prior to implementation of the proposed power uprate.

The best estimate feedwater flow rates, as a function of steam temperature, were determined. These values are based on the uprated thermal power, a feedwater temperature of 455°F, and a steam pressure of 930 psia. The flow rates were determined at the expected steam temperature of 596°F, the current steam temperature of 590°F, and the 20% tube plugging temperature of 585°F. Because the steam density increases with decreasing temperature, the dynamic pressure used for flow-induced vibration (FIV) analyses is effectively constant for all the points along the feedwater flow versus steam temperature line.

Rather than perform the subsequent uprate FIV analyses at the best estimate conditions, the analyses were performed at a greater flow rate. This provides margin for instrument uncertainty, asymmetric OTSG operating levels, changes in steam pressure, and three RCP power operations. The uprated "design" flow rate was 11,883,000 lbm/hr, 2% greater than the previous FIV analyzed condition.

The various degradation indications that were detected in the DBNPS OTSG tubes have been repaired with mechanical sleeves and continue to operate, while others have been taken out of service by plugging. Some of the plugged tubes were stabilized with various stabilizer designs. Both the virgin tubes, sleeved tubes, and plugged and stabilized tubes have been certified, by prior analyses, to be free of flow-induced vibration problems. At a power uprate of 2%, the flow velocities in the OTSG will increase correspondingly by approximately 2%. The forcing function on the tubes due to fluid flow increases approximately 4% during full power operation. The integrity of the tubes, virgin, sleeved or stabilized, were re-assessed with the latest techniques and input parameters in a flow induced vibration analysis.

To assess the margins the tubes in the OTSGs have against detrimental flow-induced vibration effects, the tubes with the smallest margins were identified and their present margin of safety was reassessed using the results of the new FIV analyses for the increased flow rate. Because the various hardware used to repair these tubes were developed over a period of 19 years, a thorough review of the past flow induced vibration qualification of both the virgin tube, sleeve, and the different stabilizers was performed.

Because the qualification analyses were performed through a span of over 19 years, there were differences in the methodology, the computer codes and the input parameters, used that result in slightly different results even for the identical hardware. The fluid-elastic stability margins for repaired tubes were back adjusted to the current industry standard for stability constant. Additionally, the stability margins were back adjusted to the current industry standard for viscous damping values for the various stabilizer designs.

The reassessment shows that the original functional integrity of the installed hardware is maintained for the increased flow rate. The tube bundle in the OTSGs will have a minimum fluid-elastic stability margin of about 27%. The minimum margin against excessive turbulence-induced stress is in the sleeve at 33.5%. The latter translates into a 15% margin in the mass flow rate. Therefore, the frequency of tube-tube impacting is determined to be insignificant with a 2% increase in the cross flow velocity in all tubes.

3.6.7.3 OTSG Hardware Changes and Addition Evaluation

OTSG Tube Repair Hardware

The OTSG tubes have been repaired using welded tube plugs, mechanical tube plugs, mechanical sleeve plugs, mechanical sleeves, and tube stabilizers.

The revised operating conditions were reviewed for impact on the existing qualification reports and design calculations for the repair hardware. Section 3.6.7.2 states that the existing steam generator loads remain valid. The evaluations showed that the temperature changes, due to the power uprate, are bounded by those used in the sleeve and plug qualifications and analyses. The effect of the flow increase was also evaluated and showed that all installed tube repair hardware maintained their functional integrity with the increased secondary side flow rates. Therefore, the existing structural and fatigue analyses remain valid for the installed tube repair hardware. Thus, the existing stress reports for the mechanical and welded plugs, mechanical sleeve plugs, mechanical sleeves, and tube stabilizers remain applicable for the uprated power conditions. Note: The above discussion is specific to hardware supplied by B&W/FTI. A review of qualification reports and design calculations for repair hardware supplied by ABB/CE is ongoing and will be completed prior to implementation of the proposed power uprate.

Tube Plugging and Repair Criteria

The DBNPS' current Steam Generator program follows the inspection guidelines contained in the latest revision of the EPRI PWR Steam Generator Examination

Guidelines. The modest power uprate will not require a change to the program. The DBNPS currently inspects for all active and potential degradation. The pre-outage degradation assessment includes DBNPS-specific degradation as well as industry degradation. The ongoing forms of degradation in the DBNPS Steam Generators are:

- Upper Tube End PWSCC
- Upper Tubesheet Roll Transition PWSCC
- Volumetric Degradation in the Tubesheet and Tube Support Crevices
- Freespan Volumetric Degradation
- Wear at Tube Supports
- Freespan Axial Outside Diameter Stress Corrosion Cracking/Intergranular Attack at Upper Bundle Denting from Auxiliary Feedwater Stabilization

Based on condition monitoring and operational assessments of inspection results, expansion of inspection plans and repairs are made. Potential degradation growth rate changes will be incorporated into the operational assessment associated with potential effects of the uprate.

The revised operating conditions were reviewed for impact on the existing analyses that support a 40% through-wall plugging criteria for a tube or sleeve. The power uprate does not change the existing operating pressure differential across the tube wall for either the 0% plugging or the 20% tube plugging conditions. The change in tube temperature has an insignificant effect on the tube strength properties. The presently predicted OTSG tube pressure differentials and tube loads during Faulted Conditions remain valid. The decrease in SCC crack initiation time of approximately 1.5% and 5%, respectively, for 0% and 20% tube plugging due to the temperature increase is not considered significant. Thus, the crack growth estimates remain valid and the current 40% tube/sleeve plugging criteria remains applicable for the uprated power conditions. However, the higher temperature will be considered in future growth rate analyses.

Tube-to-tube support plate wear calculations were not performed for the power uprate condition. However, the tube-to-tube support plate wear growth rate is estimated to increase by approximately 8%. This estimate is based on the cross flow velocity having a fourth order effect on the work rates, $(1.02)^4 = 1.08$. This estimated increase in the wear rates applies to the virgin tubes, sleeved tubes and tubes with various stabilizers. This is a qualitative estimate and should only be used as a general guideline.

The current Reg. Guide 1.121 analyses remain valid under the power uprate condition. The uprated power conditions will not impact tube inspection during future outages nor will the methodology (assumptions and parameters used) for condition monitoring and operational assessments be impacted.

3.6.8 Pressurizer Structural Evaluation

The uprated conditions were reviewed for impact on the existing design basis analyses for the pressurizer. No changes in RCS design or operating pressure were made as part of the power uprate. The design conditions in the existing analyses are based on the RCS Functional Specification. As noted in Section 3.3.4, the uprated conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. Thus, the existing stress reports for the pressurizer remain applicable for the uprated power conditions.

3.6.9 Reactor Coolant System Attached Piping and Supports (Decay Heat, Makeup and Purification, High Pressure Injection, Low Pressure Injection) Structural Evaluation

The uprated conditions were reviewed for impact on the existing design basis analyses for the reactor coolant system attached piping and supports. No changes in RCS design or operating pressure were made as part of the power uprate. The design conditions in the existing analyses are based on the RCS Functional Specification. As noted in Section 3.3.4, the uprated conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the existing loads, stresses and fatigue values remain valid. Thus, the existing stress reports for the reactor coolant system attached piping and supports remain applicable for the uprated power conditions.

3.6.10 Fuel Assembly

The DBNPS 15x15 Mark-B fuel design was evaluated to determine the impact of the power uprate on the fuel assembly structural integrity. Since the core plate motions for the seismic and LOCA evaluations are not affected by the uprated conditions, there is no impact on the fuel assembly seismic/LOCA structural evaluation. The power uprate does not increase operating and transient loads such that they will adversely affect the fuel assembly functional requirements. Therefore, the fuel assembly structural integrity is not affected and the seismic and LOCA evaluations of the 15x15 Mark-B fuel design are still applicable for the power uprate.

3.6.11 Leak Before Break (LBB)

The Leak-Before-Break (LBB) concept applies known mechanisms for flaw growth to piping designs with assumed through-wall flaws and is based on the plant's ability to detect an RCS leak. The LBB evaluation of the RCS primary piping showed that a double-ended guillotine break will not occur and that postulated flaws producing detectable leakage exhibit stable growth, and thus, allow a controlled plant shutdown before any potential exists for catastrophic piping failure (BAW-1847, Rev. 1). The major areas that contributed to this evaluation were RCS piping structural loads, leakage flaw size determination, flaw stability analysis, and RCS piping material properties. The RCS piping fracture mechanics and limit load analyses techniques are not changed by the power uprate.

The RCS piping loads used in the LBB analysis are various combinations of deadweight, thermal expansion and seismic load cases. As discussed in the evaluation of structural analysis parameters, these loads are not affected by power uprate.

The leak rate used to determine the leakage flaw size was 10 gpm to provide a margin against the Regulatory Guide 1.45 requirement of 1 gpm. Flaw length and leakage calculations are based on the pipe dimensions, thermodynamic properties and material properties. The system parameters used in the analysis represented realistic conditions existing in the plant during normal full-power operation and compare closely (1-2%) to the worst plant pressure and temperature conditions for the uprated normal operations. The calculations will not be affected by power uprate.

3.7 NSSS/BOP FLUID SYSTEMS INTERFACE

The following BOP fluid systems were reviewed to assess compliance with the NSSS/BOP uprated conditions shown in Sections 3.3 and 3.8.

3.7.1 Main Steam System (MSS)

The following summarizes the evaluation of the major components of the Main Steam System (MSS) relative to the uprated conditions.

3.7.1.1 Main Steam Safety Valves (MSSV)

The MSSVs must have sufficient capacity so that the main steam pressure does not exceed 110 percent of the MSS design pressure (the maximum pressure allowed by ASME B&PV Code). Each Main Steam line contains nine Main Steam Safety Valves, which provide overpressure protection for the OTSGs and the Main Steam Lines. The DBNPS eighteen MSSVs have a combined total relieving capacity of 14.175×10^6 lbm/hr. The limiting transient for overpressure of the OTSGs is the turbine trip event. A turbine trip analysis was recently performed at an analyzed power level of 3025 MWt and used the current installed valve capacity. This analysis confirmed that the peak pressure was less that the ASME code allowable. This also sufficiently bounds the steam flow at maximum normal power uprate conditions (approximately 12,000,000 lb/hr). Therefore, based on the uprated conditions, the capacity of the installed MSSVs meets the sizing criteria.

3.7.1.2 Main Steam Atmospheric Vent Valves (MSAVV)

The primary function of the MSAVVs 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 MSAVVs in conjunction with the Auxiliary Feedwater System (AFW) permit the plant to be cooled down from the pressure setpoint of the lowest-set MSSVs to the point where the Decay Heat System (DH) can be placed in service. During cooldown, the MSAVVs are either automatically or manually controlled.

Each MSAVV controller automatically compares steam line pressure to the pressure setpoint, which is manually set by the plant operator.

The MSAVVs function in conjunction with the Main Steam Turbine Bypass Valves (MSTBVs) to remove decay heat and stored heat from the NSSS during normal cooldown. At the uprated conditions, the percent steam flow capacity of the MSAVVs is reduced from an original design of 5% of full steam flow to 4.86% of full steam flow. The revised design conditions do not affect the safe operation of the plant. The USAR accident analyses credit the main steam safety valves for SG overpressure protection.

3.7.1.3 Main Steam Turbine Bypass Valves (MSTBV)

The MSTBVs are addressed in Section 3.7.2.

3.7.1.4 Main Steam Non-Return Valves (MSNRV)

The main steam line non-return valves are located in the Turbine Building to reduce the amount of energy that would flow out of a double-ended steam line rupture if it occurs in the Auxiliary Building or the containment structure. The proposed power uprate does not affect the safety function of the MSNRVs. The MSNRVs still function to automatically close on reverse flow. USAR Section 10.3.3, "Main Steam Supply System – Accident Analysis," evaluates the hydrodynamic effects on the valve disc due to an instantaneous break in the Auxiliary Building wall nearest the check valve. The valve disc centerline velocity at impact is 117 ft/sec at current full power conditions with a steam pressure of 925 psia. A sensitivity analysis indicates that the impact velocity increases linearly with system pressure. As a result of the proposed power uprate the main steam system pressure is expected to increase to approximately 932 psia, which results in a disc centerline velocity increase to approximately 118 ft/sec. This is still well below 129.5 ft/sec which is documented as acceptable in USAR Section 10.3.3.

3.7.1.5 Main Steam Isolation Valves (MSIV) and MSIV Bypass Valves

The MSIVs are located outside the containment and downstream of the MSSVs. The valves function in conjunction with the MSNRVs to prevent the uncontrolled blowdown of more than one steam generator and to minimize the RCS cooldown and containment pressure within acceptable limits following a MSLB. To accomplish this function, the MSIVs are required to close in 6 seconds in the event of a MSLB, 6.5 seconds in the event of a feedwater line break, and 34 minutes in the event of a steam generator tube rupture. The ability of the MSIVs to close within the required times is not affected by power uprate. Based on the valve construction, increased steam flow acts to assist in closure of the valve. The MSIVs are Flite-Flow Balanced Stop Valves. Based on the orientation of the valve seat and stem with respect to the flow direction, the forward flow across the valve assists the valve closure. As a result, the increased steam flow and pressure drop across the valve due to power uprate does not affect the ability of the MSIVs to fully close.

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

3.7.2 Main Steam Turbine Bypass Valves (MSTBV)

The MSTBVs function in conjunction with the MSAVVs to remove decay heat and stored heat from the NSSS during normal cooldown. The MSTBVs are not credited in accident mitigation as identified in the USAR. At the revised design conditions, the percent steam flow capacity of the MSTBVs is reduced from an original design of 25% of full steam flow to 24.5% of full steam flow. The revised design conditions do not affect the safe operation of the plant. The main steam turbine bypass valves (MSTBVs) create an artificial steam load by dumping steam from ahead of the turbine throttle valves to the main condenser. The sizing criterion recommends that the MSTBVs be capable of discharging 15 percent of the rated steam flow at full-load steam pressure to permit the NSSS to withstand an external load rejection or turbine trip without lifting the MSSVs or tripping the reactor. Post-TMI plant modifications removed this ability. As a result, the sizing requirement is no longer applicable. Therefore, the small power uprate will have no effect on the required MSTBV relief capability.

3.7.3 Condensate and Feedwater System

The condensate and feedwater system (C&FS) must automatically maintain steam generator water levels during steady-state and transient operations. The range of revised NSSS performance parameters results in a nominal feedwater volumetric flow increase of up to 2.0 percent during full-power operation. The higher feedwater flow has an impact on system pressure drop, which may increase by as much as 5.0 percent. The system has been evaluated to accommodate the system pressure drop for uprate.

The major components of the C&FS that interface with the NSSS are addressed below. Other C&FS components, such as feedwater heaters and piping, are evaluated in Section 3.8.

3.7.3.1 Main Feedwater Stop Valves/Main Feedwater Control Valves

The main feedwater stop valves (MFSVs) are located outside containment and downstream of the main feedwater control valves (MFCVs). The valves function in conjunction with the primary isolation signals to the MFCVs 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 cooldown. These requirements are not impacted by power uprate.

The MFSVs are required to close within 17 seconds after an SFRCS actuation from a MSLB, Main Feedwater Line Break (MFLB), or a loss of both MFPs to limit flooding and pressure/temperature transients, and to limit excessive RCS cooling and subsequent

reactivity insertion. The maximum differential pressure that the valve is to close against is calculated based on the shut-off head of the Main Feedwater Pump at the maximum (rated) turbine speed of 5150 rpm. This conservatively bounds conditions at the valve after power uprate (5025 rpm, as discussed in Section 3.7.3.2).

The MFCVs are also credited as a back-up source of isolation in the event of a failure of the MFSVs. The MFCVs are designed to fully close within 7 seconds at a normal operating pressure of 1050 psig and a maximum shut-off pressure of 1450 psig. This bounds conditions in the line after power uprate.

3.7.3.2 Condensate and Feedwater System Pumps

The C&FS available head, in conjunction with the MFCV characteristics, must provide sufficient margin for feed control to provide adequate flow to the steam generators during steady-state and transient operation. A continuous steady feed flow should be maintained at all loads. Two Main Turbine Driven Feedwater pumps are available to supply feedwater to the steam generators. The Main Feed Pump Turbines exhaust to the Main Condenser. The main feedwater pumps are currently designed to pass a combined total of 11,760,260 lb/hr. This results in a turbine speed to the Main Feedwater pump of 4927 rpm. As presented in Table 3-3, after power uprate, the bounding feedwater flow is increased to 12,044,000 lb/hr and the maximum flow through a single Main Feedwater pump is increased to 6,248,297 lb/hr. This results in an increase in turbine speed to 5025 rpm. The turbine is rated at 5150 rpm, which bounds the turbine speed after power uprate. The current high speed setting is 5300 rpm. The required flow coefficient across the feedwater control valve at maximum flow conditions increases to 2000 after the power uprate. The design flow coefficient listed in the system description is 2350. Thus, there is sufficient margin in the Main Feedwater Control Valve to produce the required flow.

3.7.4 Auxiliary Feedwater System (AFWS)

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 heat sink of the steam generators. The system provides feedwater to the OTSGs during accident conditions. The AFWS is required to prevent core damage during a loss of coolant accident and system OTSG overpressurization such as during a loss of normal feedwater.

The AFWS pumps are normally aligned to take suction from the two condensate storage tanks (CST). To fulfill the 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 minimum AFWS flow requirement is based on the loss of main feedwater (LOFW) transient. As discussed in Section 3.10.3.8, the LOFW event was analyzed based on an initial core power level of 102% of 2772 MWt, which bounds the proposed power uprate. Therefore,

the power uprate has no impact on the AFW flow control and flow rate provided for accident mitigation.

The Condensate Storage System is sized to contain two redundant tanks. The CSTs contain a total volume of 250,000 gallons each and a collective minimum usable volume of 250,000 gallons of water. The CST water is used for the removal of reactor decay heat. The CSTs are required to be operational for a period of 13 hours and then to cool down the Reactor Coolant System to 280°F (total of 19 hours), assuming the reactor is at 100 percent power at the time the cooling mode is automatically or manually initiated. Decay heat removal presently requires 91,620 gallons and sensible heat removal requires 57,280 gallons, for a total of 148,900 gallons. Increasing the decay heat removal requirement by 2% (91,620 gallons x 1.02) results in a decay heat removal requirement of 93,453 gallons after the uprate. Adding a sensible heat removal requirement of 57,280 gallons results in a total requirement of 150,733 gallons in the CST. Therefore, there is over 99,000 gallons of excess inventory available in the CST.

3.7.5 Steam Generator Blowdown System

The steam generator blowdown system is used during startup, shutdown, and at low power levels to stabilize steam generator water chemistry. The use of this system is administratively limited to power levels less than or equal to 15% RTP. This power level is low enough to allow monitoring feedwater flow on the startup feedwater flow indicators. The plant startup procedure requires that the system be isolated at 14%. The current 15% condition with respect to actual thermal power in MWt would be 14.75% (15%/1.017) after the power uprate. Therefore, the procedure limitations are not significantly changed.

3.7.6 Integrated Control System (ICS)

The ICS is a non-safety system that automatically controls the station in response to commands preset by the operator. The ICS provides control rod motion when CRDCS is in the automatic mode, normal feedwater control, and turbine control. The operator is also provided with the capability for manual override control of the station. The ICS was reviewed for the impact of the power uprate. The Maximum Continuous Rating (MCR) of the plant is expected to increase from 943 MWe to approximately 957 MWe. Total feedwater flow will increase from 11,600,000 lb/hr to 11,900,000 lb/hr at a Circulating Water temperature of 75°F. The increase in generator output will require minor adjustments to several ICS modules that use MCR to determine their settings. No additional ICS tuning is expected.

3.8 BALANCE-OF-PLANT SYSTEMS

3.8.1 Heat Balance

In addition to the benchmark heat balance, six heat balance cases were developed. Table 3-2 identifies the input conditions for each case. Case 3a represents the nominal heat balance (i. e. 75 °F Circulating Water temperature and current tube plugging) at current rated power. Case 3b represents the nominal heat balance (i. e. 75 °F Circulating Water temperature and current tube plugging) at the proposed power uprate. These two cases are included to provide a direct

comparison of the impact of power uprate with all other conditions the same. The boundary conditions are representative of the following: maximum and minimum circulating water temperature experienced during summer and winter operations, respectively; condenser cleanliness representative of the condenser in its current condition; and a maximum OTSG tube plugging. Case 1a represents the 0% tube plugging and winter condition. Case 1b represents the 0% tube plugging and summer condition. Case 2a represents the 20% tube plugging and winter condition. Case 2b represents the 20% tube plugging and summer condition.

The BOP systems were evaluated for the uprated conditions using the bounding data from these heat balances. Table 3-3 provides the BOP parameters for the power uprate. The results of the two heat balances with 75°F circulating water are shown in Table 3-3. Table 3-3 also shows the most bounding value for each parameter, along with a reference to the Table 3-2 case for which the bounding value is taken. The evaluations were performed at the bounding conditions.

Table 3-2 Heat Balance Cases

Boundary Conditions	Units	3a	3b	1a	1b	2a	2b	Bench-mark
Thermal Power	MWt	2772	2819	2819	2819	2819	2819	2752
OTSG Outlet Pressure	psia	930.6	931.5	931.4	931.4	931.8	931.8	930.3
Main Steam Throttle Pressure	psia	885.0	885.0	885.0	885.0	885.0	885.0	885.0
Circulating Water Temperature	°F	75	75	45	95	45	95	75
Condenser Cleanliness	%	(1)	(1)	(1)	(1)	(1)	(1)	(1)
OTSG Tube Plugging	%	(2)	(2)	0	0	20	20	(2)

Notes:

(1) The heat balance model was benchmarked against plant operating data. The condenser is operating at an approximate 95% cleanliness factor, and the heat balance is closely predicting actual condenser performance.

(2) The heat balance model was benchmarked against plant operating data. The OTSG model used in the heat balance was modified to match the NSSS model results for the proposed power uprate.

Table 3-3 BOP Parameters For Power Uprate

PARAMETER	Units	2772 MWt @ 75 degF CW	2819 MWt @ 75 degF CW	% Diff	Maximum at 2819 MWt	Case that is bounding
CT LP cond Inlet Temperature	degF	75	75	0.00%	95	1b,2b
CT HP Cond Outlet Temperature	degF	100.4	100.8	0.40%	121.2	1b
CT flow	#/hr	247497200	247497200	0.00%	248125408	1a, 2a
LP Cond Vacuum	in Hg	1.98	2.01	1.52%	3.38	2b
HP Cond Vacuum	in Hg	2.68	2.73	1.87%	4.55	2b
Condensate Temperature	degF	111.1	111.7	0.54%	130.2	2b
Cond Flow	#/hr	6857069	6985201	1.87%	7127676	2b
SJAE inlet Temperature	degF	112.9	113.5	0.53%	131.5	2b
SJAE outlet Temperature	degF	113.2	113.8	0.53%	131.7	2b
SPE outlet Temperature	degF	115.6	116.1	0.43%	134	2b
FW# 2Htr Condensate inlet Temperature	degF	168	168.8	0.48%	170.2	2b
FW # 2 htr Condensate outlet Temperature	degF	228.1	229.1	0.44%	229.3	2b
Deaerator Pressure	psia	70.45	71.8	1.92%	71.99	2a
Deaerator Temperature	degF	303.4	304.6	0.40%	304.8	2a
Feedwater Flow train 1	#/hr	6056442	6171678	1.90%	6248297	2a
Feedwater Flow train 2	#/hr	5620347	5729925	1.95%	5795694	2a
FW# 4 Htr inlet FW Temperature	degF	305.2	306.5	0.43%	306.7	2a
FW # 4 htr outlet FW Temperature	degF	331.2	332.5	0.39%	332.6	2a,2b
FW# 5 Htr outlet FW Temperature	degF	372.7	374.1	0.38%	374.2	1a,2a
FW # 6 htr outlet FW Temperature	degF	453.5	455.1	0.35%	455.4	1a
FW HTR # 1 Drain Temperature	degF	168.4	169.3	0.53%	170.6	2b
FW HTR # 1 Shell Pressure	psia	5.9	6	1.69%	6.1	1b, 2b
Heater Drain Pump Flow	#/hr	1096334	1120526	2.21%	1270422	2a
FW HTR # 2 Drain Temperature		186.7	187.8	0.59%	188.5	2b
FW HTR # 2 Shell Pressure	psia	20.7	21.1	1.93%	21.2	2b
Heater #2 Drain Flow	#/hr	535855	547803	2.23%	561623	2a
FW HTR # 4 Drain Temperature	degF	332.2	333.6	0.42%	333.9	2a
FW HTR # 4 Shell Pressure	psia	114.2	116.4	1.93%	116.7	2a
Heater #4 Drain Flow	#/hr	3166430	3224640	1.84%	3356900	2a
FW HTR # 5 Drain Temperature	degF	344.9	346.5	0.46%	346.9	2a
FW HTR # 5 Shell Pressure	psia	192.6	196.3	1.92%	196.8	2a
Heater #5 Drain Flow	#/hr	2885650	2937950	1.81%	3073460	2b
FW HTR # 6 Drain Temperature	degF	378.1	379.8	0.45%	380	2a
FW HTR # 6 Shell Pressure	psia	486.3	495.6	1.91%	496.4	1a

Table 3-3 BOP Parameters For Power Uprate

PARAMETER	Units	2772 MWt @ 75 degF CW	2819 MWt @ 75 degF CW	% Diff	Maximum at 2819 MWt	Case that is bounding
Heater #6 Drain Flow	#/hr	1498030	1532190	2.28%	1566600	2b
FW HTR # 1 Vent to LP Cond Temperature	degF	169.2	170	0.47%	171.1	2b
FW HTR # 1 Vent to LP Cond Flow	#/hr	70817	72364	2.18%	89578	2a
FW HTR # 1 Vent to HP Cond Temperature	degF	169	169.8	0.47%	170.9	2b
FW HTR # 1 Vent to HP Cond Flow	#/hr	69305	70815	2.18%	87621	2a
OTSG Steam Temperature	degF	590.3	590.3	0.00%	596	1a,1b
OTSG Steam Pressure	psia	930.6	931.5	0.10%	931.8	2a,2b
OTSG Steam Flow	#/hr	11676800	11901600	1.93%	12044000	2a
Steam to 2nd stage reheater Flow	#/hr	471723	479471	1.64%	499755	2a
Steam to 2nd stage reheater Temperature		584	583.9	-0.02%	589.7	1a,1b
Steam to SJAE Flow	#/hr	1500	1500	0.00%	1500	
Turbine Throttle Valve Flow	#/hr	11190770	11407830	1.94%	11529940	2a
Turbine Throttle Valve Pressure	psia	885	885	0.00%	885	
HPT Stm to MSep Flow	#/hr	9192280	9364230	1.87%	9449350	2a
HPT Stm to MSep Pressure	psia	203.7	207.8	2.01%	210.2	2a
HPT Stm to MSep Temperature	degF	383.4	385	0.42%	385.2	2a
MSep outlet Pressure	psia	202.7	206.7	1.97%	207.2	2a
MSep outlet Temperature	degF	382.9	384.6	0.44%	384.8	2a
1st stage MSep reheater outlet Pressure	psia	201.7	205.7	1.98%	206.2	2a
1st stage MSep reheater outlet Temperature	degF	431.9	433.2	0.30%	433.6	1a,1b
2nd stage MSep reheater outlet to MFPT and LPT Temperature	degF	489.3	490.3	0.20%	491	2a,2b
2nd stage MSep reheater outlet to MFPT Flow Train 1	#/hr	100212	101950	1.74%	110064	2b
2nd stage MSep reheater outlet to MFPT Flow Train 2	#/hr	90888	92462	1.73%	98458	2b
2nd stage MSep reheater outlet to MFPT & LPT Flow	#/hr	8733640	8905400	1.97%	8913140	2a
2nd stage MSep reheater outlet to MFPT & LPT Pressure	psia	199.9	203.8	1.95%	204.3	2a
HPT Extr Stm to MSep Reheater 1st stage Flow	#/hr	500440	508366	1.58%	527168	2b
HPT Extr Stm to MSep Reheater	psia	539.6	550.3	1.98%	550.9	1a

Table 3-3 BOP Parameters For Power Uprate

PARAMETER	Units	2772 MWt @ 75 degF CW	2819 MWt @ 75 degF CW	% Diff	Maximum at 2819 MWt	Case that is bounding
1st stage Pressure						
HPT Extr Stm to MSep Reheater 1st stage Temperature	degF	483.3	486.5	0.66%	491.8	1a
HPT Extr Stm to FW HTR #6 Flow	#/hr	1015510	1041920	2.60%	1056180	2b
HPT Extr Stm to FW HTR #6 Pressure	psia	539.6	550.3	1.98%	550.9	1a
HPT Extr Stm to FW HTR #6 Temperature	degF	483.3	486.5	0.66%	491.8	1a
HPT Extr Stm to FW HTR #5 Flow	#/hr	451012	461406	2.30%	468704	2a
HPT Extr Stm to FW HTR #5 Pressure	psia	205.6	209.6	1.95%	210.2	2a
HPT Extr Stm to FW HTR #5 Temperature	degF	384.1	385.8	0.44%	386	2a
MSep drain tank to FW Htr #5 Flow	#/hr	458645	458825	0.04%	537512	2b
MSep drain tank to FW Htr #5 Temperature	degF	382.9	384.6	0.44%	384.8	2a
MSep drain tank to FW Htr #5 Pressure	psia	202.7	206.7	1.97%	207.2	2a
1st stage MSep reheater outlet to FW # 5 Temperature	degF	464.1	466.1	0.43%	466.2	1a,1b
1st stage MSep reheater outlet to FW # 5 Pressure	psia	485.8	495.5	2.00%	494.1	2a
1st stage MSep reheater outlet to HP Cond Temperature	degF	463.4	465.5	0.45%	465.6	1a,1b
1st stage MSep reheater outlet to HP Cond Flow	#/hr	22480	22836	1.58%	23608	2b
2nd stage MSep reheater outlet to FW #6 Pressure	psia	863.7	863.6	-0.01%	863.8	1a,1b
2nd stage MSep reheater outlet to FW #6 Temperature	degF	527.1	527.1	0.0%	527.1	1a,1b
LPT Extr Stm to FW HTR #1 Flow	#/hr	562122	575583	2.39%	745600	2a
LPT Extr Stm to FW HTR #1 Pressure	psia	6.06	6.18	1.98%	6.33	2b
LPT Extr Stm to FW HTR #1 Temperature	degF	169.2	170	0.47%	171.1	2b
LPT Extr Stm to FW HTR #2	#/hr	535855	547803	2.23%	561623	2a

Table 3-3 BOP Parameters For Power Uprate

PARAMETER	Units	2772 MWt @ 75 degF CW	2819 MWt @ 75 degF CW	% Diff	Maximum at 2819 MWt	Case that is bounding
Flow						
LPT Extr Stm to FW HTR #2 Pressure	psia	22.97	23.41	1.91%	23.45	2b
LPT Extr Stm to FW HTR #2 Temperature	degF	233.8	234.8	0.43%	234.9	2b
LPT Extr Stm to Deaerator Flow	#/hr	554961	569240	2.57%	572798	1a
LPT Extr Stm to Deaerator Pressure	psia	74.39	75.8	1.90%	76.03	2a
LPT Extr Stm to Deaerator Temperature	degF	308.2	309	0.26%	309.8	2b
LPT Extr Stm to FW HTR #4 Flow	#/hr	280782	286686	2.10%	290087	1a
LPT Extr Stm to FW HTR #4 Pressure	psia	122.7	125.1	1.96%	125.4	2a
LPT Extr Stm to FW HTR #4 Temperature	degF	395.3	396.1	0.20%	397	2b
LPT Extr Stm to FW HTR #1A Flow	#/hr	130184	131648	1.12%	131874	1a
LPT Extr Stm to FW HTR #1A Pressure	psia	11.99	12.22	1.92%	12.25	2b
LPT Extr Stm to FW HTR #1A Temperature	degF	200.4	201.4	0.50%	201.5	2b
Gross MWe	MWe	943	957.1	1.50%	962.5	1a

3.8.2 Condensate System and Condenser

The primary function of the condensate system is to supply preheated condensate, via the feedwater heater trains, to the suction of the steam generator feedwater pumps. The condensate system pressure, temperature, and flow rate will change slightly at the uprate power level. However, these parameters will still remain below the system and component design conditions. The condensate system pumps must be able to supply stable condensate flow to the deaerator under all normal operating conditions. The condensate system flow is controlled by the deaerator level control valve located at the inlet to the deaerator. At the uprated conditions, the condensate pumps are able to supply flow to the deaerator maintaining stable level control in the deaerator, and adequate NPSH is provided at the condensate pump impeller during normal operating conditions.

Steam flow to each condenser will increase as a result of the power uprate. However, the uprate conditions are bounded by the condenser design.

The existing design basis is exceeded for the condensate flow rate in the Steam Jet Air Ejector (SJAE) tubes at current power levels. Tube velocities will be slightly increased at uprate conditions. Periodic preventive maintenance inspections will be conducted to monitor wear in the SJAE.

Low Pressure Feedwater Heaters

The increase in condensate flow due to power uprate was evaluated in the low pressure feedwater heaters. The low pressure feedwater heaters are designed to operate at 20% above the original design flow. This bounds the normal operating flow after the proposed power uprate except for the Feedwater Heater #1 shell side flow, which increased to approximately 23.5% above the design flow. Periodic preventive maintenance inspections will be conducted to monitor feedwater heater #1 shell side wear.

The increase in tube side flow also increases the required feedwater heater #2 shell side relief valve capacity due to a tube rupture. The shell side relief valve capacity has been evaluated and found to be acceptable.

3.8.3 Feedwater System

The feedwater system supplies heated feedwater to the steam generators under all load conditions. The feedwater system flow is regulated by the Integrated Control System, which is addressed in Section 3.7.6. The Main Feedwater Stop Valves, Main Feedwater Control Valves, and Feedwater pumps are addressed in Section 3.7.3.

High Pressure Feedwater Heaters

The increase in feedwater flow due to power uprate was evaluated in the feedwater heaters. The feedwater heaters are designed to continuously operate at 20% over the original design flow. Since this flow bounds the maximum operating flow after power

uprate, the ability of the feedwater heater tubes to handle the increased flow is not impacted. The increase in tube side flow also increases the required feedwater heater shell side relief valve capacity due to a tube rupture. The shell side relief valve capacities have been evaluated and found to be acceptable. For the power uprate, the feedwater flow rate will increase slightly for each unit but will remain below system design capabilities.

Deaerators and Deaerator Storage Tank

The deaerators heat and scrub incoming feedwater, heater drains, and water from miscellaneous sources to remove air and other non-condensable gases. At the increased feedwater flows after power uprate, the ability of the deaerators to remove non-condensables from the feedwater system will be verified by on-line chemistry testing.

The Deaerator Storage Tank is sized to provide a minimum of 5 minutes of water capacity at approximately 30" below the top of the tank. The power uprate results in a minimum capacity of 4.67 minutes of water being available. This parameter is not used in any transient or accident analysis.

3.8.4 Extraction Steam System

The extraction steam system transmits steam from the high- and low-pressure main turbines to the shellside of the feedwater heaters for feedwater heating. During normal operation, steam from the high-pressure turbine is used to heat feedwater flowing through the fifth and sixth stage heaters, and steam from the low-pressure turbines is used to heat feedwater flowing through the deaerator, first, second, and fourth stage heaters.

Extraction steam flow increases approximately 3-5% from the current steam flows. The design flow bounds the maximum flows after power uprate for the extraction lines between the HP Turbine and Feedwater Heaters #6 and #5. The flows from the LP Turbine to Feedwater Heaters #1, #2, the Deaerator, and Heater #4 exceed the original design. Periodic preventive maintenance inspections will be conducted to monitor wear due to the increased flows from the LP Turbine to Heaters #1, #2, the Deaerator, and Heater #4.

The tube side flow through the Moisture Separator Reheaters increased approximately 3% from current conditions, and the inlet pressure increased approximately 6 psi. The tube side flow is slightly higher than design flow, however this is not expected to have a significant effect on the performance of the MSRs. The pressure is bounded by the design pressure of the MSR, and MSR controls have been verified to be acceptable. However, shell side flow through both the first stage and second stage reheaters exceed design. Periodic preventive maintenance inspections of the MSR will be conducted to monitor wear.

The flow element in the steam line from the MSR to the Main Feed Pump Turbine and the flow element in the second stage reheater shell side drain may be slightly outside their ranges at high flow conditions. Since these flow instruments provide only non-essential monitoring capability

to secondary side systems, and provide no control or safety function, they are not being modified at this time.

3.8.5 Heater Drains System

The heater drains system and associated equipment were evaluated to ensure the ability of the system to function under power uprate conditions. Heater drain design parameters were reviewed and compared against power uprate conditions to determine that acceptable design margin exists for operation at uprate conditions.

Pressures and temperatures associated with the power uprate will remain bounded by the existing designs of the heater drain systems and its components. Heater drains flows increase approximately 3-5% from the current nominal steam flows. The design flow still bounds the maximum flows after the proposed power uprate for the heater drain lines between Feedwater Heater #6 and #5, and between Feedwater Heaters #5 and #4. The Heater Drain valve between Heater #4 and the Deaerator is currently in the wide open position for full power operation and is planned to be replaced in 13RFO with a higher capacity valve. Additionally, it is planned to replace the low pressure feedwater heater drain tank level control valves at the discharge of the heater drain tank pumps with higher capacity control valves in 13RFO. Due to limited capacity of the heater drain pumps, the levels of the heater drain tanks will be monitored when extraction steam flow across the LP feedwater heaters is greatest (cold weather conditions). Should the level in the feedwater heater increase and/or result in bypass of the heater drain water to the condenser, overall cycle thermal efficiency will be decreased.

3.8.6 Circulating Water System

The Circulating Water System (CT) is a closed-loop system that provides cooling water for the main condenser of the turbine generator unit. The total operating circulating water flow rate to the cooling tower is approximately 495,600 gpm.

The CT flow will remain essentially unchanged following power uprate. The increased levels of rejected heat, from an increase in turbine exhaust flow, will increase the CT outlet temperature by approximately 0.5°F. The heat load under power uprate conditions will result in a slight backpressure increase in the condenser. However, the increased backpressure will remain within acceptable limits. The increase in outlet temperature, due to the increased heat load, can be accommodated by the cooling tower. A slight increase in evaporation rates can also be expected, requiring an increase in makeup rates under maximum summer conditions (less than a 2-percent increase). This slight increase is within the capability of the makeup supply service water system. The condenser vacuum system and steam jet air ejectors will also continue to support reliable plant operation at uprate. No modifications to the CT or its components are required for a power uprate.

3.8.7 Component Cooling Water System

The Component Cooling Water (CCW) system provides an intermediate cooling loop for removing heat from reactor plant auxiliary systems and transferring it to the service water system.

During normal operation, one CCW pump and one CCW heat exchanger have more than sufficient capacity to transfer the design heat load from the components served. The CCW system is designed to supply 97°F water at the maximum allowable service water temperature of 90°F to the components cooled under all modes of operation. The spent fuel pooling cooling loads have been evaluated at 102% of rated thermal power. The increase in heat loads will have an insignificant effect on the Component Cooling Water system. The component cooling water system heat removal requirements for the decay removal heat exchangers at uprated conditions are bounded by existing analysis. The peak CCW supply temperature during a design basis accident coincident with a loss of the non-safety related lake intake canal remains less than the 120°F allowable temperature following the power uprate. The CCW systems will continue to remove the required heat loads under normal conditions without exceeding their design temperature limits at uprate. Since the heat load increase due to the uprate is small, no modifications or changes in flow rates and operating limits are required.

3.8.8 Service Water System/Ultimate Heat Sink (UHS)

The Service Water (SW) system provides cooling water to various safety-related and non-safety-related equipment. The power uprate will slightly increase the heat rejection. However, the SW system design pressure and temperature will not be exceeded by the uprate. During normal power operation, the power uprate will increase heat loads on the Turbine Plant Cooling Water (TPCW) system, increasing the SW flow required.

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

DBNPS License Amendment No. 242 revised the Technical Specification Limiting Condition for Operation (LCO) 3.7.5.1.b limit on Ultimate Heat Sink temperature from 85°F to 90°F. It was recognized during development of the associated license amendment application that a 90°F Ultimate Heat Sink temperature, which corresponds to the SW supply temperature, would not permit a TPCW system temperature of 85°F at the outlet of the TPCW Exchanger to be maintained. It was determined that with the Service Water system temperature approaching the 90°F TS limit, careful monitoring of TPCW-cooled components would be required, and that appropriate actions, including possible turbine load reductions, would be taken to ensure acceptable equipment operating conditions are maintained. Although the Service Water System will experience slightly higher heat loads during normal operation following the uprate, the existing system will continue to satisfy its normal and accident functions with no modifications being required to the system.

The analyses performed in support of DBNPS License Amendment No. 242 evaluated the elevated temperatures in the intake canal forebay assuming the intake canal loses its connection to Lake Erie and assuming a concurrent LOCA. The mass and energy releases following a LOCA are discussed in Section 3.11.2. The analyses assumed a decay heat load based on an initial power level of 3025 MWt (102% of 2966 MWt), using the equations given in Standard Review Plan 9.2.5 Branch Technical Position (BTP) ASB 9-2, "Residual Decay Energy Release Rate for Light-Water Reactors for Long-Term Cooling." Therefore, this analyses bounds the 1.63% power uprate request.

3.8.9 Turbine Plant Cooling Water System

The Turbine Plant Cooling Water (TPCW) system provides an intermediate cooling loop for removing heat from the turbine plant auxiliary systems and transferring it to the Service Water system. The system removes heat from designated non-safety-related turbine plant components. The TPCW system was evaluated to determine the impact due to the uprate. The results of the evaluation showed that the power uprate would slightly increase the system heat load for this system. As described in Section 3.8.8 above, an elevated SW temperature has the potential to adversely affect the TPCW system, and TPCW-cooled components will need to be carefully monitored under that condition.

3.8.10 Containment Air Coolers (CAC), Containment Spray System, and Containment Recirculation System

These systems are designed to provide the necessary cooling and depressurization of the containment following a LOCA. The LOCA analyses have been performed at a power level that bounds the core power uprate, reference section 3.11.2.1. Therefore, the CAC system, including long-term post-LOCA containment sump water cooling via the DHR heat exchangers, and the containment spray system are not impacted by the uprate.

The CAC system is also used for normal operation cooling of the containment. The capability of the CAC system to maintain the containment environment during normal operation is not impacted by the power uprate.

The containment recirculation system is used for ventilation of the containment, eliminating temperature stratification, and to assist in dispersing and dissipating gaseous accumulations and pockets that may exist. This function is not affected by the power uprate since the containment atmosphere during normal power operation remains unchanged after the power uprate.

3.8.11 Piping, Pipe Supports and Pipe Whip

The piping systems evaluated for the power uprate included the reactor coolant (including primary loop piping, primary equipment nozzles, primary equipment supports, and auxiliary piping), main steam, feedwater, high-pressure heater drains, and circulating water. The evaluation performed concluded that these piping systems remain acceptable and will continue to satisfy the design basis requirements in accordance with applicable design basis criteria, when considering temperature, pressure, and flow rate effects resulting from the power uprate

conditions. The DBNPS piping and related support systems remain within allowable stress limits in accordance with the ASME Section III Boiler and Pressure Vessel Code, 1971 edition and ANSI B31.1 Power Piping Code, 1967 edition and its addenda for class D piping. The evaluation also concluded that no piping or pipe support modifications are required as a result of the increased power level. Due to potential small increase in pipe stresses, no new postulated pipe break locations were identified in high-energy piping.

3.8.12 Turbine Generator

The capability of the Turbine Generator to perform at the proposed uprated power conditions was evaluated by the manufacturer, General Electric in a feasibility study. The review included the throttle valves, high-pressure and low-pressure turbines, the generator and exciter, as well as associated auxiliary equipment including moisture separator reheater controls and relief valves. All turbine generator components were determined to have sufficient margin to enable operation at the uprated power conditions without requiring equipment modifications, except for the sequencing of control valve operation, which will be modified in the next refueling outage (13RFO). This modification is necessary to ensure high pressure turbine first stage bucket design limits are not exceeded. The Control Valve Diode Function Generator (DFG) cards will be recalibrated to accommodate the change of the control valve sequencing.

The existing turbine missile analysis was reviewed for the uprated power level. The turbine missile protection is based on the low probability of damage to safety-related equipment from both high and low trajectory turbine missiles. This probability is based on turbine speed, inspections and intervals of inspection, which are not impacted by power uprate. Therefore, it is concluded that the proposed uprate is bounded by the existing turbine missile analysis.

The generator components are acceptable for operation at the uprated power level as long as operation of the unit remains within the original capability curves. The power uprate could result in an output as high as 962.5 MWe with a Circulating Water temperature of 45°F. This could result in operation at either a higher power factor or limit operation to lower than the new 100% rated thermal power. See Section 3.9.1 for further details.

3.9 ELECTRICAL SYSTEMS

3.9.1 AC and DC Plant Electrical Systems

3.9.1.1 Electrical Distribution System

The electrical distribution systems were reviewed to identify the major items that may be affected by uprate conditions and to evaluate the potential impact of an uprate on that equipment. Additional details are provided in the following sections.

System reviews confirmed that only large, non-safety-related, ac-powered loads were affected by unit operation at core uprate conditions. Additionally, the reviews confirmed that control of the affected loads remained unchanged. Accordingly, the direct current systems are unaffected by unit operation at uprate conditions.

3.9.1.2 Main Generator

The main generator is a four-pole machine rated at 1,068,794 KVA and 25 KV with an operating point of 961.9 MWe at a 0.9 power factor. This rating is based upon 60-psig hydrogen pressure that is supplemented with cooling water for the stator. At the current thermal rating of 2772 MWt, the main generator electrical output is typically 943 MWe. The turbine generator and auxiliaries have been evaluated for operation at the uprated conditions. A review of the applicable generator reactive capability curve confirms that the main generator is capable of operating at a maximum real power output of 1068 MWe at a 1.0 power factor (zero megavar output). Heat balance studies completed for the uprate identify gross generator output levels less than this maximum. Machine operation at a lower real output power level and a power factor of 1.0, or less, is permissible provided unit operation remains within the real and reactive power limits defined by the reactive capability curve.

3.9.1.3 Isophase Bus

The isolated phase bus duct and associated cooling equipment are designed to accept the maximum generator output (1068 MWe) and therefore will continue to support plant operations under uprated conditions.

3.9.1.4 Main Transformer

The Main Transformer has a rating of 980 MVA. Under some conditions (i.e., winter with 45°F Circulating Water) this could limit the generator power output to below the levels commensurate with the uprated power level and power factor of 0.9.

3.9.1.5 Auxiliary and Startup Transformer

The bus loading summaries for connected 4,160V switchgears under uprate conditions remain less than the Auxiliary Transformer and Startup Transformer design ratings. The associated cooling equipment will also support power uprate for continuous operation with no modifications.

The cables that connect the Startup Transformer and Unit Auxiliary Transformer to the 13.8 KV switchgears have a continuous rating of 1642 Amps per phase at 13.8 KV. The isolated phase bus tap at the Unit Auxiliary Transformer is rated continuously at 4000 amps.

The high voltage windings of Bus Tie transformers AC and BD are connected by means of 9 by 350 kcmil, 15 kV cables to the 13.8 kV, 1200 A circuit breakers HAAC and HBBD of 13.8 kV Buses A and B, respectively. These cables have a continuous rating of 1422 Amps per phase at 13.8 kV. The Bus Tie Transformer AC and BD low voltage windings are connected to the 4160 V Switchgear by cable bus systems. These cables have a continuous rating of 2000 Amps per phase at 13.8 kV.

The bus loading summaries for the connected switchgears under uprate conditions confirm that the connecting cables are adequate.

3.9.1.6 Large Loads and Cables

System evaluations have determined that the condensate pump and heater drain pump motors on non-safety-related 4,160V switchgears experience a slight brake horsepower (BHP) load change at power uprate conditions, from present loading requirements. An evaluation performed for 4,160V bus loads under uprated conditions verified acceptable loading. Therefore, the large station auxiliary loads and associated cables are considered adequate as installed, and the motors will continue to satisfactorily perform their intended functions.

3.9.1.7 Diesel Generators

The ESF (safety-related) motors do not experience a load change at uprated conditions. Therefore, the diesel generators will not be impacted by the power uprate and will remain capable of performing their safety-related functions during a LOOP/LOCA.

3.9.1.8 Protective Relay Settings

All other electrical equipment and components, including station protective schemes and setpoints, will continue to support safe and reliable plant operation at uprate. Bus voltage and fault current values at different levels of the station auxiliary electrical distribution systems will remain within acceptable limits under uprate. In addition, there are no impacts to the DC power system voltage or short circuit current levels.

3.9.1.9 Switchyard

The switchyard equipment exceeds the nameplate rating of the main generator. All 345 kV switches, breakers, and buses are rated at 2000 amperes, which exceeds the main generator maximum output current of approximately 1790 amperes at its nameplate rating of 1068.8 MVA. The switchyard will accept the additional load without the need for any hardware modifications.

3.9.2 Grid Stability

The DBNPS receives shutdown power from three physically independent and redundant offsite power sources of the 345 kV switchyard system. Under 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 analysis. The additional power generated under the proposed uprate has no significant impact on the 345 kV switchyard system and the ability of the plant to safely shut down.

An Office of Nuclear Regulatory Research report, "The Effects of Deregulation of the Electric Power Industry on the Nuclear Plant Offsite Power System: An Evaluation," dated June 30,

1999, recommended that grid stability analyses be updated by licensees periodically to reflect changes in the grid power system. A grid stability study for the DBNPS was performed in May 2000. This study included a 10% increase in gross power output, based on a potential future turbine-generator uprate. Increases in the generator rating (1120 MVA) and power factor (0.92) were used to increase the power output to 1033 MW from the current maximum gross power output of 942 MW, at a rating of 1069 MVA, and a nominal power factor of 0.88. The change in power factor also reduced the unit's reactive power capability. The grid stability study showed that the system response to one additional contingency (Contingency #4) was unstable as a result of this uprate. This contingency is a three-phase fault at the Bayshore 345 KV bus at 4.5 cycles, cleared by tripping the Bayshore breakers and the DBNPS breakers at 22.5 cycles. A letter dated July 19, 2000 from the Transmission Group/ATSI states that three-phase faults with delayed clearing are "Extreme Events" classified as Category D in National Electricity Reliability Council (NERC) Planning Standards, Section I.A. These are principally three-phase faults, some with delayed clearing, and may involve outages to complete transmission or generating stations. For events in this category, the system may experience substantial, wide spread loss of load and generation, and the system may not achieve a new, stable operating point. Category D events are considered in order to judge the robustness of the system (the limits to which the system may be "pushed"). Although this letter was intended to accept the unstable response to Contingency #8 (which is a three phase fault that occurs with the current maximum generator output of 945 MW as well as with the uprated output), the justification presented also applies to Contingency #4. This letter has been updated to include the acceptance of the unstable response in Contingency #4 as discussed. The unstable conditions identified are the result of contingencies that do not have to be considered. Other than Contingencies #4 and #8, there are no other unstable conditions.

Since the generator rating assumed in the grid stability study substantially bounds the proposed uprate, the study demonstrates acceptability. The DBNPS will continue to meet the intent of GDC 17.

3.10 NUCLEAR STEAM SUPPLY SYSTEM ACCIDENT EVALUATION

3.10.1 LOCA Related Analyses

Loss of coolant accidents are performed by NRC approved models and methods to demonstrate compliance with 10CFR 50.46. The criteria are:

- The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

- The calculated total oxidation of the fuel cladding shall not exceed 17% of the total cladding before oxidation.

- The amount of hydrogen generated from cladding metal-water reaction does not exceed 1% of the total amount of cladding in the reactor.

- The core geometry is maintained in a state that is amenable to cooling.

- The cladding temperature is reduced and maintained at an acceptably low value and decay heat is removed for extended periods of time.

The current LOCA analyses are based on an initial power level of 3025 MWt (102% of 2966 MWt). These analyses fully comply with the acceptance criteria listed above. The small power uprate that is being requested under this licensing change request is less than what has been analyzed. Therefore, the criteria of 10CFR 50.46 will not be challenged.

3.10.1.1 LBLOCA and SBLOCA

A spectrum of break sizes and break locations is postulated in the primary coolant piping. The LOCAs are considered limiting fault transients, events that are not expected to occur, but are postulated because of the potential for large releases of radiation. The acceptance criteria relate to ensuring adequate core cooling for the short and long-term post-LOCA, limiting reactor building pressure and temperature, and limiting offsite dose consequence.

A spectrum of LOCAs has been reanalyzed for the DBNPS using an initial power level of 3025 MWt (102% of 2966 MWt). The results bound the power uprate. New mass and energy release rates have also been generated for the DBNPS based on an initial power level of 3025 MWt.

3.10.1.2 Post-LOCA Long-Term Core Cooling (LTCC)

The requirements of 10CFR50.46(b)(5), "Long-term cooling," pertains to maintaining the reactor shut down by borated ECCS water residing in the RCS/containment emergency sump following a LOCA. Since credit for the control rods is not taken for large break LOCA, the borated ECCS water provided by the BWST and Core Flood Tanks must have a 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 sources of 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/DHR piping, the Makeup tank (MUT) and MU&P piping. The water volumes and associated boric acid concentrations are not directly affected by the power uprate. The cycle-specific core re-load licensing process provides confirmation that these volumes and concentrations are adequate. Thus, there is no impact on the LTCC analysis.

3.10.2 Reactor Vessel, Loop, And Steam Generator LOCA Forces Evaluation

The revised design conditions were reviewed for impact on the existing hydraulic forcing functions and the high energy line break (HELB) locations in the primary RCS piping and the piping attached to the primary RCS to the first anchor. The HELB locations chosen were those that would have an impact on the RCS components. The evaluation showed that the asymmetric cavity pressure forces, thrust loads, and jet impingement loads remain bounded by the values in

the existing analyses. The evaluation also showed that there are no additions or changes to the HELB locations or loads.

3.10.3 Transient Analyses

A review of the USAR Chapter 15 accidents was performed to support the power uprate. A summary of the evaluation for each accident is provided below.

3.10.3.1 Uncontrolled Control Rod Assembly Group Withdrawal from a Subcritical Condition (Startup Accident)

The startup accident is a postulated event that results from a withdrawal of control rods with the reactor slightly subcritical and at zero power thermal conditions. This event is classified as a moderate frequency event. The acceptance criteria for the event relates to peak reactor coolant system (RCS) pressure and maximum allowed core power. The primary RPS trip functions that are credited are the high RCS pressure and high flux trips. A faster reactivity insertion rate than is possible with the hardware is modeled, which ensures that a conservative calculation is performed.

This transient is considered a heat-up transient that results in pressurization of the RCS. The startup accident is the limiting overpressure event for the RCS. This is a very fast transient. Since the event is initiated from subcritical conditions, the power uprate will not have an effect on the initial conditions. As described in the license amendment application, the RPS high flux setpoint is being reduced to preserve the maximum power condition. Therefore this event is not affected by the power uprate.

3.10.3.2 Uncontrolled Control Rod Assembly Group Withdrawal at Power Accident

The rod withdrawal at power event is similar to the startup accident, i.e., the event classification and acceptance criteria are the same. The transient is initiated from full power conditions, but the reactivity insertion rate is limited to a maximum control rod movement of 30 inches per minute. The withdrawal of a control rod group at power, caused by either operator error or equipment failure, results in positive reactivity addition. As the positive reactivity addition increases, core power level increases. The increase in core power causes fuel rod temperatures to rise and increases the heat transferred to the reactor coolant. The increase in core power creates a mismatch between core power generation and secondary heat removal. The heat mismatch causes reactor coolant temperature and pressure to increase. The transient is terminated by a reactor trip on high RCS pressure or high flux. The reactor trip limits the peak core thermal power to an acceptable level. The reactor trip and subsequent steam relief through the primary safety valves ensures that the peak primary pressure meets the acceptance criterion.

The initial core power level for the Rod Withdrawal at Power accident analyses is 2772 MWt. The high flux trip setpoint used in the analyses was 112 percent of 2772

MWt. The 112 percent high flux trip setpoint includes a 2 percent heat balance error. Including the heat balance error in the high flux trip setpoint versus starting from an initial power level of 102 percent of 2772 MWt, results in the same core thermal power at the time of reactor trip. In addition, starting with an initial power level of 2772 MWt and including the heat balance error in the high flux trip setpoint maximizes the increase in core power from the initial power level to the power level at the time of trip. Maximizing the increase in core power between event initiation and the time of reactor trip results in a larger energy mismatch between core heat generation and secondary heat removal. This produces a higher peak RCS pressure with all other things remaining the same. Therefore, the Rod Withdrawal at Power accident analyses performed at 2772 MWt remains acceptable for the power uprate.

3.10.3.3 Control Rod Assembly Misalignment (Stuck-Out, Stuck-In, or Dropped Control Rod Assembly)

The dropped control rod accident bounds the stuck-out and stuck-in cases. The dropped control rod transient is a moderate frequency event. The acceptance criteria relate to peak RCS pressure and minimum departure from nucleate boiling ratio (DNBR). The limiting time-in-life for the dropped control rod accident is near the middle of life (MOL) when the combination of moderator temperature coefficient (MTC) and the worth of the dropped control rod are sufficient to prevent reactor trip on low RCS pressure. In this case, a new steady-state operating condition is obtained. The MOL transient provides the greatest challenge to minimum DNBR because at beginning of cycle (BOC) or end of cycle (EOC), depending on the worth of the control rod, the reactor will trip on low RCS pressure (BOC) or on high flux and high pressure (EOC). Conservative reactivity parameters coupled with a spectrum of cases, where the dropped rod worths and MTCs are varied, ensure a bounding calculation.

The dropped control rod accident was reanalyzed for the power uprate. The power level used in the calculation was 3025 MWt (102% of 2966 MWt), which bounds the power uprate. The results of the analysis were acceptable for the power uprate to 2817 MWt.

3.10.3.4 Makeup and Purification System Malfunction (Moderator Dilution) Accident

The moderator dilution accident (MDA) is a moderate frequency event and results from an uncontrolled dilution of the primary coolant. The reactivity addition results in an increase in power similar to a rod withdrawal accident. The acceptance criteria for this accident relate to peak RCS pressure, maximum allowed power, and minimum subcritical margin. Conservative reactivity parameters and dilution flow rates are modeled to ensure a bounding calculation.

The analysis was initiated from 100% rated thermal power (RTP). This event progression is determined by the combination of the dilution flow rate and the cycle-specific reactivity parameters. Typically, the earlier cycle designs resulted in greater reactivity additions, such that they are more limiting than the current cycle designs. The

acceptance criteria for this event is that peak power will not exceed 112% RTP and that the peak RCS pressure will be less than 110% of the design. The moderator dilution accident is a relatively slow event such that neutron and thermal power increases at approximately the same rate. As the power increases, the RCS pressure will also increase. Since it is a slow transient, the pressurizer safety valve will be more than sufficient to prevent over pressurizing the RCS.

The high flux setpoint is based on a maximum allowed power level, typically set at 112% of RTP. The actual in-plant setpoint is determined based on the overpower setpoint adjusted for uncertainties. A reanalysis of all of the events that require reactor trip on high power for mitigation has not been performed. For those analyses, the trip setpoint is not input as percent of power but rather an ultimate power level. This ultimate power level will not be changed. Instead, the overpower setpoint will be reduced in order to preserve, or limit the actual power. The license amendment application describes the proposed change to the RPS high flux setpoint.

3.10.3.5 Loss of Forced Reactor Coolant Flow (Partial, Complete, and Single Reactor Coolant Pump Locked Rotor) Accident

The loss of coolant flow (LOCF) accidents result from either loss of power or mechanical failure of one or more of the reactor coolant pumps (RCPs). The LOCF accidents are comprised of three different transients. The simultaneous coastdown of all four RCPs is considered an infrequent event. The single RCP (4-to-3) coastdown is considered a moderate frequency event. The single locked pump rotor is considered a limiting fault transient. Although the four-pump (4-to-0) coastdown is considered an infrequent event, it is analyzed to the more restrictive criteria of the moderate frequency event category. These events are evaluated for each new fuel reload. The acceptance criteria for these events relate to the minimum allowed DNBR for each specific category of accident.

The primary trip for the 4-to-0 pump coastdown is the power-to-pump monitor. The primary trip for the 4-to-3 pump coastdown and the locked rotor transient is the power/imbalance/flow (P/I/F) trip, or specifically, the flux-to-flow setpoint. The system response is not affected by the initial conditions relating to the core power level. Once the control rods begin to insert, the DNBR will begin to increase and the transient is terminated. Therefore the system response will not be affected. The core power, RCS pressure, RC flow and core inlet temperature are normalized. For each new fuel cycle, the normalized data is applied to bounding conditions, and the minimum DNBR is recalculated. Since this analysis is performed for each new reload, a specific evaluation for the power uprate is not required.

3.10.3.6 Startup of an Inactive Reactor Coolant Loop (Pump Startup Accident or Cold Water Accident)

This transient results from the startup of an idle loop while the plant is operating at reduced power. The cold water accident (CWA) is a moderate frequency event and the acceptance criteria relate to peak RCS pressure and minimum DNBR.

Even though there is a licensing restriction that prohibits the plant from being critical with less than three reactor coolant pumps operating, the analysis assumed that the plant was operating with one reactor coolant pump in each loop at 50% of rated power when the remaining two pumps were started. The increase in primary coolant flow and negative reactivity coefficients result in a positive reactivity insertion and subsequent increase in core power. The increase in core power limits the primary coolant temperature decrease and the plant reaches equilibrium at a new power level below the rated core power. The increase in coolant flow combined with an increase in power does not result in an unacceptable minimum DNBR. The RCS pressure remains below the high pressure reactor trip setpoint.

The Cold Water Accident is analyzed from 50 percent of rated core power. An increase in rated core power will not change the dynamics of the event evolution and will not result in a violation of the event acceptance criteria. In addition, the licensing restriction on operation specifies that three RCPs must be operating when the reactor is critical. The licensing restriction limits the consequences of the Cold Water Accident to the startup of one RCP. In either case, the maximum power increase is to approximately 80%. Since this is less than during normal operation, the DNBR will be bounded. Initiating the event from a slightly higher power level will not invalidate this conclusion.

3.10.3.7 Loss of External Load, Turbine Trip, Loss of AC (Offsite) Power and/or Station Blackout

The station blackout and loss of AC power transients are caused by a loss of power. The loss of electrical load and turbine trip events are a consequence of the failure of the turbine or closure of the turbine stop valves. The loss of external load, the turbine trip, and the loss of offsite power (LOOP) accidents are considered moderate frequency events. The total station blackout, with reactor coolant pump seal leakage, is considered beyond the original design bases of the plant, however, the DBNPS added a separate and independent station blackout diesel to provide an alternate source of AC power, as described in Section 4.3. The acceptance criteria for these accidents relate to peak RCS pressure and minimum DNBR. The station blackout transient, since it is also a loss of primary coolant event, also requires that adequate core cooling be maintained. The loss of external load that is reported in the USAR is for historical purposes. The original plant design would allow for the plant to runback from full power conditions to prevent a reactor trip. As part of NUREG-0737, the pilot operated relief valve (PORV) lift setpoint was increased to a value above the high RCS pressure trip setpoint. As a result, the plant will trip if the transient is initiated from full power conditions. The loss of power transient will result in a reactor trip without actuation of the RPS because the reactor trip breakers will de-energize allowing the control rods to drop. The transient evolves into a steady-state natural circulation condition assuming auxiliary feedwater (AFW) initiation. AFW initiation will be on low steam generator level or loss of main feedwater flow. The power uprate will not have an effect on these events.

The turbine trip is the limiting overheating (overpressure) event for the steam generators. The reactor is tripped on high RCS pressure. The anticipatory reactor trip (ARTS) on

turbine trip is not credited. A turbine trip analysis was recently performed at an analyzed power level of 3025 MWt and used the current installed valve capacity. This analysis confirmed that the peak OTSG pressure was less than the ASME code allowable. The small increase in power that is proposed is bounded by the current analysis.

3.10.3.8 Loss of Normal Feedwater, Feedwater Line Break, and Total Loss of All Feedwater

The loss of normal feedwater due to pump failure or valve closure is considered a moderate frequency event. The acceptance criteria relate to peak RCS pressure and minimum DNBR. Additional criteria may also be imposed, i.e., peak pressurizer liquid level and average shell to average steam generator tube temperature difference. The feedwater line break event is considered a limiting fault event. The acceptance criteria are peak RCS pressure and offsite dose. Although the feedwater line break is a limiting fault event, typically, a minimum DNBR limit is imposed such that fuel failure will be prevented. The total loss of all feedwater was analyzed following the TMI-2 transient. This event is considered to be beyond the original plant design basis. The acceptance criterion was that the core remained covered. The reactor was tripped on high RCS pressure and manual operator action was required by 20 minutes to initiate SFAS. The primary reactor trip function is provided by the high RCS pressure trip. The LOFW event was analyzed based on an initial core power level of 102% of 2772 MWt. The consequences of this event bound the power uprate. The power uprate was determined to have no impact on the capability of providing core cooling via primary system feed and bleed in the event of a total loss of all feedwater.

3.10.3.9 Excessive Heat Removal due to Feedwater System Malfunction and Excessive Load Increase

The excessive heat removal due to increased feedwater flow rate, decreased feedwater temperature, or increased steam flow (excessive load increase) events are bounded by the steam line break accident.

3.10.3.10 Anticipated Variations in the Reactivity of the Reactor Coolant

This original plant startup accident was performed to show that variations in reactivity during the cycle change slowly and are well within the capability of the control systems or by manual operator action to mitigate. No safety system actuation is required to mitigate this event.

The reactivity changes for fuel depletion and xenon buildup result in negative reactivity additions to the core. These additions will lead to power reductions if compensating actions are not taken. During normal operation, the control system will take action to increase the core reactivity by an equal amount to maintain a constant power level. The reactivity changes due to xenon burnup result in a positive reactivity addition to the core. This addition will lead to a power increase and a corresponding average coolant temperature increase if left uncompensated. During normal operation, the control system

will take action to decrease the core reactivity by an amount equal to the reactivity addition to maintain a constant power level and constant average temperature.

The plant and control system response to reactivity changes resulting from fuel depletion, burnable poison depletion, and changes in fission product poison concentration are not significantly affected by the initial core power level. As a result, the change in the magnitude of reactivity changes caused by fuel depletion, burnable poison depletion, and/or changes in fission product poison concentration will be negligible. Therefore, the current analyses of uncompensated reactivity changes support the power uprate.

3.10.3.11 Failure of Regulating Instrumentation

Failure of regulating instrumentation is the basis of many of the accidents analyzed. A malfunction of components in the ICS or the control rod drive system would be bounded by the startup accident. A control system failure in the SG secondary system would be bounded by the loss of feedwater accident.

3.10.3.12 Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes which Actuates Emergency Core Cooling and the Classical Large- and Small Break Loss of Coolant Accidents (LOCAs)

A spectrum of break sizes and break locations is postulated in the primary coolant piping. The LOCAs are considered limiting fault transients, events that are not expected to occur, but are postulated because of the potential for large releases of radiation. The acceptance criteria relate to ensuring adequate core cooling for the short and long term post-LOCA, limiting reactor building pressure and temperature, and limiting offsite dose consequence.

A spectrum of LOCAs has been reanalyzed for the DBNPS using an initial power level of 3025 MWt (102% of 2966 MWt). The results bound the power uprate. New mass and energy release rates have also been generated for the DBNPS based on an initial power level of 3025 MWt.

3.10.3.13 Secondary System Pipe Break

The steam line breaks are the most severe overcooling events and are considered limiting fault transients. The acceptance criteria relate to effective core cooling, offsite dose release, reactor coolant system integrity, and reactor building integrity. The NRC-approved methodology (BAW-10193P-A) for analyzing these events, relative to the core response, is to initiate the transient from nominal conditions. The power level uncertainty is accounted for by conservatively increasing the OTSG inventory. The MSLB accident was recently analyzed for the DBNPS based on an initial power level of 2772 MWt. However, the OTSG inventory was increased to greater than 102% and therefore bounds the power uprate. Revised mass and energy release rates for the MSLB accident have also been calculated. This analysis is based on a core power level of 102% of 2772 MWt and will also bound the power uprate.