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United States Nuclear Regulatory Commission
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Washington, DC 20555-0001

Subject: Davis-Besse Nuclear Power Station
License Amendment Application for Measurement Uncertainty Recapture Power Uprate (License Amendment Request No. 05-0007)

Ladies and Gentlemen:

Pursuant to 10 CFR 50.90, a license amendment is requested for the Davis-Besse Nuclear Power Station, Unit Number 1 (DBNPS). The proposed amendment would make the Operating License and Technical Specification changes necessary to allow an increase in the Rated Thermal Power from 2772 megawatts thermal (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 plant power calorimetric measurement.

This license amendment application has been prepared using the guidelines contained in Nuclear Regulatory Commission (NRC) Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications."

The proposed changes have been reviewed by the DBNPS Plant Operations Review Committee and Company Nuclear Review Board. Approval of the proposed amendment is requested by October 31, 2007. Once approved the amendment shall be implemented within 120 days.

Enclosure 1 is the Davis-Besse evaluation of the proposed amendment. Enclosure 2 is Attachment A to the Davis-Besse Nuclear Power Station Measurement Uncertainty Recapture Power Uprate Summary Report, prepared by AREVA NP Inc. Enclosure 2 was prepared in accordance with the guidelines contained in Attachment 1 to NRC RIS 2002-03. Enclosure 3 is AREVA NP Calculation 32-5012428-08, Davis-Besse Heat Balance Uncertainty, which is considered non-proprietary in its entirety.

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A list of regulatory commitments made in this submittal is included as Attachment A to this letter. If there are any questions or if additional information is required, please contact Mr. Henry L. Hegrat, FENOC Fleet Licensing Supervisor, at (330) 374-3114.

The statements contained in this submittal, including its associated enclosures and attachments, are true and correct to the best of my knowledge and belief. I am authorized by the FirstEnergy Nuclear Operating Company to make this submittal. I declare under penalty of perjury that the foregoing is true and correct.

Executed on April 12, 2007.

By: ______________________________
Mark B. Bezilla, Vice President-Nuclear

MKL

Attachment:

A. Commitment List

Enclosures:

1. Davis-Besse Nuclear Power Station Evaluation for License Amendment Request Number 05-0007

2. Attachment A to the Davis-Besse Nuclear Power Station Measurement Uncertainty Recapture Power Uprate Summary Report prepared by AREVA NP Inc.

3. AREVA NP Calculation 32-5012428-08, Davis-Besse Heat Balance Uncertainty April 2007.

cc: Regional Administrator, NRC Region III
NRC/NRR Project Manager
NRC Senior Resident Inspector
Executive Director, Ohio Emergency Management Agency,
State of Ohio (NRC Liaison)
Utility Radiological Safety Board

COMMITMENT LIST

The following list identifies those actions committed to by the Davis-Besse Nuclear Power Station, Unit Number 1, (DBNPS) in this document. Any other actions discussed in the submittal represent intended or planned actions by the DBNPS. They are described only for information and are not regulatory commitments. Please contact Mr. Henry L. Hegrat, FENOC Fleet Licensing Supervisor (330-374-3114), with any questions regarding this document or associated regulatory commitments.

COMMITMENT	DUE DATE
1. Changes to the DBNPS Updated Final Safety Analysis Report (UFSAR) Technical Requirements Manual (TRM) are being made in support of this LAR. These changes include the addition of required actions should the Ultrasonic Flow Meter instrumentation be inoperable or not used in the performance of the daily heat balance measurement required by Functional Unit 2, "High Flux," of TS Table 4.3-1, "Reactor Protection System Instrumentation Surveillance Requirements." These changes will be made subject to the provisions of 10 CFR 50.59. (See Section 2.0 of Enclosure 1.)	Prior to or concurrent with implementation of the proposed license amendment.
2. Implementation of the power uprate license amendment will include developing the necessary procedures and documents required for operation, maintenance, calibration, testing, and training at the uprated power level with the new LEFM system. (See Section I.1.D.1.1 of Enclosure 2.)	Prior to or concurrent with implementation of the proposed license amendment.
3. FENOC will complete any modifications [except feedwater heater level optimization, an enhancement currently planned for 15RFO, which is scheduled to commence in December 2007] identified in Item VII.2 of Enclosure 2 (including the training of operators), prior to implementation of the power uprate (See Section VII.3 of Enclosure 2.)	Prior to or concurrent with implementation of the proposed license amendment.

COMMITMENT	DUE DATE
4. FENOC will revise existing plant operating procedures related to temporary operation above "full steady-state licensed power levels" to reduce the magnitude of the allowed deviation from the licensed power level. The magnitude shall be reduced from the pre-power uprate value of 2 percent to 0.37%, the value corresponding to the uncertainty in power level credited by this proposed power uprate application. (See Section VII.4 of Enclosure 2.)	Prior to or concurrent with implementation of the proposed license amendment.

DAVIS-BESSE NUCLEAR POWER STATION
EVALUATION
FOR
LICENSE AMENDMENT REQUEST NUMBER 05-0007

Subject License Amendment Application for Measurement Uncertainty Recapture Power Uprate

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1.0 DESCRIPTION

The proposed amendment to the Davis-Besse Nuclear Power Station (DBNPS), Unit Number 1, Facility Operating License Number NPF-3 would increase the authorized RATED THERMAL POWER from 2772 megawatts thermal (MWt) to 2817 MWt (approximately 1.63%), based on the use of Caldon Inc. Leading Edge Flow Meter (LEFM) CheckPlus™ System ("Ultrasonic Flow Meter") instrumentation to improve the accuracy of the plant power calorimetric measurement. The LEFM CheckPlus™ System has been installed at the DBNPS.

FirstEnergy Nuclear Operating Company (FENOC) contracted AREVA NP Inc. to evaluate the impact of a 1.63% uprate to 2817 MWt for applicable systems, structures, components, and safety analyses, and to determine that such a power uprate is acceptable for DBNPS Unit Number 1. This evaluation is documented in AREVA Report No. 51-9004090-005, "FirstEnergy Nuclear Operating Company, Davis-Besse Nuclear Power Station, Measurement Uncertainty Recapture, Power Uprate Summary Report" (Reference 1).

This License Amendment Request (LAR) addresses the application of the Caldon LEFM CheckPlus™ System, the power uprate related changes and a change to the minimum usable volume of the Condensate Storage Tanks. The change to the minimum usable volume of the Condensate Storage Tanks is being made to reflect a calculation that was revised in support of the proposed power uprate.

2.0 PROPOSED CHANGE

The proposed changes to the Operating License (OL) and Technical Specifications (TS) pages are provided in Attachment A[1]. The retyped OL and TS pages are provided in Attachment B.

The proposed OL and TS changes are described in the following sections.

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).

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

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[1] The proposed changes to the Operating License and Technical Specifications have been prepared electronically. Deletions are shown with a strike-through and insertions are shown double-underlined. This presentation allows the reviewer to readily identify the information that has been deleted and added.

follows:

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

TS 2.2.1, Limiting Safety System Settings – Reactor Protection System Setpoints

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 change the Allowable Value listed in Table 2.2-1 for Functional Unit 2, High Flux, from ≤ 105.1% to ≤ 104.9% of RATED THERMAL POWER with four pumps operating. A footnote is provided to require the High Flux Allowable Value to be reduced to ≤ 103.3% of RATED THERMAL POWER when licensee controlled requirements are being implemented due to the Ultrasonic Flow Meter instrumentation being inoperable or not used in the performance of the daily heat balance.

TS 3/4.3.1, Reactor Protection System Instrumentation

It is proposed to add a reference to Note 10 to Functional Unit 2, High Flux, in Table 4.3-1. The proposed added reference is consistent with NRC guidance for treatment of Limiting Safety System Settings as established in DBNPS License Amendment No. 274 (Reference 2). Amendment No. 274 implemented Framatome Mark B-HTP Fuel design, and modified the Reactor Coolant Pressure-Temperature trip setpoint for the Reactor Protection System.

TS 3.7.1.3, Condensate Storage Tanks

It is proposed to revise TS 3.7.1.3 to reflect a new minimum volume by having it read as follows:

> The condensate storage tanks shall be OPERABLE with a minimum usable volume of 270,300 gallons of water.

It is also proposed to revise Surveillance Requirement 4.7.1.3.1 to clarify that the value to be confirmed is the usable water volume of the tanks.

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, an actual value of 100.37% of RATED THERMAL POWER may be used when the input for reactor thermal power measurement of feedwater mass flow and temperature is from the Ultrasonic Flow Meter.

In addition, TS 6.9.1.7 has been further modified by adding references to 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 (Reference 3), and Caldon Inc. Engineering Report-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM√™ or LEFM CheckPlus™ System," Revision 5, dated October, 2001 (Reference 4) to the section as NRC approved documents.

TS Bases

Changes to the associated TS Bases are being made in support of this LAR. These changes incorporate the uprate power level, the new High Flux Allowable Values, and the change of condensate storage tank volume from "contained" to "usable." The Bases changes also apply the NRC guidance for treatment of Limiting Safety System Settings to the High Flux Allowable Value, consistent with the approach established in DBNPS License Amendment No. 274 (Reference 2). The TS Bases changes will be processed under the DBNPS TS Bases Control Program and are provided, for information only, in Attachment C[2].

Technical Requirements Manual

Changes to the DBNPS Updated Final Safety Analysis Report (UFSAR) Technical Requirements Manual (TRM) are being made in support of this LAR. These changes include the addition of required actions should the Ultrasonic Flow Meter instrumentation be inoperable or not used in the performance of the daily heat balance measurement required by Functional Unit 2, "High Flux," of TS Table 4.3-1, "Reactor Protection System Instrumentation Surveillance Requirements." These changes will be made subject to the provisions of 10 CFR 50.59 and are provided, for information only, in Attachment D.

To meet format requirements the Index, Technical Specifications and Technical Specification Bases will be revised and repaginated as necessary to reflect the changes being proposed by this License Amendment Request.

3.0 BACKGROUND

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

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[2] The proposed changes to the Technical Specification Bases have been prepared electronically. Deletions are shown with a strike-through and insertions are shown double-underlined. This presentation allows the reviewer to readily identify the information that has been deleted and added.

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 that will be used, in turn, for determining the reactor thermal power above 50% of RATED THERMAL POWER. This system uses acoustic energy pulses to determine the main feedwater mass flow rate and temperature. The LEFM CheckPlus™ System consists of a measurement section (spool pieces in the two 18-inch main feedwater lines that each hold 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 to perform the plant calorimetric measurement calculation for reactor thermal power above 50% of RATED 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 of 2817 MWt.

The LEFM system was installed at DBNPS in 2002. Enclosure 2 provides information regarding additional DBNPS experience with the LEFM. FENOC plans to place the LEFM in service (for input to the calorimetric calculation) prior to the proposed uprate. The basis for placing the LEFM in service is to maintain 2772 MWt using a feedwater flow and temperature measuring instrument that is not subject to a loss of accuracy due to fouling of the feedwater flow venturis.

4.0 TECHNICAL ANALYSIS

The DBNPS is presently licensed for a core thermal power rating of 2772 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 by AREVA (Reference 1). Attachment A to Reference 1, provided in support of this application (see Enclosure 2), presents applicable results in a format consistent with NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications."

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 (Reference 3), as supplemented by Caldon Inc. Engineering Report-157P (Reference 4). 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 (Reference 5). Engineering Report-157P was approved in the NRC Safety Evaluation for the Waterford Steam Electric Station Unit 3, River Bend Station, and Grand Gulf Nuclear Station, dated December 20, 2001 (Reference 6).

Evaluation of Proposed Operating License and Technical Specification Changes

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 1. Based on this evaluation, this proposed change 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 1. Based on this evaluation, this proposed change will have no adverse effect on nuclear safety.

TS 2.2.1, Limiting Safety System Settings – Reactor Protection System Setpoints

Functional Unit 2 of Table 2.2-1 lists two Allowable Values, one for four reactor coolant pump operation and one for three pump operation. The setpoint for the three pump operating condition is validated for each new core reload. For both cycle 14 and cycle 15, a trip setpoint of 80.6% of 2817 MWt was validated for the three pump operation, and does not require a change for the proposed power uprate. Four pump operation, however, could not be supported using current analyses at the proposed higher power level without changing the trip setpoint.

The proposed change to the TS Table 2.2-1 Allowable Value for Functional Unit 2, High Flux, with four pumps operating and the Ultrasonic Flow Meter in use for the Heat Balance calculations, is in accordance with updated analyses in support of the proposed power uprate, as described and provided in Section VIII.2.C of Attachment A to Reference 1 (see Enclosure 2).

The Allowable Value for Functional Unit 2, High Flux with four pumps operating, is derived using Method 1 as described in Section 7.3 of ISA-RP67.04.02-2000, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation" (Reference 7). Uncertainties that are random, normally distributed, and independent are combined by the square-root-sum-of-squares (SRSS) method. Uncertainties that are not random, not normally distributed, or are dependent are combined algebraically. The total uncertainty is then subtracted from the Analytical Limit to establish the Allowable Value. This is further described in Section 7.4 of AREVA NP Document No. 43-10179PA-01, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses", BAW-10179P-A, Revision 1 (Reference 8).

Following this methodology, the trip setpoint will assure a reactor trip 95% of the time at a 95% confidence level.

The High Flux setpoint is a primary reactor trip function that is credited in the plant accident analyses. The trip setpoint is expressed as a percentage of the RATED THERMAL POWER (RTP). For current plant operation, the Allowable Value in the DBNPS Technical Specifications is 105.1% of RTP (105.1% of 2772 MWt). The safety analyses account for uncertainties in instrumentation and process measurement in the trip setpoint that is modeled. For current plant operation, the setpoint modeled in the accident analyses is 112% of RTP (112% of 2772 MWt). The derivation of the Allowable Value from what is modeled in the safety analyses has been reviewed and approved by the NRC and is described in Section 7.4 of Reference 8 and is summarized below:

$$SP_4'' = SP_4 + e_{hb} + e_m + e_{sp} \qquad \text{(Equation 1)}$$

where:

FP = full power

SP_4'' = accident analysis value of the high flux trip, %FP (Analytical Limit)

SP_4 = Technical Specification value of the high flux trip, %FP (Allowable Value)

e_{hb} = heat balance error, %FP

e_m = flux calibration error, %FP (steady-state and transient induced)

e_{sp} = bistable, %FP

The heat balance error for current plant operation is 2% FP. The steady-state error, which accounts for neutron measurement and long-term reactivity changes, is 2% of core power. The transient-induced error, which accounts for variations in the flux shape and detection that takes place during a given transient, is 2% of core power. In addition to the neutron measurement error, a bistable error is accounted for in the setpoint adjustment. This error is determined by statistical treatment of the individual error terms for reference accuracy, drift, and design range. The bistable uncertainty, 0.84% is based on the power range (125%) and not an absolute power level.

To define the Technical Specification Allowable Value, Equation 1 is rearranged to the following and the above values are substituted in:

$$SP_4 = SP_4'' - e_{hb} - e_m - e_{sp} \qquad \text{(Equation 2)}$$

$$= 112\% - 2\% - 4\% - 0.84\%$$

$$= 105.1\% \qquad \text{(rounded down for conservatism)}$$

For this submittal, the plant RATED THERMAL POWER level will be increased to 2817 MWt with a new heat balance error of 0.37% (Reference 9).

Reference 1 contains an evaluation of the plant systems, structures and components as well as the accident analyses for DBNPS for the proposed power uprate. The review of the accident analyses concluded that in order for the current analyses of record to remain valid for the uprated power level, the analysis value of the high flux setpoint (in terms of absolute power) must be preserved. As a result, the accident analysis trip setpoint value becomes:

$$SP_4'' = (112\%) * \frac{2772\ MWt}{2817\ MWt}$$

$$= 110.2\% \qquad \text{(rounded down for conservatism)}$$

By substitution into Equation 2 of the analysis trip setpoint and revised heat balance error, the new Technical Specification reactor trip setpoint will be:

$$SP_4 = SP_4'' - e_{hb} - e_m - e_{sp}$$

$$= 110.2\% - 0.37\% - 4\% - 0.84\%$$

$$= 104.9\% \qquad \text{(rounded down for conservatism)}$$

With four pumps operating and the Ultrasonic Flow Meter inoperable or *not* in use for the Heat Balance calculations, the footnote provided in TS Table 2.2-1 for Functional Unit 2 requires the High Flux Allowable Value to be reduced to ≤ 103.3% of RTP when licensee controlled requirements are being implemented. This value is calculated by accounting for the difference in heat balance error between the LEFM (0.37%) and the feedwater venturis (2%), using the same Equation 2 above:

$$SP_4 = SP_4'' - e_{hb} - e_m - e_{sp}$$

$$= 110.2\% - 2\% - 4\% - 0.84\%$$

$$= 103.3\% \qquad \text{(rounded down for conservatism)}$$

Based on the use of NRC-approved methodology noted above (Reference 8), the proposed changes will have no adverse effect on nuclear safety.

<u>TS 3/4.3.1, Reactor Protection System Instrumentation</u>

The proposed change to Functional Unit 2, High Flux, in Table 4.3-1 is the addition of a reference to Note 10. This note establishes requirements for the treatment of as-found and as-left setpoints during and after surveillance testing. TS Table 4.3-1 Note 10 ensures that unexpected as-found conditions are evaluated prior to returning the channel to service, and that as-left settings provide sufficient margin for uncertainties. This change is consistent with NRC guidance on instrument setpoint methodology previously established in DBNPS License Amendment No. 274 (Reference 2). Thus, this change will have no adverse effect on nuclear safety.

TS 3.7.1.3, Condensate Storage Tanks

The condensate storage tank volume calculation was revised in support of the proposed power uprate, as described in Section VI.1.A.3 of Attachment A to Reference 1. The revised calculation used current standards and updated assumptions, resulting in a more conservative value for minimum usable tank volume. The change in volume will ensure that sufficient water is available to maintain the Reactor Coolant System at HOT STANDBY conditions for 13 hours with steam discharge to atmosphere and to cooldown the Reactor Coolant System to less than 280°F under normal conditions (i.e., no loss of offsite power) which is in accordance with the current licensing basis. Accordingly, this change 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. This results in utilizing the increased accuracy associated with the LEFM to achieve an increased power level.

The revision of each of the methodologies 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 methodologies to apply to the proposed power uprate, conditioned upon the LEFM being used to measure feedwater mass flow and temperature as the input to the reactor thermal power measurement. 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 above evaluation of each individually proposed OL and TS change, it is concluded that the proposed changes will have no adverse effect on nuclear safety.

5.0 REGULATORY SAFETY ANALYSIS

The proposed amendment would increase the authorized RATED THERMAL POWER from 2772 megawatts thermal (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 plant power calorimetric measurement.

The portions of the Operating License (OL) and Technical Specifications (TS) affected by the proposed power uprate include:

- OL paragraph 2.C.(1), Maximum Power Level, and TS 1.3, Definitions – RATED THERMAL POWER, where the licensed power level is increased from 2772 MWt to 2817 MWt;
- TS 2.2.1, Limiting Safety System Settings – Reactor Protection System Setpoints, where the Reactor Protection System (RPS) High Flux setpoint Allowable Value for four pump operation is changed from ≤105.1% to ≤104.9% RTP in Table 2.2-1. Additionally, a reference to a note (#) is added to call for the reduction of the Allowable Value to ≤103.3% when licensee controlled requirements are being implemented due to the Ultrasonic Flow Meter being inoperable or not used for the Heat Balance calculation;
- TS 3/4.3.1, Reactor Protection System Instrumentation, with the addition of a reference to Note 10 to Functional Unit 2, High Flux, in Table 4.3-1;
- TS 3.7.1.3, Condensate Storage Tanks, where the water volume of the condensate storage tanks is specified as usable and increased; and
- TS 6.9.1.7, Core Operating Limits Report, with a revision of administrative controls associated with the Core Operating Limits Report.

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 Nuclear Regulatory Commission (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. LEFM CheckPlus™ System instrumentation, the allowance for power measurement uncertainties can be reduced.

5.1 No Significant Hazards Consideration

An evaluation has been performed to determine whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

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

   Response: No.

   Under contract to the FirstEnergy Nuclear Operating Company, AREVA NP Inc. performed evaluations of the Davis-Besse Nuclear Power Station (DBNPS) Nuclear Steam Supply System (NSSS) and balance of plant systems, components, and analyses that could be affected by the proposed change to the licensed power level. A power uncertainty calculation was performed and the effect of increasing core thermal power by 1.63 percent to 2817 MWt on the DBNPS design and licensing basis was evaluated. The evaluations determined that all structures, systems and components will continue to be capable of performing their design function at the proposed uprated power level of 2817 MWt. An evaluation of the accident analyses demonstrates that the applicable analysis acceptance criteria continue to be met with the proposed changes. No accident initiators are affected by the power uprate and no challenges to any plant safety barriers are created by any of the proposed changes.

   The proposed change to the licensed power level does not affect the release paths, the frequency of release, or the analyzed source term for any accidents previously evaluated in the DBNPS Updated Final Safety Analysis Report (UFSAR). Systems, structures, and components required to mitigate transients will continue to be capable of performing their design functions with the proposed changes, and thus were found acceptable. The reduced uncertainty in the power calorimetric measurement ensures that applicable accident analyses acceptance criteria will continue to be met with operation at the proposed power level of 2817 MWt. Analyses performed to assess the effects of mass and energy remain valid. The source term used to assess radiological consequences has been reviewed and determined to bound operation at the proposed power level.

   The proposed change to the RPS high flux setpoint Allowable Value does not alter the typical manner in which systems or components are operated, and, therefore, will not result in an increase in the probability of an accident. The proposed High Flux Trip Allowable Values preserve assumptions of current accident analyses at the higher thermal power allowed by the proposed amendment, irrespective of the source of Heat Balance calculation input data. This proposed change does not alter any assumption previously made in the radiological consequence evaluations, nor does it affect mitigation of the radiological consequences of an accident previously evaluated. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The addition of references to Note 10 to Functional Unit 2, High Flux, in Table 4.3-1 is administrative and does not impact the probability or consequences of an accident previously evaluated because its inclusion does not involve an accident initiator or impact any radiological analyses. This change is made to incorporate NRC guidance in a manner previously determined to be acceptable in DBNPS License Amendment No. 274.

The proposed change to the volume of the condensate storage tanks does not alter the typical manner in which the system or component is operated, and, therefore, will not result in a significant increase in the probability of an accident. The condensate storage tanks are not accident initiators. The proposed change preserves the assumptions previously made in the radiological consequence evaluations and the radiological consequences of accidents previously evaluated. Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to the Core Operating Limits Report (COLR) portion of the Administrative Controls Section of the TS are administrative and do not impact the probability or consequences of an accident previously evaluated because their inclusion do not involve accident initiators or impact any radiological analyses. These changes are made to include the NRC-approved documents pertaining to the Caldon Leading Edge Flow Meter.

In summary, none of the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated.

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

Response: No.

No new accident scenarios, failure mechanisms, or single failures are introduced as a result of any of the proposed changes. Use of the Caldon CheckPlus™ System has been analyzed, and failures of the system will have no adverse effect on any safety-related system or any systems, structures, and components required for transient mitigation. Systems, structures, and components previously required for the mitigation of a transient continue to be capable of fulfilling their intended design functions. The proposed changes have no significant adverse affect on any safety-related structures, systems or components and do not significantly change the performance or integrity of any safety-related system.

The proposed changes do not adversely affect any current system interfaces or create any new interfaces that could result in an accident or malfunction of a different kind than previously evaluated. Operating at a core power level of 2817 MWt does not create any new accident initiators or precursors. The reduced uncertainty in the power calorimetric measurement ensures that applicable accident analyses acceptance criteria continue to be met, to support operation at the proposed core power level of 2817 MWt. Credible malfunctions continue to be bounded by the current accident analyses of record or recent evaluations that demonstrate that applicable criteria will continue to be met with the proposed changes.

The proposed change to the RPS high flux setpoint Allowable Value does not introduce new accident scenarios, failure mechanisms or single failures. The change does not alter the manner in which plant systems or components are operated. The proposed High Flux Trip Allowable Values preserve assumptions of current accident analyses at the higher thermal power allowed by the proposed amendment, irrespective of the source of Heat Balance calculation input data. Therefore, this proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The addition of a reference to Note 10 to Functional Unit 2, High Flux, in Table 4.3-1 is administrative and will not create the possibility of a new or different kind of accident from any accident previously evaluated because its inclusion will not change the manner in which any equipment is operated.

The proposed change to the volume of the condensate storage tanks does not introduce new accident scenarios, failure mechanisms or single failures. Therefore, this proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes to the COLR portion of the Administrative Controls Section of the TS are administrative and will not create the possibility of a new or different kind of accident from any accident previously evaluated because their inclusion will not change the manner in which any equipment is operated.

In summary, none of the proposed changes will create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Response: No.

   The margins of safety associated with the power uprate are those pertaining to core thermal power. These include those associated with the fuel cladding, Reactor Coolant System pressure boundary, and containment barriers. An engineering evaluation of the proposed 1.63 percent increase in core thermal power was performed. The power uprate required revised NSSS design thermal and hydraulic parameters to be established to serve as the basis for all of the NSSS analyses and evaluations. This engineering review identified the design modifications necessary to accommodate the revised NSSS design conditions. Evaluations determined that the NSSS systems and components will continue to operate satisfactorily at the uprated power level with these modifications and the proposed changes. The NSSS accident analyses were evaluated at the uprated power level. In all cases, the evaluations demonstrate that the applicable analyses acceptance criteria will continue to be met with approval of the proposed changes. As such, the margins of safety will continue to be bounded by the analyses for all the changes being proposed.

   Therefore, none of the proposed changes will involve a significant reduction in a margin of safety.

Based on the above, it is concluded that the proposed license amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is acceptable.

5.2 <u>Applicable Regulatory Requirements/Criteria</u>

The summary evaluation provided in support of this application was performed in a format consistent with Attachment 1 to NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications."

The updated instrumentation setpoint calculations have been prepared in accordance with Instrument Society of America (ISA) Standard S67.04.01-2000, "Setpoints for Nuclear Safety-Related Instrumentation," and Recommended Practice ISA-RP67.04.02-2000, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation." The applicable portions of ANSI/ISA-67.04-01-2000 and ISA-RP67.04.02-2000 are equivalent to the corresponding NRC-endorsed sections of ISA-S67.04-1994. The calculations are consistent with Method 1 in ISA-RP67.04.02-2000 Section 7.3. Using this method, uncertainties that are random, normally distributed, and independent are combined by the square-root-sum-of-squares (SRSS) method. Uncertainties that are not random, not normally distributed, or are dependent are combined algebraically. These uncertainties account for reference accuracy, drift, design range and process measurement. The purpose of the Allowable Value is to identify a value that, if exceeded, may mean that the instrument has not performed within the assumptions of the setpoint calculation. Using Method 1, the Allowable Value is developed by subtracting the instrument uncertainties from the Analytical Limit. The difference between the Trip Setpoint and the Allowable Value reflects instrument uncertainties expected during normal operation, including drift and calibration. This is consistent with ISA-S67.04-1994 Section 4, Establishment of Setpoints. ISA-S67.04-1994 Part I has been endorsed by the Nuclear Regulatory Commission (NRC) through Regulatory Guide (RG) 1.105, Revision 3, "Setpoints for Safety-Related Instrumentation," subject to four listed exceptions and clarifications. The four listed exceptions and clarifications, taken verbatim from RG 1.105, and the DBNPS-specific response to each are as follows:

RG 1.105 Regulatory Position C.1

Section 4 of ISA-S67.04-1994 specifies the methods, but not the criteria, for combining uncertainties in determining a trip setpoint and its Allowable values. The 95/95 tolerance limit is an acceptable criterion for uncertainties. That is, there is a 95% probability that the constructed limits contain 95% of the population of interest for the surveillance interval selected.

The trip function design requirement, as specified in Section 7.4 of BAW-10179P-A, Rev. 06, is that the trip setpoint shall assure a reactor trip 95% of the time at a 95% confidence level. The 95/95 tolerance is used to establish acceptable uncertainty values for the instrument strings. At the DBNPS, this is assured by means of the calculation methods, instrument string calibration, and setpoint verification.

RG 1.105 Regulatory Position C.2

Sections 7 and 8 of Part 1 of ISA-S67.04-1994 reference several industry codes and standards. If a referenced standard has been incorporated separately into the NRC's regulations, licensees and applicants must comply with that standard as set forth in the regulation. If the referenced standard has been endorsed in a regulatory guide, the standard constitutes an NRC-accepted method of meeting a regulatory requirement as described in the regulatory guide. If a referenced standard has been neither incorporated into the NRC's regulations nor endorsed in a regulatory guide, licensees and applicants may consider and use the information in the referenced standard if appropriately justified, consistent with current regulatory practice.

Of the standards listed in Section 7 of Part 1 of ISA-S67.04-1994, Standard ANSI/ISA-S51.1, "Process Instrumentation Terminology," is not known to be incorporated separately into the NRC's regulations nor endorsed in a regulatory guide. However, since this standard addresses only terminology, and has negligible impact on the technical content of the submittal and its associated calculation, its use does not require further justification. None of the other standards listed in Section 7 and none of the standards listed in Section 8 of Part 1 of ISA S67.04-1994 are used as part of the basis for this License Amendment Request.

RG 1.105 Regulatory Position C.3

Section 4.3 of ISA-S67.04-1994 states that the limiting safety system setting (LSSS) may be maintained in Technical Specifications or appropriate plant procedures. However, 10 CFR 50.36 states that the Technical Specifications will include items in the categories of safety limits, Limiting Safety System Settings (LSSS), and limiting control settings. Thus, the LSSS may not be maintained in plant procedures. Rather, the LSSS must be specified as a Technical Specification-defined limit in order to satisfy the requirements of 10 CFR 50.36. The LSSS should be developed in accordance with the setpoint methodology set forth in the standard, with the LSSS listed in the Technical Specifications.

In accordance with Section 4.3 of Part 1 of ISA S67.04-1994, the purpose of a LSSS is to assure that protective action is initiated before the process conditions reach the analytical limit. In addition, the LSSS may be the Allowable Value, the trip setpoint, or both. Consistent with NRC guidance for treatment of Limiting Safety System Settings as established in DBNPS License Amendment No. 274 (Reference 2), the LSSS for the RPS high flux functional unit is the Limiting Trip Setpoint, and, in accordance with the TS Bases (Attachment C), the Limiting Trip Setpoint will be specified in the UFSAR Technical Requirements Manual.

RG 1.105 Regulatory Position C.4

ISA-S67.04-1994 provides a discussion on the purpose and application of an Allowable Value. The Allowable Value is the limiting value that the trip setpoint can have when tested periodically, beyond which the instrument channel is considered inoperable and corrective action must be taken in accordance with the Technical Specifications. The Allowable Value relationship to the setpoint methodology and testing requirements in the Technical Specifications must be documented.

The Allowable Value relationship to the setpoint methodology and testing requirements in the Technical Specifications is documented in the setpoint calculation. The setpoint calculation is maintained as part of plant records.

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the

Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

Section 10 CFR 51.22(c)(9) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (i) involve a significant hazards consideration, (ii) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) result in a significant increase in individual or cumulative occupational radiation exposure.

FirstEnergy Nuclear Operating Company (FENOC) has reviewed this license amendment request and has determined that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the proposed license amendment. The basis for this determination is provided in Section VII.5 of Attachment A of Reference 1.

7.0 PRECEDENT

License amendment applications based on the use of the Caldon Inc. LEFM CheckPlus™ System were approved for Donald C. Cook Nuclear Plant, Unit 2 (Reference 10) and Seabrook Station, Unit 1 (Reference 11). A review of these submittals indicates that each requests NRC approval to use the Caldon CheckPlus™ System for feedwater flow measurement and that the DBNPS submittal is comparable in this regard. The DBNPS submittal differs from those referenced in that it proposes multiple setpoints for the Reactor Protection System High Flux trips.

8.0 REFERENCES

1. AREVA Report No. 51-9004090-005, "FirstEnergy Nuclear Operating Company, Davis-Besse Nuclear Power Station, Measurement Uncertainty Recapture, Power Uprate Summary Report", dated April 2007.

2. NRC Letter to FirstEnergy Nuclear Operating Company, "Davis-Besse Nuclear Power Station, Unit 1 – Issuance of Amendment Regarding Framatome Mark B-HTP Fuel Design for Cycle 15 (TAC No. MC6888)," License Amendment No. 274, dated March 2, 2006.

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

4. ER-157P, Revision 5, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System," Caldon, Inc., dated October 2001.

5. Letter from Project Directorate IV-1, Division of Licensing Project Management, Office of Nuclear Reactor Regulation, to C.L. Terry, TU Electric, "Comanche Peak Steam Electric Station, Units 1 and 2 – Review of Caldon Engineering Topical Report ER 80P, 'Improving Thermal Power Accuracy and Plant Safety while Increasing Power Level Using the LEFM System' (TAC Nos. MA2298 and 2299)," dated March 8, 1999.

6. Letter from S. A. Richards, NRC, to M. A. Krupa, Entergy, "Waterford Steam Electric Station, Unit 3; River Bend Station; and Grand Gulf Nuclear Station – Review of Caldon, Inc. Engineering Report ER-157P (TAC Nos. MB2397, MB2399 and MB2468)," dated December 20, 2001.

7. Section 7 of ISA RP67.04.02-2000, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation."

8. AREVA NP Document No. 43-10179PA-01, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," BAW-10179P-A, Revision 1, approved by NRC letter dated January 22, 1996.

9. AREVA NP Calculation 32-5012428-08, "Davis-Besse Heat Balance Uncertainty Calculation," April 2007.

10. NRC Letter to Indiana Michigan Power Company Nuclear Generation Group, "Donald C. Cook Nuclear Plant, Unit 2 – Issuance of Amendment Regarding Measurement Uncertainty Power Uprate (TAC No. MB6751)," dated May 2, 2003 (ADAMS Accession No. ML030990094).

11. NRC Letter to FPL Energy Seabrook, "Seabrook Station Unit 1 – Issuance of Amendment Regarding Measurement Uncertainty Recapture Power Uprate (TAC No. MC8434)," dated May 22, 2006.

9.0 ATTACHMENTS

A. Proposed Mark-Up of Operating License and Technical Specification Pages

B. Proposed Retyped Operating License and Technical Specification Pages

C. Proposed Mark-Up of Technical Specification Bases Pages

D. Proposed Technical Requirements Manual Pages

Docket Number 50-346
License Number NPF-3
Serial Number 3198
Enclosure 1
Attachment A

PROPOSED MARK-UP
OF
OPERATING LICENSE AND TECHNICAL SPECIFICATION PAGES

The following is a list of the affected pages:

Operating License Page
4
Technical Specification Page
1-1
2-4 *
2-5
2-6
3/4 3-7
3/4 3-8 *
3/4 7-6
6-16 *
6-17

* **No change. Page included for context only.**

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 2817 ~~2772~~ 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.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. **[LATER]**~~275~~, are hereby incorporated in the license. FENOC shall operate the facility in accordance with the Technical Specifications.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the license supported by a favorable evaluation by the Commission:

(a) FENOC shall not operate the reactor in operational Modes 1 and 2 with less than three reactor coolant pumps in operation.

(b) Deleted per Amendment 6

(c) Deleted per Amendment 5

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).

2.2 LIMITING SAFETY SYSTEM SETTINGS

No change. Page included for context only.

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.

Table 2.2-1 Reactor Protection System Instrumentation Trip Setpoints

Functional unit	Allowable values
1. Manual reactor trip	Not applicable.
2. High flux	≤~~105.1~~104.9% 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 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 T_{out}°F – 7899.0) 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.

<u># < 103.3% of RATED THERMAL POWER, to be implemented in accordance with licensee controlled requirements for Ultrasonic Flow Meter instrumentation that is inoperable or not used in the performance of the daily heat balance.</u>

TABLE 4.3-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. High Flux	S	D(2), and Q(6,9,10)	N.A.	1, 2
3. RC High Temperature	S	R	SA(9)	1, 2
4. Flux - ΔFlux - Flow	S(4)	M(3) and Q(6,7,9)	N.A.	1, 2
5. RC Low Pressure	S	R	SA(9)	1, 2
6. RC High Pressure	S	R	SA(9)	1, 2
7. RC Pressure-Temperature	S	R(10)	SA(9,10)	1, 2
8. High Flux/Number of Reactor Coolant Pumps On	S	Q(6,9)	N.A.	1, 2
9. Containment High Pressure	S	E	SA(9)	1, 2
10. Intermediate Range, Neutron Flux and Rate	S	E(6)	N.A.(5)	1, 2 and *
11. Source Range, Neutron Flux and Rate	S	E(6)	N.A.(5)	2, 3, 4 and 5
12. Control Rod Drive Trip Breakers	N.A.	N.A.	Q(8,9) and S/U(1)(8)	1, 2 and *
13. Reactor Trip Module Logic	N.A.	N.A.	Q(9)	1, 2 and *
14. Shutdown Bypass High Pressure	S	R	SA(9)	2**, 3**, 4**, 5**
15. SCR Relays	N.A.	N.A.	R	1, 2 and *

Notation

No change. Page included for context only.

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - When THERMAL POWER [TP] is above 50% of RATED THERMAL POWER [RTP], and at a steady state, compare out-of-core measured AXIAL POWER IMBALANCE [API_O] to incore measured AXIAL POWER IMBALANCE [API_I] as follows:

$$\frac{RTP}{TP}[API_O - API_I] = \text{Offset Error}$$

Recalibrate if the absolute value of the Offset Error is $\geq$ 2.5%

(4) - AXIAL POWER IMBALANCE and loop flow indications only.

(5) - CHANNEL FUNCTIONAL TEST is not applicable. Verify at least one decade overlap prior to each reactor startup if not verified in previous 7 days.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Flow rate measurement sensors may be excluded from CHANNEL CALIBRATION. However, each flow measurement sensor shall be calibrated at least once each REFUELING INTERVAL.

(8) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of both the undervoltage and shunt trip devices of the Reactor Trip Breakers.

(9) - Performed on a STAGGERED TEST BASIS.

(10) - If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Limiting Trip Setpoint, or a value that is more conservative than the Limiting Trip Setpoint; otherwise, the channel shall be declared inoperable. The Limiting Trip Setpoint and the methodology used to determine the Limiting Trip Setpoint, the predefined as-found acceptance criteria band, and the as-left setpoint tolerance band are specified in a document incorporated by reference into the Updated Safety Analysis Report.

* - With any control rod drive trip breaker closed.

** - When Shutdown Bypass is actuated.

PLANT SYSTEMS

CONDENSATE STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tanks shall be OPERABLE with a minimum ~~contained~~ usable volume of ~~250,000~~270,300 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tanks inoperable, within 4 hours either:

a. Restore the condensate storage tanks to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Verify by administrative means the OPERABILITY of the service water system as a backup supply to the auxiliary feedwater system, verify once per 12 hours thereafter, and restore the condensate storage tanks to OPERABLE status within 7 days or be in HOT SHUTDOWN within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tanks shall be demonstrated OPERABLE at least once per 12 hours by verifying the ~~contained~~ usable water volume to be within its limits when the tanks are the supply source for the auxiliary feedwater pumps.

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 Deleted

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.

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, an actual value of 100.37% of RATED THERMAL POWER may be used when the input for reactor thermal power measurement of feedwater mass flow and temperature is from the Ultrasonic Flow Meter. The following NRC approved documents are applicable to the use of the Ultrasonic Flow Meter with a 0.37% measurement uncertainty:

> 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.
>
> Caldon Inc. Engineering Report-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM√™ or LEFM CheckPlus™ System," Revision 5, dated October, 2001.

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.

Docket Number 50-346
License Number NPF-3
Serial Number 3198
Enclosure 1
Attachment B

PROPOSED RETYPED
OPERATING LICENSE AND TECHNICAL SPECIFICATION PAGES

The following is a list of the affected pages:

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2-5
2-6
3/4 3-7
3/4 3-8 *
3/4 7-6
6-16 *
6-17

* **No change. Page included for context only.**

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 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.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. **[LATER]**, are hereby incorporated in the license. FENOC shall operate the facility in accordance with the Technical Specifications.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the license supported by a favorable evaluation by the Commission:

(a) FENOC shall not operate the reactor in operational Modes 1 and 2 with less than three reactor coolant pumps in operation.

(b) Deleted per Amendment 6

(c) Deleted per Amendment 5

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 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).

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% 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 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 T_{out}°F – 7899.0) 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.

\# ≤ 103.3% of RATED THERMAL POWER, to be implemented in accordance with licensee controlled requirements for Ultrasonic Flow Meter instrumentation that is inoperable or not used in the performance of the daily heat balance.

TABLE 4.3-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. High Flux	S	D(2), and Q(6,9,10)	N.A.	1, 2
3. RC High Temperature	S	R	SA(9)	1, 2
4. Flux - ΔFlux - Flow	S(4)	M(3) and Q(6,7,9)	N.A.	1, 2
5. RC Low Pressure	S	R	SA(9)	1, 2
6. RC High Pressure	S	R	SA(9)	1, 2
7. RC Pressure-Temperature	S	R(10)	SA(9,10)	1, 2
8. High Flux/Number of Reactor Coolant Pumps On	S	Q(6,9)	N.A.	1, 2
9. Containment High Pressure	S	E	SA(9)	1, 2
10. Intermediate Range, Neutron Flux and Rate	S	E(6)	N.A.(5)	1, 2 and *
11. Source Range, Neutron Flux and Rate	S	E(6)	N.A.(5)	2, 3, 4 and 5
12. Control Rod Drive Trip Breakers	N.A.	N.A.	Q(8,9) and S/U(1)(8)	1, 2 and *
13. Reactor Trip Module Logic	N.A.	N.A.	Q(9)	1, 2 and *
14. Shutdown Bypass High Pressure	S	R	SA(9)	2**, 3**, 4**, 5**
15. SCR Relays	N.A.	N.A.	R	1, 2 and *

No change. Page included for context only.

Notation

(1) - If not performed in previous 7 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - When THERMAL POWER [TP] is above 50% of RATED THERMAL POWER [RTP], and at a steady state, compare out-of-core measured AXIAL POWER IMBALANCE [API_O] to incore measured AXIAL POWER IMBALANCE [API_I] as follows:

$$\frac{RTP}{TP}[API_O - API_I] = \text{Offset Error}$$

Recalibrate if the absolute value of the Offset Error is $\geq$ 2.5%

(4) - AXIAL POWER IMBALANCE and loop flow indications only.

(5) - CHANNEL FUNCTIONAL TEST is not applicable. Verify at least one decade overlap prior to each reactor startup if not verified in previous 7 days.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Flow rate measurement sensors may be excluded from CHANNEL CALIBRATION. However, each flow measurement sensor shall be calibrated at least once each REFUELING INTERVAL.

(8) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of both the undervoltage and shunt trip devices of the Reactor Trip Breakers.

(9) - Performed on a STAGGERED TEST BASIS.

(10) - If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.

The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Limiting Trip Setpoint, or a value that is more conservative than the Limiting Trip Setpoint; otherwise, the channel shall be declared inoperable. The Limiting Trip Setpoint and the methodology used to determine the Limiting Trip Setpoint, the predefined as-found acceptance criteria band, and the as-left setpoint tolerance band are specified in a document incorporated by reference into the Updated Safety Analysis Report.

* - With any control rod drive trip breaker closed.

** - When Shutdown Bypass is actuated.

PLANT SYSTEMS

CONDENSATE STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tanks shall be OPERABLE with a minimum usable volume of 270,300 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tanks inoperable, within 4 hours either:

a. Restore the condensate storage tanks to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Verify by administrative means the OPERABILITY of the service water system as a backup supply to the auxiliary feedwater system, verify once per 12 hours thereafter, and restore the condensate storage tanks to OPERABLE status within 7 days or be in HOT SHUTDOWN within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tanks shall be demonstrated OPERABLE at least once per 12 hours by verifying the usable water volume to be within its limits when the tanks are the supply source for the auxiliary feedwater pumps.

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 Deleted

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.

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, an actual value of 100.37% of RATED THERMAL POWER may be used when the input for reactor thermal power measurement of feedwater mass flow and temperature is from the Ultrasonic Flow Meter. The following NRC approved documents are applicable to the use of the Ultrasonic Flow Meter with a 0.37% measurement uncertainty:

> 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.
>
> Caldon Inc. Engineering Report-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM√™ or LEFM CheckPlus™ System," Revision 5, dated October, 2001.

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.

Docket Number 50-346
License Number NPF-3
Serial Number 3198
Enclosure 1
Attachment C

PROPOSED MARK-UP
OF
TECHNICAL SPECIFICATION BASES PAGES

Note: The Bases pages are provided for information only.

The following is a list of the affected pages:

Page
B 2-1
B 2-4
B 2-8
B 3/4 3-1 *
B 3/4 3-2 *
B 3/4 3-3
B 3/4 3-4
B 3/4 3-5 *
B 3/4 7-2

* **No change. Page included for readability only.**

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, BWC and BHTP 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. The BWC correlation applies to Mark-B fuel with zircaloy or M5 spacer grids. The BHTP correlation applies to the Mark-B-HTP fuel. The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30 (B&W-2), 1.18 (BWC) and 1.132 (BHTP). 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.

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 shutdown 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.

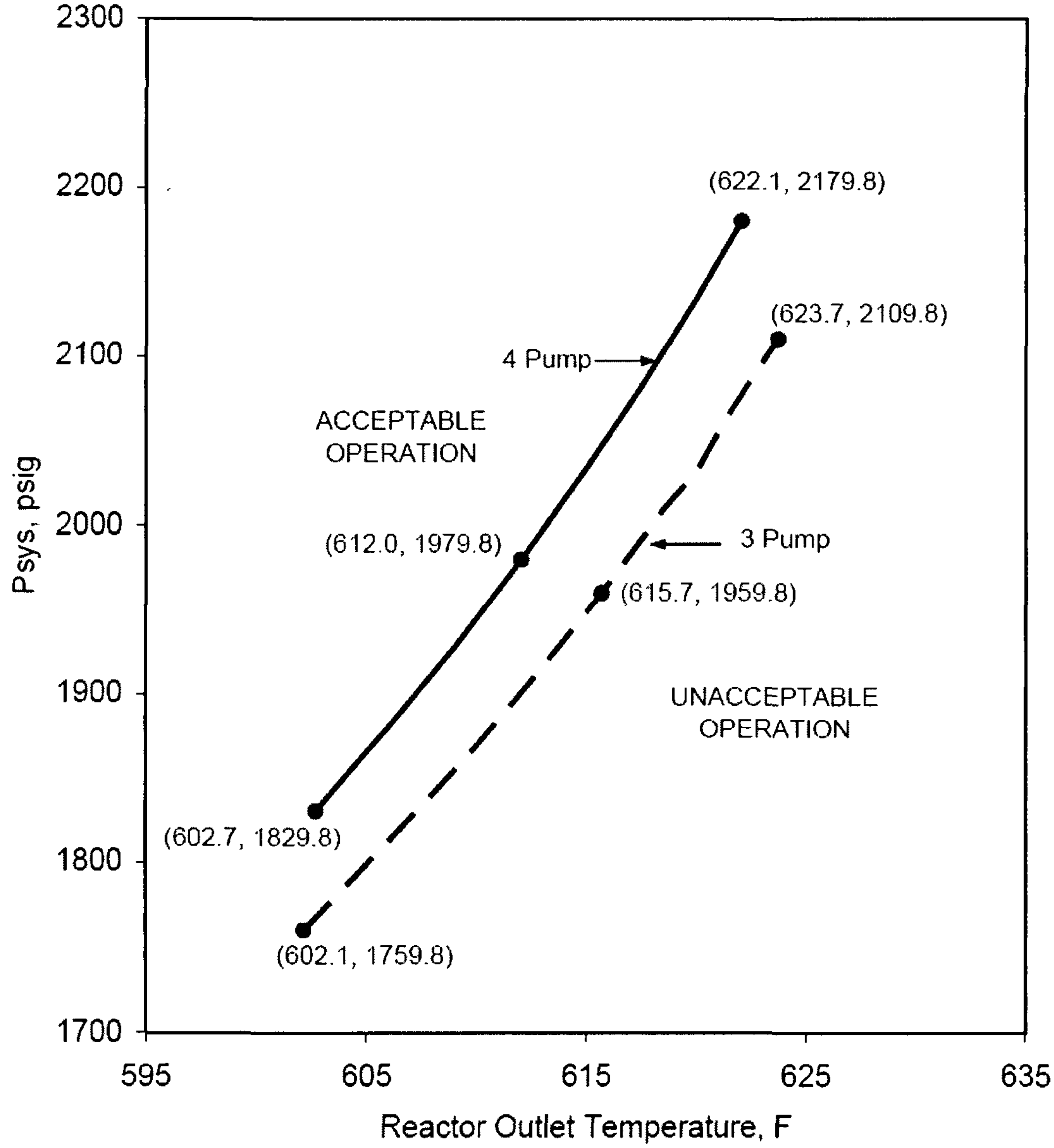
High Flux

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 except as noted below, a reactor trip is initiated when the reactor power level reaches the Allowable Value of ≤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.

Requirements regarding the Ultrasonic Flow Meter instrumentation are described in the Technical Requirements Manual (TRM). The TRM includes a required action to adjust the High Flux setpoints in the event the Ultrasonic Flow Meter instrumentation is inoperable or not used in the performance of the daily heat balance. The TRM does not specify the actual Allowable Value; it refers to TS 2.2.1. Under the TRM required action, the Allowable Value is ≤ 103.3% of RATED THERMAL POWER (2910 MWt) when operating with four reactor coolant pumps, thereby maintaining the same level of protection of the analytical limit. No changes to the TS high flux trip setpoints for reduced pump operation are required; the setpoint provided is conservative and bounds operation irrespective of the use of the Ultrasonic Flow Meter for heat balance calculation.

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



Pumps	Flow, gpm	2772 MWt Power	2817 MWt Power	Required Measured Flow to Ensure Compliance, gpm
4	380,000	112%	110.2%	389,500
3	283,860	90.1%	89.4%	290,957

No change. Page included for context only.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM AND SAFETY SYSTEM INSTRUMENTATION

OPERABILITY of the RPS, SFAS and SFRCS instrumentation systems ensures that 1) the associated action and/or trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for RPS, SFAS and SFRCS purposes from diverse parameters.

OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the required design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. The response time limits for these instrumentation systems are located in the Updated Safety Analysis Report and are used to demonstrate OPERABILITY in accordance with each system's response time surveillance requirements.

As indicated in RPS Table 4.3-1 for Functional Units 1 and 12, a CHANNEL FUNCTIONAL TEST is required to be performed for the Manual Reactor Trip function and the CRD Trip Breakers function once prior to each reactor startup, i.e., prior to Mode 2 entry, if not performed within the previous 7 days. These surveillance requirements ensure the OPERABILITY of these Functional Units prior to achieving criticality.

If the plant is in MODE 2 or if the plant is in MODE 3, 4, or 5 with the control rod drive trip breakers in the closed position and the control rod drive system capable of rod withdrawal, then Functional Unit 11.A of Table 3.3-1 applies. With THERMAL POWER level > 10^{-10} amps on both Intermediate Range channels, high voltage to the Source Range detectors may be de-energized. If the plant is in MODE 3, 4, or 5 with the control rod drive trip breakers not in the closed position or the control rod drive system not capable of withdrawal, Functional Unit 11.B of Table 3.3-1 applies. Applicability of Functional Units 11.A and 11.B is dependent upon the plant MODE and control rod drive system status and is not dependent on if the plant is in the process of a startup or a shutdown.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM AND SAFETY SYSTEM INSTRUMENTATION (Continued)

To comply with TS 3.3.1.1, all four CRD Trip Breakers and all four RPS Reactor Trip Modules (RTMs) must be OPERABLE, or entry into RPS Table 3.3-1, ACTION 7 and/or ACTION 8 is required:

> If an RTM is inoperable, depending on the reason, it may be appropriate to apply Action 7.a.1, 7.a.2, or both, as needed to ensure that a valid reactor trip signal will cause a reactor trip when needed, despite the inoperability of the RTM. The most conservative action would be to open the CRD Trip Breaker and physically remove the RTM from its cabinet, however, this may make the plant susceptible to an unnecessary reactor trip and transient due to a spurious signal.
>
> If a CRD Trip Breaker is inoperable, it is appropriate to apply Action 7.a.2.
>
> Action 7.b affords limited flexibility with respect to surveillance testing in the Reactor Trip System opposite the Reactor Trip System with the inoperable channel.

The Shutdown Bypass High Pressure trip provides the means for removing four automatic RPS trip parameters (i.e., RC Low Pressure, RC Pressure-Temperature, Flux-ΔFlux-Flow, High Flux/Number of Reactor Coolant Pumps On) from the RPS circuitry during a plant shutdown and heatup, or control rod drive testing and zero power physics testing, and inserts a lower RC high pressure trip in the circuit. If the CRD breakers are open and the CRD system is not capable of rod withdrawal, the trip function of the Shutdown Bypass is not considered actuated with the keyswitch initiated since it will not cause tripping of the CRD breakers. Therefore, under this condition the "Applicable MODES" of Functional Unit 14 of Table 3.3-1 and the "MODES in Which Surveillance Required" for Functional Unit 14 of Table 4.3-1 are not entered.

SFAS Table 3.3-3, ACTION 10, allows entry into this ACTION statement to be delayed for up to 8 hours when a functional unit is placed in an inoperable status solely for performance of a CHANNEL FUNCTIONAL TEST, provided at least two other corresponding functional units remain OPERABLE. The term "corresponding functional units" refers to the functional units (Total No. of Units column in Table 3.3-3) for the same trip setpoint in the other SFAS channels. For example, the corresponding functional units for a Containment Pressure - High unit are the other three Containment Pressure - High units. This 8-hour allowance provides a reasonable time to perform the required surveillance testing without having to enter the ACTION statement and implement the required ACTIONS.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM AND SAFETY SYSTEM INSTRUMENTATION (Continued)

SFRCS Table 3.3-11, ACTION 16, allows entry into this ACTION statement to be delayed for up to 8 hours when a channel is placed in an inoperable status solely for performance of a CHANNEL FUNCTIONAL TEST, provided the remaining actuation channel remains OPERABLE. This 8-hour allowance provides a reasonable time to perform the required surveillance testing without having to enter the ACTION statement and implement the required ACTIONS.

For the RPS, SFAS Table 3.3-4 Functional Unit Instrument Strings b, c, d, e, and f, and Interlock Channel a, and SFRCS Table 3.3-12 Functional Unit 2: [Note: Setpoint methodology for RPS Functional Units 2 and 7 are discussed on page B 3/4 3-4.]

Only the Allowable Value is specified for each Function. Nominal trip setpoints are specified in the setpoint analysis. The nominal trip setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable provided that operation and testing are consistent with the assumptions of the specific setpoint calculations. Each Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis to account for instrument uncertainties appropriate to the trip parameter. These uncertainties are defined in the specific setpoint analysis.

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Setpoints must be found within the specified Allowable Values. Any setpoint adjustment shall be consistent with the assumptions of the current specific setpoint analysis.

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis.

The frequency is justified by the assumption of an 18 or 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM AND SAFETY SYSTEM INSTRUMENTATION (Continued)

For RPS Functional Units 2 and 7, the Limiting Trip Setpoint is specified in the USAR Technical Requirements Manual. The Limiting Trip Setpoint is based on the calculated total loop uncertainty per the plant-specific methodology identified below. The Limiting Trip Setpoint may be established using Method 1 or Method 2 from Section 7 of ISA RP67.04.02-2000, *"Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation."* Additional information is contained in the Technical Requirements Manual.

The purpose of a Limiting Safety System Setting is to ensure that protective action is initiated before the process conditions reach the analytical limit, thereby limiting the consequences of a design-basis event to those predicted by the safety analyses. For RPS Functional Units 2 and 7, the Limiting Trip Setpoint is the Limiting Safety System Setting required by 10 CFR 50.36. This definition of the LSSS is consistent with the guidance issued to the industry through correspondence with NEI (Reference NRC-NEI Letter dated September 7, 2005). The definition of LSSS values continues to be discussed between the industry and the NRC, and further modifications to these TS Bases will be implemented as guidance is provided.

TS Table 4.3-1 Note 10 ensures that unexpected as-found conditions are evaluated prior to returning the channel to service, and that as-left settings provide sufficient margin for uncertainties. Specifically, for RPS Functional Units 2 and 7, the following additional requirements are added by TS Table 4.3-1 Note 10:

1) If the as-found Trip Setpoint is non-conservative with respect to the Allowable Value specified in the Technical Specification, the channel shall be declared inoperable and the associated Technical Specification ACTION statement shall be followed.

2) If the as-found Trip Setpoint is conservative with respect to the Allowable Value, and outside the pre-defined as-found acceptance criteria band, but the instrument is functioning as required and the channel can be reset to within the setting tolerance of the Limiting Trip Setpoint, or a value more conservative than the Limiting Trip Setpoint, then the channel may be considered OPERABLE. If it cannot be determined that the instrument channel is functioning as required, the channel shall be declared inoperable and the associated Technical Specification actions shall be followed.

3) If the as-found Trip Setpoint is outside the predefined as-found acceptance criteria band, the condition must be entered into the corrective action program for further evaluation.

3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM AND SAFETY SYSTEM INSTRUMENTATION (Continued)

The pre-defined as-found acceptance criteria band is determined by adding or subtracting (as applicable) credible uncertainties to the as-left value from the most recently completed surveillance test. Credible uncertainties include instrument uncertainties during normal operation including drift and measurement and test equipment uncertainties. In no case shall the pre-defined as-found acceptance criteria band overlap the Allowable Value. If one end of the pre-defined as-found acceptance criteria band is truncated due to its proximity to the Allowable Value, this does not affect the other end of the pre-defined as-found acceptance criteria band. If equipment is replaced, such that the previous as-left value is not applicable to the current configuration, the as-found acceptance criteria band is not applicable to calibration activities performed immediately following the equipment replacement.

Measurement of response time at the specified frequencies provides assurance that the RPS, SFAS, and SFRCS action function associated with each channel is completed within the time limit assumed in the safety analyses.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

The SFRCS RESPONSE TIME for the turbine stop valve closure is based on the combined response times of main steam line low pressure sensors, logic cabinet delay for main steam line low pressure signals and closure time of the turbine stop valves. This SFRCS RESPONSE TIME ensures that the auxiliary feedwater to the unaffected steam generator will not be isolated due to a SFRCS low pressure trip during a main steam line break accident.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM (Continued)

Following any modifications or repairs to the Auxiliary Feedwater System piping from the Condensate Storage Tank through auxiliary feed pumps to the steam generators that could affect the system's capability to deliver water to the steam generators, following extended cold shutdown, a flow path verification test shall be performed. This test may be conducted in MODES 4, 5 or 6 using auxiliary steam to drive the auxiliary feed pumps turbine to demonstrate that the flow path exists from the Condensate Storage Tank to the steam generators via auxiliary feed pumps.

Verification of the turbine plant cooling water valves (CW 196 and CW 197), the startup feedwater pump suction valves (FW 32 and FW 91), and the startup feedwater pump discharge valve (FW 106) in the closed position is required to address the concerns associated with potential pipe failures in the auxiliary feedwater pump rooms, that could occur during operation of the startup feedwater pump.

Exceptions to Specification 4.0.4 are provided for Surveillance Requirements 4.7.1.2.1.a.1, 4.7.1.2.1.c.2, and 4.7.1.2.1.g.1 for entry into MODE 3. Upon entering MODE 3, the requirements of Specification 4.0.3 are immediately applicable. Surveillance Requirements provided with exceptions to Specification 4.0.4 that have exceeded their required interval may be treated as missed surveillances in accordance with Specification 4.0.3. The time of discovery of the missed surveillance should be considered the time at which MODE 3 was entered. All other Surveillance Requirements, to which Specification 4.0.4 is applicable, must have been performed within the required surveillance interval or as otherwise specified prior to entry into MODE 3.

In all cases, reasonable assurance of Auxiliary Feedwater System OPERABILITY must exist prior to entry into MODE 3 including a reasonable expectation that any Surveillance Requirements provided with exceptions to Specification 4.0.4 which may not have been performed within the time interval required by Specification 4.0.2 will successfully demonstrate OPERABILITY when performed.

3/4.7.1.3 CONDENSATE STORAGE TANKS

The OPERABILITY of the Condensate Storage Tanks with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 13 hours with steam discharge to atmosphere and to cooldown the Reactor Coolant System to less than 280°F under normal conditions (i.e., no loss of offsite power). The water volume limit does not include water that is not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation does will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

Docket Number 50-346
License Number NPF-3
Serial Number 3198
Enclosure 1
Attachment D

PROPOSED
TECHNICAL REQUIREMENTS MANUAL PAGES

The following is a list of the affected pages:

Page
3/4 3-31
3/4 3-32

Note: The Technical Requirements Manual pages are provided for information only.

3/4.3 INSTRUMENTATION

3/4.3.4 ULTRASONIC FLOW METER INSTRUMENTATION

LCO 3.3.4.1 The Ultrasonic Flow Meter instrumentation shall be OPERABLE and used in the performance of the daily heat balance measurement required by Functional Unit 2, "High Flux," of Technical Specification Table 4.3-1, "Reactor Protection System Instrumentation Surveillance Requirements".

APPLICABILITY: **MODE 1, when greater than 50% RATED THERMAL POWER**

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Ultrasonic Flow Meter instrumentation inoperable or not used in the performance of the daily heat balance.	A.1 Reduce thermal power to $\leq$ 98.4% of RATED THERMAL POWER with four reactor coolant pumps operating or $\leq$73.8% of RATED THERMAL POWER with three reactor coolant pumps operating. <u>AND</u>	Prior to the completion of the next required daily heat balance measurement
	A.2 Reduce the setpoint for the Reactor Protection System High Flux (Technical Specification Table 2.2-1 Reactor Protection System Instrumentation Trip Setpoints – Functional Unit 2) in accordance with Technical Specification 2.2.1 for operation with the daily heat balance measurement not using the Ultrasonic Flow Meter.	Within 10 hours after the next required daily heat balance measurement

SURVEILLANCE REQUIREMENT

SURVEILLANCE	FREQUENCY
4.3.4.1 The Ultrasonic Flow Meter instrumentation shall be demonstrated OPERABLE by performance of a CHANNEL CHECK.	24 hours

3/4.3.4.1 Ultrasonic Flow Meter Instrumentation

Due to its higher accuracy, the use of OPERABLE Ultrasonic Flow Meter (Leading Edge Flow Meter (LEFM) CheckPlus™ System) instrumentation is preferred for the performance of daily heat balance calculations required by Technical Specification (TS) Surveillance Requirement (SR) 4.3.1.1.1 (Table 4.3-1, Functional Unit 2 - Reactor Protection System High Flux). The use of the LEFM instrumentation for the secondary-side feedwater flow and feedwater temperature inputs into the heat balance calculation provides an uncertainty of 0.37% above 50% of RATED THERMAL POWER (RTP). An uncertainty of 2% is assumed when non-LEFM instrumentation is used for the secondary-side feedwater flow and feedwater temperature inputs into the heat balance calculation. Below 50% of RTP, the heat balance is performed using primary-side instrumentation. Hence, this LCO is only applicable above 50% RTP. In addition, below 75% of RTP, the safety analyses have adequate margin to accommodate a 2% heat balance error either with or without the LEFM being used to perform the daily heat balance calculation.

If the LEFM is not available for use, the heat balance will be performed using inputs from less accurate installed instrumentation. Continued power operation is allowed; however, THERMAL POWER must be limited to ≤98.4% of RTP with four reactor coolant pumps operating, or ≤73.8% of RTP (75% of 2772 MWt) with three reactor coolant pumps operating. Given the larger heat balance uncertainty, these limits preserve the core power used in the UFSAR accident analysis and the initial conditions for DNB as required by the regulating group operating limits in the COLR.

Also, when operating with four reactor coolant pumps at the reduced power, the Reactor Protection System High Flux trip setpoint Allowable Value must be reduced from ≤104.9% to ≤103.3% within ten hours of completion of the heat balance calculation using the less accurate instrumentation, in accordance with the requirements of TS 2.2.1. This reduction ensures that when the increased uncertainty of the instrumentation is considered, the maximum analytical setpoint value of 110.2% will not be exceeded.

Historical comparison of the two feedwater flow measurement systems used for secondary-side heat balance calculations above 50% RTP, LEFM-based and feedwater venturi-based, indicates that the two methods do not diverge significantly during power operations over short periods. The long-term fouling of the venturis results in a more conservative feedwater flow input to the heat balance calculation. Nuclear Instrumentation trend analysis indicates that the NI to heat balance comparison will not drift significantly over a three-week period, and surveillance data indicates essentially no drift of the high flux setpoints. Accordingly, the accuracy and conservatism of the RPS high flux trip is acceptable in the ten hour period provided for setpoint reduction after completion of the non-LEFM-based heat balance calculation.

The LEFM includes a flow meter measurement section in each of the two main feedwater flow headers. Each measurement section consists of sixteen ultrasonic transducers. With any transducer inoperable, the Ultrasonic Flow Meter instrumentation system is considered inoperable and the required actions are to be applied.

The daily CHANNEL CHECK utilizes the on-line verification and self-diagnostic features of the LEFM to ensure the instrumentation is performing as designed.

Enclosure 2

Attachment A to the Davis-Besse Nuclear Power Station Measurement Uncertainty Recapture Power Uprate Summary Report prepared by AREVA NP Inc.

License Report for the MUR

I. Feedwater flow measurement technique and power measurement uncertainty

1. A detailed description of the plant-specific implementation of the feedwater flow measurement technique and the power increase gained as a result of implementing this technique. This description should include:

    A. Identification (by document title, number, and date) of the approved topical report on the feedwater flow measurement technique

    B. A reference to the NRC's approval of the proposed feedwater flow measurement technique

    C. A discussion of the plant-specific implementation of the guidelines in the topical report and the staff's letter/safety evaluation approving the topical report for the feedwater flow measurement technique

    D. The dispositions of the criteria that the NRC staff stated should be addressed (i.e., the criteria included in the staff's approval of the technique) when implementing the feedwater flow measurement technique

    E. A calculation of the total power measurement uncertainty at the plant, explicitly identifying all parameters and their individual contribution to the power uncertainty

    F. Information to specifically address the following aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric:

        i. maintaining calibration
        ii. controlling software and hardware configuration
        iii. performing corrective actions
        iv. reporting deficiencies to the manufacturer
        v. receiving and addressing manufacturer deficiency reports

    G. A proposed allowed outage time for the instrument, along with the technical basis for the time selected

    H. Proposed actions to reduce power level if the allowed outage time is exceeded, including a discussion of the technical basis for the proposed reduced power level

RESPONSE to I.

I. Feedwater Flow Measurement Technique and Power Measurement Uncertainty

I.1. Detailed Description of the Davis-Besse Implementation of the Caldon LEFM CheckPlus™ Instrumentation and the 1.63% Power Increase

The feedwater flow measurement system installed at Davis-Besse is an LEFM CheckPlus™ ultrasonic, multi-path, transit time flowmeter. This equipment also provides a highly accurate feedwater temperature that will be input to the heat balance. The design of this advanced flow measurement system is addressed in detail by the manufacturer, Caldon, Inc., in Topical Reports ER-80P, Revision 0 (Reference I-1), and ER-157P, Revision 5 (Reference I-2). The original Bailey Feedwater flow venturis (FESP2A and FESP2B) will continue to measure main feedwater flow as well.

The LEFM ultrasonic flow meter system consists of an electronic cabinet (C5757E) in the main control room and two measurement section/spool pieces (FESP1A &FESP1B) located in the Turbine Building El. 623 in each of the two 18 inch main feedwater flow headers (18"-EBD-12) that feed each steam generator. The measurement sections are located upstream of the existing Feedwater Flow Venturis (FESP2A) in the Turbine Building El. 623, room 514 and (FESP2B) in the Auxiliary Building El. 603, room 400. The 'A' LEFM is located approximately 10 feet downstream of a 90° vertical-to-horizontal elbow and 15 feet upstream of one of the existing venturis. The 'B' LEFM is located approximately 10 feet downstream of a 90° vertical-to-horizontal elbow and 11 feet upstream of a 90° horizontal-to-vertical elbow. The LEFM flow meters were calibrated at the Alden Research Laboratory, Inc. facility using the plant's current piping configuration and variations of the plant's configuration.

Each measurement section consists of sixteen (16) ultrasonic transducer housings, forming the pressure boundary. Each transducer may be removed at full-power conditions without disturbing the pressure boundary. The installation location of these flow elements conforms to the requirements in Topical Reports ER-80P and ER-157P.

The LEFM system measures the transit times of pulses of ultrasonic energy traveling along chordal acoustic paths through the flowing fluid. This technology provides significantly higher accuracy than the existing flow instruments, which use differential pressure measurements; and temperature instruments, which use conventional thermocouple or resistance thermometers. The sound will travel faster when the pulse traverses the pipe with the flow and will travel slower when the pulse traverses the pipe against the flow, due to the Doppler effect. The LEFM uses these transit times and time differences between pulses to determine the fluid velocity. The LEFM also measures the speed of sound in water and uses this to determine the feedwater temperature.

The system uses a digital system controlled by software to employ the ultrasonic transit time method to measure the velocities at precise locations with respect to the pipe centerline. The system numerically integrates the measured velocities. The system's software has been developed and 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 and feedwater temperature are displayed on the electronic cabinet (C5757E) and transmitted to the plant process computer for use in the calorimetric measurement. The feedwater mass flow rate and feedwater temperature are 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 and a change in the way instrument uncertainty is combined for other parameters (e.g., steam temperature) results in a total uncertainty of 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 instrumentation and RTDs.

The LEFM indications of feedwater mass flow and temperature may be directly substituted for the venturi-based flow and the resistance temperature detector (RTD) temperature inputs currently used in the plant calorimetric measurement calculation performed with the plant computer. The plant computer will then calculate enthalpy and thermal power. The venturi-based feedwater flow and RTD temperature will continue to be used for feedwater control and other functions that they currently fulfill.

The Caldon Panel (C5757E) has outputs for internally generated system trouble alarms, which will be wired into the main control room annunciator and Plant Process Computer (PPC). The Caldon Equipment is monitored via a new annunciator window ("MFW FLOW CALDON SYS {system} TRBL {trouble}") located on Alarm Panel 10, actuated from C5754C.

I.1.A. Caldon Topical Reports Applicable to the LEFM CheckPlus™ System

The referenced Topical Reports are:

i. ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, dated March 1997 (Reference I-1)

ii. ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System," Revision 5, dated October 2001 (Reference I-2)

I.1.B. NRC Approval of Caldon LEFM CheckPlus™ System Topical Reports

The NRC approved the subject Topical Reports referenced in Item (A) above on the following dates:

i. ER-80P, NRC SER dated March 8, 1999 (Reference I-3)

ii. ER-157P, NRC SER dated December 20, 2001 (Reference I-4)

I.1.C. Davis-Besse Implementation of Guidelines and NRC SER for the Caldon LEFM CheckPlus™ System

The LEFM CheckPlus™ system was installed in Davis-Besse in accordance with the requirements of Topical Reports ER-80P and ER-157P. This system will be used for continuous calorimetric power determination by serial link with the PPC and incorporates self-verification features to ensure that hydraulic profile and signal processing requirements are met within its design basis uncertainty analysis.

The Davis-Besse LEFM CheckPlus™ system was calibrated in a site-specific model test at Alden Research Laboratories, with traceability to National Standards. The LEFM CheckPlus™ system was installed and commissioned according to Caldon procedures, which include verification of ultrasonic signal quality and hydraulic velocity profiles as compared to those tested during site-specific model testing.

I.1.D. Disposition of NRC Criteria in the SER during Installation

In approving Caldon Topical Reports ER-80P and ER-157P, the NRC established four criteria to be addressed by each licensee. The four criteria and a discussion of how each will be satisfied for Davis-Besse follow:

I.1.D.1 Criterion 1

Discuss maintenance and calibration procedures that will be implemented with the incorporation of the LEFM, including processes and contingencies for unavailable LEFM instrumentation and the effect on thermal power measurements and plant operation.

I.1.D.1.1 Response to Criterion 1

Implementation of the power uprate license amendment will include developing the necessary procedures and documents required for operation, maintenance, calibration, testing, and training at the uprated power level with the new LEFM system. A preventative maintenance program has been developed for the LEFM that is to be performed every refueling outage. The preventative maintenance activity is described in maintenance plan 83956 (Reference I-11). The preventative maintenance program for the LEFM was developed using the vendor's maintenance and troubleshooting manual. The preventative maintenance activity performs the following check:

- General inspection of the terminal and cleanliness
- Power Supply inspection of magnitude and noise
- Central Processing Unit inspection
- Acoustic Processor Unit Checks of the 5 MHz clock and LED status
- Analog Input checks of the A/D converter
- Watchdog Timer checks that ensures the software is running
- Transducer Cable checks of continuity and meggarring the cables
- Wall thickness check of each Feedwater spool piece
- Calibration checks of each of the Feedwater pressure transmitters.

The preventative maintenance program and continuous monitoring of the LEFM ensures that the LEFM remains bounded by the analysis and assumptions set forth in the Topical Report ER-80P. The incorporation of, and continued adherence to, these requirements will assure that the LEFM system is properly maintained and calibrated. Note that the LEFM provides both feedwater flow and temperature inputs to the core thermal power calculation.

The LEFM operability requirements will be contained in the DBNPS UFSAR Technical Requirements Manual (TRM). A Limiting Condition for Operation (LCO) has been drafted for inclusion in the TRM stating that an operable LEFM shall be used in the performance of the daily calorimetric heat balance measurements whenever power is greater than 50% RTP. If the LEFM is not operable, the LCO will require that either the LEFM is restored to operable status prior to the next required daily heat balance measurement or power shall be reduced to ≤ 98.4% RTP (or ≤ 2772 MWt) and subsequent required heat balance measurements shall be taken using the feedwater flow venturis and the non-LEFM feedwater temperature instruments. This preserves the core power level used in the accident analyses. When operating with four reactor coolant pumps at this reduced power, the high flux trip allowable value setpoint must also be reduced to 103.3% RTP (or ≤ 2910 MWt) to maintain the same level of protection of the analytical limit with the heat balance based on the MFW flow venturis.

The small difference in power between current trip setpoint of 105.1 % of 2772 MWt (2913 MWt) and 103.3% of 2817 MWt (2910 MWt) is due to additional conservatism.

Trending of the Nuclear Instrumentation indicates that the NI-to-heat balance comparison will not drift significantly over at least a three week period of time, Reference I-15. Also, surveillance of the high flux trip setpoint confirms that there is essentially no drift in the setpoint, Reference I-12. Since the plant technical specifications require that a daily heat balance be performed, the plant staff will have until the next scheduled daily heat balance to restore the LEFM to operable status. At most, this would be for a period of 30 hours based on the current requirements for Functional Unit 2, "High Flux," of Technical Specification Table 4.3-1, which includes taking credit for the 25% surveillance interval extension. If the LEFM is not available for input to the next daily heat balance calculation, core power level and the high flux trip setpoint must be reduced. As discussed above, the instrumentation will not drift significantly such that a reasonable action time can be established. Using the Standard Technical Specifications for Babcock and Wilcox plants, Reference I-13, as a basis and modeling the action time requirement after the quadrant power tilt specification, LCO 3.2.4, a 10 hour action time will be applied. Consequently, if the LEFM is not restored by the next daily heat balance, core power will be reduced to 98.4% and the high flux trip allowable value will be reset to 103.3% within 10 hours after the completion of NI verification without the LEFM. The required actions and times for when the LEFM becomes unavailable will be included in the Davis-Besse UFSAR TRM. The determination of the appropriate flow measurement instrument for use in the determination of the heat balance will also be controlled in the technical specifications with reference provided for necessary actions and settings for each method of flow measurement calculation.

The TRM LCO will require that core power shall be maintained ≤ 98.4% RTP with four reactor coolant pumps operating or ≤ 73.8% RTP (or ≤ 75% of 2772 MWt) with three reactor coolant pumps operating until the LEFM is restored to OPERABLE status and the measurements have been performed using the LEFM. These requirements ensure that an operable LEFM shall be used whenever power is greater than the pre-uprate RTP level of 2772 MWt. With these requirements in place, the effect on plant operations is that power will be reduced and maintained to the pre-uprate level of 2772 MWt or lower, and that the venturis and RTDs will be used until the LEFM is returned to operable status. These requirements return the measurement techniques and maximum steady state power level to pre-uprate conditions.

I.1.D.2 Criterion 2

For plants that currently have LEFMs installed, provide an evaluation of the operational and maintenance history of the installed installation and confirmation that the installed instrumentation is representative of the LEFM system and bounds the analysis and assumptions set forth in Topical Report ER-80P.

I.1.D.2.1 Response to Criterion 2

The LEFM was installed in 2002. Active monitoring of the LEFM has been ongoing since February 2005. The Feedwater flow and Feedwater temperature data captured from the LEFM has been compared to Feedwater Flow venturis output and the Feedwater RTD output. The data comparison showed that the LEFM is consistent with the feedwater flow and temperature.

The LEFM functioned as designed until a transducer failed in June 2005. The transducer failure resulted in an alarm that would have caused the LEFM to be removed from service if it's data were being used for performing the heat balance calculation. Since the LEFM was not being used for that purpose, no repair was necessary at that time. Subsequent to the initial transducer failure, one additional transducer has failed. Replacement of these two transducers, as well as the other 30 transducers, was performed in June 2006. All 32 transducers were replaced in March 2007.

The LEFM system installed at DBNPS is planned to be placed in service (for input to the calorimetric calculation) prior to the proposed uprate. The basis for placing the LEFM in service is to maintain 2772 MWt using a Feedwater flow that is not subject to a loss of accuracy due to fouling of the Feedwater Flow venturis.

The Feedwater flow and temperature input to the calorimetric calculation will allow the operator to select between the LEFM flow/temperature and the Feedwater Flow venturis/feedwater RTD, using a software switch.

The preventative maintenance program developed for the LEFM is described in item I.1.D.1.1 above. The preventative maintenance program and continuous monitoring of the LEFM ensures that the LEFM remains bounded by the analysis and assumptions set forth in the Topical Report ER-80P.

I.1.D.3 Criterion 3

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

I.1.D.3.1 Response to Criterion 3

The LEFM uncertainty calculation is based on the ASME PTC 19.1 methodology (Reference I-8) and the Alden Research Laboratory Inc. calibration tests (Reference I-10). This ASME PTC 19.1 methodology was reviewed by the NRC as part of the Seabrook MUR application and Safety Evaluation Report (Reference I-9). The feedwater flow and temperature uncertainties were then combined with other plant measurement uncertainties (steam temperature, steam pressure, feedwater pressure) to calculate the overall heat balance uncertainty using accepted plant setpoint methodology.

This LEFM uncertainty calculation method is consistent with the current heat balance uncertainty calculation that uses the feedwater venturis and feedwater RTDs. The current calculation is based on a square-root-sum-squares calculation, which is also the basis for the ASME PTC 19.1 methodology.

I.1.D.4 Criterion 4

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

I.1.D.4.1 Response to Criterion 4

Criterion 4 does not apply to DBNPS. The calibration factor for the DBNPS spool pieces was established by tests of these spools at Alden Research Laboratory in October 2001 (Reference I-10). These included tests of a full-scale model of the DBNPS hydraulic geometry and tests in a straight pipe. An Alden data report for these tests (Reference I-10) and a Caldon engineering report evaluating the test data (Reference I-7) are on file in the D-B Records Management System. The calibration factor used for the LEFM CheckPlus™ at DBNPS is based on these reports. The uncertainty in the calibration factor for the spools is based on the Caldon engineering report (Reference I-7). The site-specific uncertainty analysis (Enclosure 3) documents these analyses. This document is maintained on file, as part of the technical basis for the DBNPS MUR uprate.

Final acceptance of the site-specific uncertainty analyses occurred after the completion of the commissioning process. The commissioning process verified bounding calibration test data (See Appendix F of Reference I-1). This step provided final positive confirmation that actual performance in the field meets the uncertainty bounds established for the instrumentation as described in Reference I-6. Final commissioning was completed in July 2004.

I.1.E. Total Power Measurement Uncertainty at Davis-Besse

The total power uncertainty using the LEFM CheckPlus™ at DBNPS is 0.37%. This calculation is provided in AREVA NP Calculation 32-5012428-08 (Reference I-5 and Enclosure 3). The parameters, their uncertainty, and relative contributions are shown in Table I-1. Note that this calculation was performed using preliminary uncertainty values for feedwater flow and feedwater temperature. The actual feedwater flow and temperature uncertainty values were shown to be less than those used in the calculation (Preliminary feedwater flow uncertainty was 0.30%, as-tested was 0.26%. Preliminary feedwater temperature uncertainty was 0.60°F, as-tested was 0.58°F).

Table I-1 below summarizes the core thermal power measurement uncertainty.

The current LEFM vendor, Cameron Measurement Systems (formerly Caldon Inc.), has performed an evaluation of the uncertainty involved in replacing LEFM CheckPlus™ transducers in the field (Reference I-14). This evaluation determined that consideration of transducer replacement uncertainty for the installed LEFM spool pieces at the Davis-Besse Nuclear Power Station would result in a change in overall mass flow uncertainty from 0.26% to 0.29%. Since, as noted above, the total power uncertainty calculation (Reference I-5) assumed a feedwater mass uncertainty of 0.30%, the total power uncertainty of 0.37% is unaffected. The LEFM vendor will revise the Davis-Besse analysis to reflect the new uncertainty term.

Table I-1 Process Parameter Inputs to Reactor Thermal Power

Symbol	Description	Units	Nominal Value	Absolute Systematic Uncertainty	Absolute Std. Dev. of the Mean	Absolute Sensitivity	Absolute Systematic Uncertainty Contribution	Absolute Random Uncertainty Contribution	Relative Systematic Uncertainty Contribution	Relative Random Uncertainty Contribution
WFW	Feedwater Flow Rate	lbm/hr	1.18E+07	3.55E+04	0	8.170E+02	2.105E+14	0.000E+00	71.27%	0.00%
TS	Steam Temperature	F	596	1.56	0.153	9.969E+06	6.047E+13	2.327E+12	20.47%	12.42%
PS	Steam Pressure	psia	930	1.42	1.52	-1.340E+06	9.056E+11	4.150E+12	0.31%	22.15%
TFW	Feedwater Temperature	F	455	0.6	0.24728	-1.323E+07	1.574E+13	1.070E+13	5.33%	57.09%
PFW	Feedwater Pressure	psia	1005	14.6	1.35	-6.157E+03	2.020E+09	6.908E+07	0.00%	0.00%
WMU	Makeup Flow Rate	lbm/hr	2.23E+04	1.11E+03	2.23E+03	7.396E+01	1.696E+09	2.713E+10	0.00%	0.14%
TMU	Makeup Temperature	F	100	5	2	2.201E+04	3.027E+09	1.937E+09	0.00%	0.01%
PMU	Makeup Pressure	psia	2250	50	50	5.812E+01	2.112E+06	8.446E+06	0.00%	0.00%
WLD	Letdown Flow Rate	lbm/hr	2.23E+04	1.11E+03	2.23E+03	5.555E+02	9.566E+10	1.531E+12	0.03%	8.17%
TLD	Letdown Temperature	F	557	5	2	2.791E+04	4.868E+09	3.116E+09	0.00%	0.02%
PLD	Letdown Pressure	psia	2250	50	50	-4.009E+01	1.004E+06	4.017E+06	0.00%	0.00%
QRCP	RCP Power	Btu/hr	6.75E+07	4.93E+06	0	1.000E+00	6.076E+12	0.000E+00	2.06%	0.00%
QLOSS	Ambient Heat Loss	Btu/hr	2.23E+06	2.50E+06	0	1.000E+00	1.563E+12	0.000E+00	0.53%	0.00%
Symbol	Description	Units	Nominal Value	Absolute Systematic Uncertainty	Absolute Random Uncertainty	Absolute Uncertainty Btu/hr	Relative Uncertainty %			
Qc	Core Thermal Power	Btu/hr	9.621E+09	3.438E+07	8.657E+06	3.545E+07	**0.36852537 or 0.37%**			

I.1.F. Calibration and Maintenance Procedures of All Instruments Affecting the Power Calorimetric

Information to specifically address the following aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric:

I.1.F.i. Maintaining Calibration

Calibration of the LEFM will be ensured by the preventative maintenance activity described in maintenance plan 83956. The preventative maintenance activity will verify the calibration of the 5 MHz clock in the Acoustic Processor unit and power supplies. In addition, it will verify the wall thickness of the Feedwater spool pieces are within tolerances. See Section I.1.D.1.1 for a complete list of maintenance activities.

The other instruments that contribute to the power calorimetric were unaffected by the addition of the LEFM and will be maintained according to existing calibration and maintenance procedures.

I.1.F.ii. Controlling Hardware and Software Configuration

Hardware configuration will be controlled in accordance with D-B station procedure NG-EN-00307, Configuration Management.

Software will be controlled in accordance with FENOC fleet procedure NOP-SS-1001, FENOC Administrative Program for Computer Related Activities.

LEFM software will be properly classified in accordance with NOP-SS-1001, FENOC Administrative Program for Computer Related Activities.

Davis-Besse Station Engineering Department personnel perform system monitoring in accordance with the system monitoring guidelines. This would include instruments that affect the power calorimetric such as the Caldon LEFM CheckPlus™ System. Equipment problems for plant systems, including the Caldon LEFM CheckPlus™ System equipment, fall under site work control processes. Conditions that are adverse to quality are documented under the corrective action program. Corrective action procedures, which ensure compliance with the requirements of 10 CFR 50, Appendix B, include instructions for notification of deficiencies and error reporting.

I.1.F.iii. Performing Corrective Actions

Corrective actions will be monitored and performed in accordance with FENOC fleet procedure NOP-WM-0001, Work Management Process.

I.1.F.iv. Reporting Deficiencies to the Manufacturer

Reporting deficiencies to the manufacturer will be performed in accordance with FENOC fleet procedure NOP-LP-2001, Corrective Action Program.

I.1.F.v. Receiving and Addressing Manufacturer Deficiency Reports

Manufacturer deficiency reports will be received and addressed in accordance with D-B station procedure EN-DP-01040, Vendor Technical Information Processing.

I.1.G. Allowed Outage Time and Technical Basis

The Nuclear Instrumentation (NI) indicated power is compared against heat balance power on a daily basis. In the event that the LEFM becomes unavailable, it must be restored prior to the next performance of the heat balance comparison or the plant power will be reduced to 98.4% ($\leq$ 2772 MWt) (see Item H below). The justification for the allowed outage time of the LEFM is that the NI's were compared to last known good heat balance calculation using the LEFM measurement under Davis-Besse's surveillance requirements and thus can continue to be relied upon for power measurement until the next daily comparison.

I.1.H. Actions for Exceeding Allowed Outage Time and Technical Basis

A requirement will be placed in the DBNPS UFSAR TRM to address LEFM unavailability. Should the LEFM system become unavailable, the current venturi-based feedwater flow and RTD feedwater temperature 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. The reactor operators will be provided with procedural guidance for those occasions when the LEFM CheckPlus™ is not available. As summarized below, for those instances a new section of the DBNPS UFSAR TRM will specify the appropriate actions to be taken when the LEFM CheckPlus™ system is unavailable. The DBNPS UFSAR TRM and other appropriate plant procedures will specify that if the LEFM CheckPlus™ becomes unavailable during the interval between daily performances of the heat balance comparison with the neutron detector (Technical Specification Table 4.3-1), plant operations may remain at a thermal power of 2817 MWt while continuing to use the power indications from the neutron detector power range channels.

However, in order to remain in compliance with the bases for operation at a Rated Thermal Power of 2817 MWt, the LEFM CheckPlus™ system must be returned to service prior to the next performance of the heat balance comparison required by Technical Specification Table 4.3-1. If the LEFM CheckPlus™ system has not been returned to service prior to the next performance of the heat balance comparison, a TRM action statement would require that the reactor power be reduced to, or maintained at, a power level of no greater than 98.4% RTP (≤ 2772 MWt) and will continue to be performed without the use of the LEFM until the LEFM is restored to OPERABLE status and a heat balance measurement has been performed using the LEFM. In addition, the high flux trip allowable value setpoint will be reduced to 103.3% RTP (or ≤ 2910 MWt) with four reactor coolant pumps operating. This ensures that when the difference in heat balance error between the LEFM and the main feedwater venturis is included, the analytical value of ≤ 3105 MWt is not exceeded.

No changes are required to the technical specification high flux trip setpoints for partial pump operation because the current setpoint was conservatively set and bounds operation with or without the NIs being verified by a heat balance using the LEFM. However, similar to 4-pump operation, core power must also be reduced to 73.8% RTP (or ≤ 75% of 2772 MWt) if the plant is operating with three reactor coolant pumps and the LEFM cannot be used for the heat balance. This preserves the maximum core power for initial condition DNB as required by the regulating group operating limits in the COLR.

Instead of recalibrating the Nuclear Instrumentation and RPS hardware to restore the pre-MUR plant configuration, i.e., 100% power equal to 2772 MWt, the Technical Specification RPS High Flux allowable value setpoint will be reduced to preserve the safety analysis analytical trip setpoint within 10 hours from the time of the NI-to-daily heat balance comparison without the LEFM. In essence, there will be up to a maximum time of 30 hours (24 hrs plus the 25% surveillance extension allow by plant technical specifications) to restore the LEFM to operable status. If the LEFM is not operable at that time, core power will be immediately reduced to 98.4% - with 4 RCPs in operation. In the following 10 hours, the high flux trip setpoint will be reduced to 103.3%, again with all 4 RCPs in operation. This will result in a maximum time of 40 hours from the time the LEFM first became inoperable until the revised trip setpoints are implemented.

The other instruments important to the calorimetric were not affected by the addition of the LEFM.

I. References:

I-1 ER-80P, Revision 0, Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System, Caldon, Inc., dated March 1997.

I-2 ER-157P, Revision 5, Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM✓™ or LEFM CheckPlus™ System, Caldon, Inc., dated October 2001.

I-3 Letter from Project Directorate IV-1, Division of Licensing Project Management, Office of Nuclear Reactor Regulation, to C.L. Terry, TU Electric, Comanche Peak Steam Electric Station, Units 1 and 2 – Review of Caldon Engineering Topical Report ER 80P, Improving Thermal Power Accuracy and Plant Safety while Increasing Power Level Using the LEFM System (TAC Nos. MA2298 and 2299), dated March 8, 1999.

I-4 Letter from S. A. Richards, NRC, to M. A. Krupa, Entergy, Waterford Steam Electric Station, Unit 3; River Bend Station; and Grand Gulf Nuclear Station – Review of Caldon, Inc. Engineering Report ER-157P (TAC Nos. MB2397, MB 2399 and MB2468), dated December 20, 2001.

I-5 AREVA NP Calculation 32-5012428-08, Davis-Besse Heat Balance Uncertainty, April 2007 (Enclosure 3).

I-6 ER-202 revision 2, Bounding Uncertainty Analysis for Thermal Power Determination at Davis-Besse Nuclear Power Station Using the LEFM CheckPlus™ System, Caldon, Inc., July 2004.

I-7 ER-227 Revision 1, Profile Factor Calculation and Accuracy Assessment for the Davis Besse LEFM CheckPlus™ Spool Pieces, Caldon, Inc., September 2003.

I-8 ASME PTC 19.1-1998, Test Uncertainty, Instruments and Apparatus, American Society of Mechanical Engineers, NY, NY, 1998.

I-9 NRC Letter, G. Edward Miller to Gene F. St. Pierre, Subject: Seabrook Station, Unit No. 1 – Issuance of Amendment RE: Measurement Uncertainty Recapture Power Uprate (TAC No. MC8434), May 2006.

I-10 Alden Research Laboratory Inc. Report No. ARL-310-01/C730, Calibration of Two 18" Leading Edge Flow Meters for Caldon Inc., PO Number 18350, October 2001.

I-11 Davis-Besse Maintenance Plan 83956.

I-12 D-B Calculation C-ICE-58.01-008, Rev 3, RPS RX Power Related Field Trip Setpoints.

I-13 Standard Technical Specifications for Babcock and Wilcox Plants, NUREG-1430, Vol. 1, Rev. 3.1, US Nuclear Regulatory Commission, December 2005.

I-14 Letter from Ed Madera, Cameron Measurement Systems, to Tim Laurer, Davis-Besse Nuclear Power Station, "Cameron Measurement Systems Response to Transducer Replacement Sensitivity," dated March 8, 2007.

I-15 Davis-Besse Notification No. 600352177, "APDX K NI to Heat Balance Eng Evaluation," December 6, 2006.

II. Accidents and transients for which the existing analyses of record bound plant operation at the proposed uprated power level

1. A matrix that includes information for each analysis in this category and addresses the transients and accidents included in the plant's updated final safety analysis report (UFSAR) (typically Chapter 14 or 15) and other analyses that licensees are required to perform to support licensing of their plants (i.e., radiological consequences, natural circulation cooldown, containment performance, anticipated transient without scram, station blackout, analyses to determine environmental qualification parameters, safe shutdown fire analysis, spent fuel pool cooling, flooding):

   A. Identify the transient or accident that is the subject of the analysis

   B. Confirm and explicitly state that

      i. the requested uprate in power level continues to be bounded by the existing analyses of record for the plant
      ii. the analyses of record either have been previously approved by the NRC or were conducted using methods or processes that were previously approved by the NRC
      iii. the analyses of record are not changed by the requested power uprate

   C. Confirm that bounding event determinations continue to be valid

   D. Provide a reference to the NRC's previous approvals discussed in Item B. above

RESPONSE to II.

II. Accidents and Transients Bounded by the Analyses of Record for the MUR

In order to support the Davis-Besse Measurement Uncertainty Recapture (MUR) Power Level Uprate, with respect to the accident analyses, a review of the Updated Final Safety Analysis Report (UFSAR) Chapters 6 and 15 and other related sub-sections was performed. Evaluations were performed on other analyses (e.g., HELB, flooding, Seismic) and determined there was no impact from the MUR. The purpose of the review was to confirm that the analysis results, as currently presented in the UFSAR, were performed conservatively and bounded the proposed power uprate. All of the analyses that are included in the UFSAR have been performed using NRC-approved tools and methods. Therefore, if the event is found to be bounding for the requested power uprate, no new analysis was required. If any event was determined to be not bounded by the current UFSAR analyses, then a new analysis was performed. Any new analyses were performed using methods that were previously approved by the NRC, except that the RELAP5/MOD2-B&W computer code was used. The RELAP5/MOD2-B&W code was separately reviewed and approved by the NRC as an appropriate substitute for the codes described in the UFSAR. It was confirmed in this evaluation that the existing analyses of record remain valid and bounding for the MUR power uprate.

Considered with the review of the Davis-Besse accident analyses, the high flux analytical limit of 3105 MWt must be preserved. Consequently, the reactor trip setpoint value will be adjusted to reference the new Reactor Thermal Power value of 2817 MWt. This will require a change from the current value of 112%RTP (2772 MWt) to 110.2% RTP (2817 MWt). This change maintains the reactor trip at the same net power level (3105 MWt) modeled in the safety analysis. After accounting for Measurement Uncertainty, instrumentation and process errors, with the reduced heat balance uncertainty using the LEFM CheckPlus™ system, the Technical Specification RPS High Flux trip allowable value will be reduced from 105.1% of 2772 MWt (or ≤ 2913.4 MWt) to 104.9% of 2817 MWt (or ≤ 2955 MWt) with four reactor coolant pumps operating. These limits are applicable when power range nuclear instrumentation is verified consistent with the heat balance results calculated using the LEFM system for feedwater flow measurement.

The development of the Allowable Value for the High Flux setpoint is derived using Method 1 as described in Section 7.3 of ISA RP67.04.02-2000, "Methodologies of the Determination of Setpoints for Nuclear Safety-Related Instrumentation." Uncertainties that are random, normally distributed, and independent are combined by the square-root-sum-of-squares (SSRS) method. Uncertainties that are not random, not normally distributed, or are dependent are combined algebraically. The total uncertainty is then subtracted from the Analytical Limit to establish the Allowable Value. This is further described in Section 7.4 of "Safety Criteria and Methodology for Acceptable Cycle Reload Analysis," AREVA NP Topical Report BAW-10179P-AB. Following this methodology, the trip setpoint will assure a reactor trip 95% of the time at a 95% confidence level.

If the LEFM system cannot be used for the feedwater input to the heat balance, a heat balance using the installed venturis will be performed. Whenever power range nuclear instrumentation verification to a heat balance using the venturis is required, power will be reduced to 98.4% RTP (or ≤ 2772 MWt). This preserves the core power level used in the accident analyses. When operating with four reactor coolant pumps at this reduced power, the high flux trip allowable value setpoint will be reduced to 103.3% RTP (or ≤ 2910 MWt) to maintain protection of the analytical limit. The small difference in power between 105.1 % of 2772 MWt (2913 MWt) and 103.3% of 2817 MWt (2910 MWt) is due to additional conservatism.

No changes are required to the technical specification high flux trip setpoints for partial pump operation because the current setpoint was conservatively set and bounds operation with or without the NIs being verified by a heat balance using the LEFM.

Table II-1 provides a brief overview of the accident analyses that are contained in the Davis-Besse UFSAR and whether the analyses remain valid for the MUR power uprate. A discussion of each of the UFSAR events that remain bounding for the MUR is presented in subsection 2 Discussion of Events below.

Table II-1 provides references to the NRC's previous approvals for each accident analysis. Where indicated, subsequent to the NRC's previous approval, changes to the accident analyses were made under the provisions of 10CFR50.59.

II.1. MATRIX

Table II-1 - Davis-Besse UFSAR Accident Analyses

UFSAR Section(s)	Event	Initial Core Power Used in UFSAR Analysis (% of 2772 MWt)	Bounded by Current UFSAR Analysis	Supported / Bounded by Other Analyses	Discussion
15.1.2	Anticipated Transient Without Scram	100 %	NA	NA	The ATWS transients are considered beyond the original design basis of the B&W-designed plants. In order to comply with 10CFR 50.62, a diverse scram system (DSS) and an automated signal to trip the turbine and start AFW were designed and installed on that basis. There is significant margin to accommodate any small change in the peak pressure prediction when crediting both the DSS and the ATWS Mitigation System Actuation Circuitry (AMSAC) functions provided by the SFRCS system.
15.2.1	Uncontrolled Control Rod Assembly Group Withdrawal from a Subcritical Condition (Startup Accident)	10^{-7} %	X		The technical specification high flux reactor trip setpoint will be reduced from 112% (2772 MWt) to 110.2% (of 2817 MWt) to ensure that the reactor is tripped at the same net power level. NRC Approval in Reference II-4.
15.2.2	Uncontrolled Control Rod Assembly Group Withdrawal at Power	100 %	X		The technical specification high flux reactor trip setpoint will be reduced from 112% (2772 MWt) to 110.2% (of 2817 MWt) to ensure that the reactor is tripped at the same net power level. The heat balance uncertainty is accounted for in the reactor trip setpoint to ensure the maximum amount of energy is added to the coolant. NRC Approval in Reference II-5.
15.2.3	Control Rod Assembly Misalignment (Stuck-out, Stuck-in, Dropped Control Rod Assembly)	100 %	X	X	An analysis was performed for the MUR power uprate. NRC approved methods and tools were used in this reanalysis. It was concluded that the current UFSAR analysis remains bounding, NRC Approval in Reference II-5 and supplemented by additional evaluations under the provisions of 10CFR50.59.
15.2.4	Makeup and Purification system Malfunction	100 %		X	This transient is bounded by the rod withdrawal at power event such that the only issue is to verify shutdown margin. The shutdown margin is independent of rated core power.
15.2.5	Loss of Forced Reactor Coolant Flow (Partial, Complete, or single Reactor Coolant Pump Locked Rotor)	102 %	X		The DNBR response is verified for each new fuel cycle. Cycle 14 and Cycle 15 DNBR analyses support the MUR power uprate. The UFSAR analyses were approved by the NRC in Reference II-5

Table II-1 - Davis-Besse UFSAR Accident Analyses

UFSAR Section(s)	Event	Initial Core Power Used in UFSAR Analysis (% of 2772 MWt)	Bounded by Current UFSAR Analysis	Supported / Bounded by Other Analyses	Discussion
15.2.6	Startup of an Inactive Reactor Coolant Loop (Pump Startup Accident)	60 %	X		Current UFSAR analysis remains bounding. The power uncertainty bounds the MUR power uprate. NRC Approval in Reference II-5.
15.2.7	Loss of External Load and/or Turbine Trip	100 %		X	The UFSAR states that the transient would be bounded by the loss of normal feedwater accident. The loss of normal feedwater accident was analyzed at 102% of 2772 MWt and is bounding for the MUR power uprate.
15.2.8	Loss of Normal Feedwater	102 %	X		The analysis in the UFSAR is bounding for the MUR power uprate. NRC Approval in Reference II-5 and supplemented by additional evaluations under the provisions of 10CFR50.59.
15.2.9	Loss of AC Power to the Station Auxiliaries (Station Blackout)	100 %		X	This event will transition from forced circulation to natural circulation but with a slightly higher power level. A higher core decay heat level will promote more natural circulation as long as sufficient AFW is available. The minimum AFW requirement is set by the Loss of Feedwater (LOFW) accident (UFSAR Section 15.2.8) based on 102% of 2772 MWt. The DNBR response is bounded by the loss of coolant flow accidents (UFSAR Section 15.2.5). NRC Approval in Reference II-5.
15.2.10	Excessive Heat Removal Due to Feedwater System Malfunction	102 %	X		The analysis in the UFSAR is bounding for the MUR power uprate. NRC Approval in Reference II-5.
15.2.11	Excessive Load Increase	NA		X	This section of the UFSAR states that the event is bounded by MSLB analysis in UFSAR Section 15.4.4. NRC Approval in Reference II-5.
15.2.12	Anticipated Variations in the Reactivity of the Reactor	100 %		X	This normal operational transient is retained for historical purposes to demonstrate that RCS response to normal variations in reactivity during the cycle will be slow and that the variations are within the capability of the operator or the Integrated Control System (ICS) to control. The reactivity changes are much smaller and therefore bounded by the reactivity transients presented in UFSAR Sections 15.2.1 through 15.2.4. NRC Approval in Reference II-5.

Table II-1 - Davis-Besse UFSAR Accident Analyses

UFSAR Section(s)	Event	Initial Core Power Used in UFSAR Analysis (% of 2772 MWt)	Bounded by Current UFSAR Analysis	Supported / Bounded by Other Analyses	Discussion
15.2.13	Failure of Regulating Instrumentation	NA		X	This section of the UFSAR states that the event is bounded by other events described in Chapter 15 of the UFSAR. This failure is independent of initial core power level. Therefore, these events remain bounding for the MUR power uprate. NRC Approval in Reference II-5.
15.2.14	External Causes	NA	X		This section of the UFSAR states that the event is addressed in Chapters 3 and 6 of the UFSAR in that these conditions must be considered. Earthquakes and tornadoes are independent of the core power level as precursors, however they are considered in the analyses. Therefore, this event remains bounding for the MUR power uprate. NRC Approval in Reference II-5.
15.3.1	Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuates Emergency Core Cooling	109.1%	X		The leak calculation is independent of core power as it is simply an estimate of the mismatch between the leak for a given area and the available makeup flow rate. The spectrum of LOCAs was analyzed for Davis-Besse using an initial power level of 109.1% of 2772 MWt. The analyses were performed with NRC approved methods and models. These analyses bound the MUR power uprate. NRC Approval in Reference II-5 and supplemented by additional evaluations under the provisions of 10CFR50.59.
15.3.2	Minor Secondary System Pipe Breaks	NA		X	This section of the UFSAR states that the discussion has been moved to the MSLB analysis in UFSAR Section 15.4.4. NRC Approval in Reference II-5.
15.3.3	Inadvertent Loading of a Fuel Assembly Into an Improper Position	NA	X		These considerations are independent of the rated core power level. Therefore the existing analyses remain valid for the MUR power uprate. The NRC Approval of the original analyses is in Reference II-5.
15.4.1	Waste Gas Decay Tank Rupture	NA	X		This event is not power level dependent. Therefore the existing analyses remain valid for the MUR power uprate. NRC Approval in Reference II-5 and supplemented by additional evaluations under the provisions of 10CFR50.59.

Table II-1 - Davis-Besse UFSAR Accident Analyses

UFSAR Section(s)	Event	Initial Core Power Used in UFSAR Analysis (% of 2772 MWt)	Bounded by Current UFSAR Analysis	Supported / Bounded by Other Analyses	Discussion
15.4.2	Steam Generator Tube Rupture	102 %	X		The system response for the SGTR calculation is independent of power level based on the analytical method used. The dose consequences were evaluated at 102% of 2772 MWt, using RCS iodine activity of ~5 μci/gm as described in the UFSAR Appendix 15A, which is higher than 1 μci/gm allowed by the plant Technical Specifications. These analyses bound the MUR power uprate. NRC Approval in Reference II-5 and supplemented by additional evaluations under the provisions of 10CFR50.59.
15.4.3	CRA Ejection Accident	102 %	X		The event consequences were evaluated at 102% of 2772 MWt. These analyses bound the MUR power uprate. NRC Approval in Reference II-5 and supplemented by additional evaluations under the provisions of 10CFR50.59.
15.4.4	Steam Line Break	102 %	X		The analyses were based on 102% of 2772 MWt. Therefore, relative to the core response, the UFSAR analysis remains bounding. The dose consequences were evaluated at 102% of 2772 MWt, using RCS iodine activity of ~5 μci/gm as described in the UFSAR Appendix 15A, which is higher than 1 μci/gm allowed by the plant Technical Specifications. These analyses bound the MUR power uprate. NRC Approval in Reference II-5 and supplemented by additional evaluations under the provisions of 10CFR50.59.
15.4.5	Break in Instrument Lines or Lines from Primary system that Penetrate Containment	102 %	X		The dose consequences were evaluated at 102% of 2772 MWt, using RCS iodine activity of ~5 μci/gm as described in the UFSAR Appendix 15A, which is higher than 1 μci/gm allowed by the plant Technical Specifications. These analyses bound the MUR power uprate. NRC Approval in Reference II-5 and supplemented by additional evaluations under the provisions of 10CFR50.59.
15.4.6	Major Rupture of Pipes Containing Reactor Coolant up to and Including Double-Ended Rupture of the Largest Pipe in the Reactor Coolant system (Loss-of-Coolant Accident)	102 %	X		The dose consequences were evaluated at 102% of 2772 MWt. These analyses bound the MUR power uprate. NRC Approval in Reference II-5 and supplemented by additional evaluations under the provisions of 10CFR50.59.

Table II-1 - Davis-Besse UFSAR Accident Analyses

UFSAR Section(s)	Event	Initial Core Power Used in UFSAR Analysis (% of 2772 MWt)	Bounded by Current UFSAR Analysis	Supported / Bounded by Other Analyses	Discussion
15.4.7	Fuel Handling Accident	102 %	X		The dose consequences were evaluated at 102% of 2772 MWt. These analyses bound the MUR power uprate. NRC Approval in Reference II-5 and supplemented by additional evaluations under the provisions of 10CFR50.59.
15.4.8	Effects of Toxic Material Release on the Control Room	NA	X		Toxic materials are not stored in volumes which would affect control room habitability. Further, this accident is independent of core power. NRC Approval in Reference II-5.
6.2.1.1.2	LOCA Mass and Energy Release	109.1 %	X		Mass and energy release data was generated based on a power level of 109.1 % of 2772 MWt. NRC Approved methods and tools were used in this analysis. NRC Approval in Reference II-5 and supplemented by additional evaluations under the provisions of 10CFR50.59.
6.2.1.3.2 15.4.4.2.3.3	MSLB Mass and Energy Release	102 %	X		Mass and energy release data was generated based on a power level of 102 % of 2772 MWt. NRC Approved methods and tools were used in this analysis. NRC Approval in Reference II-5 and supplemented by additional evaluations under the provisions of 10CFR50.59.
6.3.3.1.2 6.3.3.1.3	Large and Small Break LOCA	109.1 %	X		The spectrum of LOCAs was analyzed for Davis-Besse using an initial power level of 109.1% of 2772 MWt. The analyses were performed with NRC approved methods and models. The analyses bound the MUR power uprate. NRC Approval in Reference II-5 and supplemented by additional evaluations under the provisions of 10CFR50.59.
5.2.2.3	Overpressure Protection	100 %		X	The turbine trip event is the limiting overpressure transient for the steam generators. A turbine trip analysis was performed to support plant operability for main steam safety valve testing during the 2002 refueling outage. This analysis modeled a 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. Consequently, the small increase in pressure due to the MUR power uprate would be bounded.

II.2. Discussion of Events

A review of the UFSAR Chapter 15 accidents was performed to support the measurement uncertainty recapture power uprate for Davis-Besse. A summary of the evaluation for each accident is provided below.

15.1.2 *Anticipated Transients Without Scram (ATWS)*

The ATWS transients are considered beyond the original design basis of the B&W-designed plants. The results of these cases were used to predict the maximum pressure that would be reached if the reactor was not tripped. Because pressure well in excess of 4000 psi was predicted, the NRC required that additional protection systems be installed. In order to comply with 10CFR 50.62, the Babcock & Wilcox Owners Group (B&WOG) plant owners installed a diverse scram system (DSS) and the ATWS Mitigation System Actuation Circuitry (AMSAC) to trip the turbine and actuate auxiliary feedwater flow. Davis-Besse does not have a specific AMSAC system, but credits the Steam and Feedwater Rupture Control System (SFRCS) to provide those functions. The DSS initiates a redundant trip signal on high RCS pressure with an actuation setpoint corresponding to the RCS design pressure of 2500 psig. Turbine trip and emergency feedwater actuation are assured by SFRCS and Anticipatory Reactor Trip System (ARTS). Consequently, the peak RCS pressures predicted for the ATWS events for B&W-designed plants are significantly below the maximum pressure criterion allowed for the ATWS event, i.e. 3200 psig. Because the DSS setpoint, pressurizer safety valve setpoints, and pressurizer safety valve flow characteristics are not affected by the power uprate, a small increase in the initial core power will only result in a small increase in the rate of pressurization. There is significant margin in the analysis, however, to accommodate any small change in the peak pressure prediction.

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

The startup accident is a moderate frequency event that results from a spurious control rod withdrawal from hot zero power conditions. The acceptance criteria for the event are that the peak reactor coolant system (RCS) pressure does not exceed 2750 psig and the maximum allowed core power does not exceed 112% of rated power. Therefore, the primary reactor protection system (RPS) trip functions that are credited for this event are the high RCS pressure and core over-power. This reactivity addition event is considered a heat-up transient that results in pressurization of the RCS. The startup accident is also the limiting overpressure event for the RCS.

The transient is initiated from hot zero power conditions and as a result, the MUR power uprate has no affect on the initial conditions within the RCS. A spectrum of reactivity insertion rates (RIRs) is simulated to demonstrate compliance with the event acceptance criteria. For slow RIRs, the neutron and thermal power increase at nearly the same rate. The RCS temperature, and hence RCS pressure, increases rapidly to the high pressure trip setpoint. Regardless of the plant rated power level, the rate of power increase will not be changed and it will require the same amount of energy to be added to the RCS coolant to raise the temperature and pressure to the trip setpoint. Therefore, the MUR power uprate will still be bounded by the current plant UFSAR analysis from the peak pressure perspective. Since this transient is the limiting RCS pressure transient, no change to the high pressure trip setpoint will be required.

For faster insertion rates, the neutron power greatly exceeds the thermal power and the reactor will be tripped on high flux. The neutron power ranges from approximately 80% power to a peak of 400% power while the maximum thermal power is less than 70% power for the cases analyzed. As a result, changes to the high flux trip setpoint will have a negligible affect on peak thermal power. Consequently, the MUR power uprate will also be bounded by the current plant UFSAR analysis.

Based on the discussions presented above, the startup accident will remain bounding for the MUR power uprate. The Analyses of Record for the startup accident were accepted by the NRC as part of the approval of Technical Specification Amendment 128 (Reference II-4).

15.2.2 *Uncontrolled Control Rod Assembly Group Withdrawal at Power Accident*

The rod withdrawal accident is a moderate frequency event that results from a spurious control rod withdrawal from rated power conditions. The acceptance criteria for the event are that the peak reactor coolant system (RCS) pressure does not exceed 2750 psig and maximum allowed core power does not exceed 112% of rated power. Therefore, the primary reactor protection system (RPS) trip functions that are credited for this event are the high RCS pressure and core over-power. This reactivity addition event is considered a heat-up transient that results in pressurization of the RCS.

The initial core power level for the current rod withdrawal at power accident analyses is 100% of 2772 MWt. The over-power trip setpoint used in the analyses was 112% of 2772 MWt. With the MUR power uprate, the RCS average temperature and initial pressurizer level will not change. The steam space in the pressurizer will also not be affected.

A spectrum of reactivity insertion rates (RIRs) is simulated to demonstrate compliance with the event acceptance criteria. For slow RIRs, the neutron and thermal power increase at nearly the same rate. The RCS temperature, and hence RCS pressure, increases rapidly to the high pressure trip setpoint. Similar to the startup accident, regardless of the plant rated power level, it will require the same amount of energy over and above the initial value to raise the RCS temperature and pressure to the trip setpoint. A different RIR will become limiting, but the MUR power uprate will still be bounded by the current plant UFSAR analysis from the peak pressure perspective.

For fast RIRs, reactor protection is provided by the over-power trip setpoint. The transient response will be governed by the power difference between the initial core power and the over-power trip setpoint. The larger the difference between these values will result in a more severe transient. For the current UFSAR analysis, this difference (or the net energy added) is 12% of 2772 MWt. The over-power setpoint will be reduced to 110.2% of the MUR power level of 2817 MWt. This would result in an increase of 10.2% of 2817 MWt of energy added to the RCS, which is the same amount of net energy as in the current UFSAR analyses. Therefore, the rod withdrawal at power accident analyses described in the UFSAR will remain applicable for the MUR power uprate from the peak power perspective.

Based on the discussions present above, the rod withdrawal accident at power will remain bounding for the MUR power uprate. The Analyses of Record for this accident were accepted by the NRC as part of the approval of the original Davis-Besse FSAR, Reference II-5.

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

In the event that a control rod assembly can not be moved, localized power peaking and shutdown margin are evaluated. However, the dropped control rod accident is more severe and bounds both the stuck-out and stuck-in cases. The dropped control rod transient is a moderate frequency event. The acceptance criteria for the event are that the peak reactor coolant system (RCS) pressure does not exceed 2750 psig and maximum allowed core power does not exceed 112% of rated power. Therefore, the primary reactor protection system (RPS) trip functions that are credited for this event are the high RCS pressure and over-power. 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. 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 over-power (EOC). Conservative reactivity parameters coupled with a spectrum of cases, wherein the worth of the dropped rod and MTC are varied, ensure a bounding calculation.

The dropped control rod accident was reanalyzed. The power level used in the calculation was 102% of 2966 MWt (109.1% of 2772 MWt), which bounds the power uprate. The new analyses were performed using NRC approved methods and tools with the only change being the initial core power level. The results of the analyses showed that the existing UFSAR analyses remain bounding.

15.2.4 *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, but bounded by the rod withdrawal at power 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). The transient progression is determined by the combination of the dilution flow rate and the cycle-specific reactivity parameters. The earlier cycle/core designs resulted in greater reactivity addition rates, such that they are more limiting than the current cycle designs. The acceptance criteria for this event are that peak power will not exceed 112% rated thermal power and that the peak RCS pressure will be less than code pressure limits, i.e., 110% of the design pressure. Also, a minimum shutdown margin of 1% $\Delta k/k$ must be maintained during refueling conditions. Compliance to the shutdown requirement is demonstrated as part of the cycle-specific reload calculations because no system level transient is simulated and the results are largely unaffected by the MUR power uprate.

The moderator dilution accident is a relatively slow event. The resulting reactivity insertion rates (RIRs) from the dilution at power are bounded by the rates presented for the rod withdrawal at power accident, UFSAR Section 15.2.2. The peak RCS pressure and peak thermal power were also significantly less than the results of the rod withdrawal at power event. Although the UFSAR transient was initiated from 100% of 2772 MWt, no new analysis was performed to support the MUR because the system response is bounded by another accident. Therefore, the MDA analysis remains acceptable for the MUR power uprate. The Analyses of Record for this accident were accepted by the NRC as part of the approval of the original Davis-Besse UFSAR, Reference II-5.

15.2.5 *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 typically analyzed to the more restrictive criteria of the moderate frequency event category. For the locked rotor transient, no fuel cladding failure is allowed. These events are evaluated for each new fuel reload. The acceptance criteria for these events relate to the minimum allowed DNBR based on the applicable critical heat flux correlation for the fuel design being analyzed.

The analyses described in the UFSAR were performed at 102% of 2772 MWt. This power level bounds the MUR power uprate. In addition, the DNBR calculations are verified for each new core design. For both Davis-Besse Cycle 14 and Cycle 15, the DNBR analyses accounted for the MUR power uprate. Therefore, no new analyses were required to justify the power uprate. The original Analyses of Record for this accident are reflected in the Davis-Besse UFSAR, Reference II-5.

15.2.6 *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. The acceptance criteria for the event are that the peak reactor coolant system (RCS) pressure does not exceed 2750 psig and maximum allowed core power does not exceed 112% of rated power.

The analysis assumed that the plant was operating with one reactor coolant pump in each loop at 60% 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 that is still less than the rated core power. No RPS trip setpoints are challenged. The increase in coolant flow combined with an increase in power to 73% (thermal) does not result in an unacceptable minimum DNBR. The RCS pressure increases approximately 115 psi and remains below the high pressure reactor trip setpoint.

The maximum allowed power level for 2-pump operation is 55.1% of rated power, Table 2.2-1 of Reference II-2. Consequently, the analysis was performed with approximately 5% of rated power margin which envelopes the power measurement uncertainty. Therefore, the current UFSAR cold water accident bounds plant operation with the MUR power uprate. The Analyses of Record for this accident were accepted by the NRC as part of the approval of the original Davis-Besse FSAR, Reference II-5.

15.2.7 *Loss of External Electric Load and/or Main Turbine Trip*

The station was originally designed to withstand the effects of a load rejection transient without reactor or turbine trip. The reactor power would automatically be runback to the power level corresponding to the steam generator low level limit of 28.5 percent full power. The pilot operated relief valve (PORV) was available to relieve pressure to prevent reactor trip. The acceptance criteria for this event are that fuel damage would not occur and that the RCS pressure would not exceed the code pressure limit of 110% of the design pressure. The Analyses of Record for this accident were accepted by the NRC as part of the approval of the original Davis-Besse FSAR, Reference II-5.

As a result of plant changes, the PORV lift setpoint was increased to 2450 psig and the Anticipatory Reactor Trip System (ARTS) was added. This was approved by the NRC in Reference II-4. The ARTS results in a reactor trip whenever a turbine trip occurs with the reactor power at or above 45% rated power. These plant modifications essentially eliminated the possibility of an automatic plant runback except from a very low power level. The integrated control system will still try to run the plant power level back. If successful, the results described in the UFSAR would remain bounding for current plant operation as well as for the MUR power uprate.

If the loss of load occurs at a high power level, the reactor will be tripped on high RCS pressure and the UFSAR states that the transient would be bounded by the loss of normal feedwater accident. The loss of normal feedwater accident was analyzed at 102% of 2772 MWt and is bounding for the MUR power uprate. Therefore, the turbine trip will also be bounded.

15.2.8 *Loss of Normal Feedwater*

A loss of feedwater accident results from either a reduction in or the complete loss of secondary feedwater to the steam generators. The loss of feedwater may be caused by pump failure, valve closure or a feedwater line break. The normal loss of feedwater is a moderate frequency event. The acceptance criteria are that fuel failure shall not occur and that the peak RCS pressure will not exceed code pressure limits of 110% of the design pressure. This transient is also used to establish the minimum required auxiliary feedwater (AFW) flow rate. The feedwater line break event is considered a limiting fault event. The acceptance criteria are peak RCS pressure does not exceed ASME Service Level C limits and offsite dose is limited to a small fraction of 10CFR100 limits. 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 feedwater line break analyses presented in the UFSAR were accepted by the NRC as part of the approval of the original Davis-Besse FSAR, Reference II-5.

As a result of plant modifications, the Steam and Feedwater Line Rupture Control System (SFRCS) was added. For the feedwater line break scenario, the reverse differential pressure trip will be actuated. This will result in a nearly instantaneous automatic actuation of AFW, feedwater valve closure, a reactor trip and turbine trip. As a result, the severity of the transient is greatly reduced. The timing is independent of the power uprate. Further, the minimum AFW flow requirements are established by the loss of main feedwater transient which is based on 102% of 2772 MWt. As a result, the feedwater line break will remain valid for the MUR power uprate.

With the addition of the SFRCS, the loss of normal feedwater was reanalyzed to support increasing the normal pressurizer level from 200 to 220 inches. This analysis was performed at a power level of 102% of 2772 MWt and bounds the MUR power uprate. The RELAP5/MOD2-B&W computer code was used in the analysis with no other changes to methods or inputs. The RELAP5 code has been reviewed and approved by the NRC for analysis of LOCA and non-LOCA transients, Reference II-7.

15.2.9 *Loss of all AC Power to the Station Auxiliaries (Station Blackout)*

The loss of all AC power is a hypothetical case in which all unit AC power is lost. The unit batteries are available to supply DC power. There are redundant, quick-starting emergency diesel generators available to supply essential loads. In addition, a dedicated alternate AC power source (Station Blackout Diesel Generator) is also available. The acceptance criteria for this accident are that fuel damage will not occur and RCS pressure will not exceed code pressure limits of 2750 psig. The UFSAR analysis was initiated from 100% of 2772 MWt. The Analyses of Record for this accident were accepted by the NRC as part of the approval of the original Davis-Besse FSAR (Reference II-5).

On loss of power, the control rod breakers will deenergize and the control rods will be inserted. The turbine is tripped and the reactor coolant pumps will also be tripped. These actions occur regardless of initial core power level. As a result, the plant system response is bounded by the 4-to-0 pump coast down event. Since the pump coastdown transient was analyzed for 102% of 2772 MWt, the short-term response to loss of AC power event will be bounded. For the long-term, AFW will be automatically actuated on loss of reactor coolant pumps, which is not affected by the power uprate. The minimum required AFW flow rate is determined based on the loss of normal feedwater accident. Since the loss of feedwater event was analyzed for 102% of 2772 MWt, the AFW flow requirement will not be affected. Therefore, the loss of AC power level event will remain applicable for the MUR power uprate.

15.2.10 and 15.2.11 *Excessive Heat Removal due to Feedwater System Malfunction and Excessive Load Increase*

Excessive heat removal from the RCS can result from a malfunction or inadvertent operator adjustment of the feedwater control system which causes a reduction in feedwater temperature or an excessive increase in feedwater flow. A reduction in feedwater temperature could result if feedwater flow was diverted around the feedwater heaters without a corresponding reduction in feedwater flow. An increased feedwater flow could result if a feedwater control valve was opened to greater than its normal operating position. Excessive heat removal from the RCS will result in a maximum reactivity insertion to the core since the average reactor coolant temperature will decrease and the moderator temperature coefficient is most negative. The acceptance criteria are that no fuel damage will occur and the RCS pressure will not exceed code pressure limits.

For the decrease in feedwater temperature and increase in feedwater flow transients, the analyses were performed at 102 % of 2772 MWt to maximize the feedwater flow to the SGs. Therefore, the MUR power uprate will be bounded.

For the excessive load increase transient, the UFSAR states that this transient is bounded by the steam line break accident. Since the steam line break remains bounding for the MUR power uprate, this transient will also remain valid. The Analyses of Record for these accidents were accepted by the NRC as part of the approval of the original Davis-Besse FSAR (Reference II-5).

15.2.12 *Anticipated Variations in the Reactivity of the Reactor*

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 (MWe). 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 (MWe) and constant average temperature. Reactivity changes due to fuel depletion and xenon burnup occur very slowly; therefore, the positive reactivity insertion rates associated with these events are very small and are bounded by the control rod withdrawal at power event.

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. Since the reactivity insertion rates are bounded by a more severe transient, the current analyses of uncompensated reactivity changes support the power uprate. The Analyses of Record for this accident were accepted by the NRC as part of the approval of the original Davis-Besse FSAR (Reference II-5).

15.2.13 *Failure of Regulating Instrumentation*

This section of the UFSAR states that the event is bounded by other events described in Chapter 15 of the UFSAR. This failure is independent of initial core power level. Therefore, these events remain bounding for the MUR power uprate. The NRC Approval of the original analyses is in Reference II-5.

15.2.14 *External Causes*

This section of the UFSAR states that the event is addressed in Chapters 3 and 6 of the UFSAR in that these conditions must be considered. Earthquakes and tornadoes are independent of the core power level as precursors, however they are considered in the analyses. Therefore, this event remains bounding for the MUR power uprate. The NRC Approval of the original analyses is in Reference II-5.

15.3.1 *Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes which Actuates Emergency Core Cooling*

The 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, containment vessel pressure and temperature, and offsite dose consequence.

A spectrum of LOCAs has been analyzed for Davis-Besse using an initial power level of 109.1% of 2772 MWt. The analyses were performed with NRC approved methods and models (Reference II-8). The analyses bound the MUR power uprate.

15.3.2 *Minor Secondary Pipe Break*

The UFSAR only states that this accident is discussed as part of Subsection 15.4.4 (Steam Line Break).

15.3.3 *Inadvertent Loading of a Fuel Assembly Into an Improper Position*

The arrangement of assemblies with different fuel enrichments in the core will determine the power distribution of the core during normal operation. The loading of fuel assemblies into improper core positions or the incorrect preparation of the fuel assembly enrichment could alter the power distribution of the core.

Following each refueling, an incore power distribution is taken during startup testing and compared to calculated power distributions. Gross fuel assembly misplacement would be detected by the incore detectors during this phase by the fact that a radial power tilt is present or developing. Similarly, the out-of-core detectors will indicate quadrant tilt conditions. Strict administrative controls prevent enrichment errors during fuel fabrication and during fuel loading. These considerations are independent of the rated core power level. Therefore the existing analyses remain valid for the MUR power uprate. The NRC Approval of the original analyses is in Reference II-5.

15.4.1 *Waste Gas Tank Rupture*

The waste gas decay tank is used in the radioactive waste disposal system to store radioactive gaseous waste from the station until such time that the radioactive decay renders the gas safe for release to the site environment. Rupture of a waste gas tank would result in the premature release of its radioactive contents to the station ventilation system and to the atmosphere through the station vent.

This accident is not power dependent. The reactor coolant activities that were used in the offsite dose calculation were higher than the Technical Specification RCS activity limits. Consequently MUR uprate will be bounded.

15.4.2 *Steam Generator Tube Rupture (SGTR) Accident*

A Steam Generator Tube Rupture (SGTR) is a postulated double-ended rupture of a steam generator tube with unrestricted discharge from both ends of the tube. The acceptance criterion is related to offsite dose and further degradation of the primary-to-secondary pressure boundary beyond the affected tube.

A SGTR is a breach of the reactor coolant pressure boundary and results in a transfer of primary coolant to the secondary system. The core protection aspects of a SGTR are bounded by small break LOCA. Therefore, the SGTR event is analyzed to determine the offsite doses resulting from the release of contaminated primary coolant into the steam generator and to the atmosphere via the MSSVs.

The system response for the SGTR analysis of record is based on a constant leak rate of 435 gpm. The leak flow rate is based on critical flow from each end of the ruptured tube. This leak rate was assumed constant until the plant was cooled down to the decay heat removal cut-in temperature. This leak flow rate is conservative because it does not credit the decrease in the leakage rate with RCS depressurization or the secondary side pressurization following reactor trip and turbine trip. The SGTR calculation is independent of power level based on the analytical method used. Therefore, there is no impact on the system response due to the power uprate.

Beginning with Cycle 5, the fuel cycle length was extended, Section 15.A.7.0 of the UFSAR. Dose calculations are not revised for extended fuel cycles for accidents that do not involve a release of activity from the fuel, (i.e., the accident source term is based on the RCS activity) because the original analyses used the RCS activity ~ 5 μci/gm as described in the UFSAR Appendix 15A which are much higher than the Technical Specification 3.4.8 value of 1 μci/gm Dose equivalent I-131. The fission product inventory in the core was evaluated at 102% of 2772 MWt. Consequently, the MUR power uprate will be bounded.

15.4.3 *CRA Ejection Accident*

The rod ejection event is a postulated event involving a physical failure of a pressure barrier component in the Control Rod Drive assembly and subsequent ejection of the control rod. The event is classified as an infrequent event. The acceptance criteria for the Rod Ejection from full power event relate to peak RCS pressure and peak fuel enthalpy.

The ejection of a control rod with the reactor at full power causes a rapid positive reactivity insertion. Core power and fuel temperatures increase rapidly. The rapid fuel temperature rise produces negative Doppler reactivity feedback that terminates the power excursion. A reactor trip occurs on over-power and the reactor is returned subcritical by control rod insertion. The primary safety valves provide steam relief to limit the peak RCS pressure to less than the acceptance criterion. Limiting the reactivity worth of a given rod in the fuel design and the initial fuel enthalpy at full power will ensure that the peak fuel enthalpy does not exceed the maximum allowable limit. A rod ejection accident initiated from zero power will not be affected by the MUR power uprate because the over-power trip setpoint will be reduced such that the reactor will be tripped at the same net power level, i.e., 112% of 2772 MWt or 110.2% of 2817 MWt.

The current analysis for the Control Rod Ejection Accident for Davis-Besse was initiated from 102% of 2772 MWt.

Beginning with Cycle 5, the fuel cycle length was extended, Section 15.A.7.0 of the UFSAR. The fission product inventory in the core was evaluated at 102% of 2772 MWt. The radiation doses are included in the UFSAR. Consequently, the MUR power uprate will be bounded.

15.4.4 *Steam Line Break*

A steam line break is a rupture in the steam lines between the steam generators and the turbine. The rapid depressurization causes an increase in the main feedwater flow rate. The increase in steam flow to the break and the turbine results in a large overcooling of the RCS. The steam line break accident is the most severe overcooling transient. The acceptance criteria relates to effective core cooling, offsite dose release, reactor coolant system integrity, and containment vessel integrity. The core power level that is modeled in the analysis is 102% of rated power. The large temperature change coupled with end-of-cycle reactivity coefficients will result in the greatest challenge to maintaining adequate shutdown margin. The power level measurement uncertainty is also accounted for by conservatively increasing the OTSG inventory to bound plant operation at 102% power. Consequently, the analyses bound the MUR power uprate for the core response calculations.

Beginning with Cycle 5, the fuel cycle length was extended. Dose calculations are not revised for extended fuel cycles for accidents that do not involve the release of activity from the fuel, (i.e., the accident source term is based on the RCS activity) because the original analyses used the RCS activity ~ 5 μci/gm as described in the UFSAR Appendix 15A which are much higher than the Technical Specification 3.4.8 value of 1 μci/gm Dose equivalent I-131. The fission product inventory in the core was evaluated at 102% of 2772 MWt. Consequently, the MUR power uprate will be bounded.

15.4.5 *Break in Instrument Lines or Lines from Primary System that Penetrate Containment*

A break in fluid-bearing lines which penetrate the containment vessel could result in the release of radioactivity to the environment. This transient is typically represented by a double-ended break of the letdown line. This transient is considered a small break loss of coolant accident and the core cooling aspect is bounded by the spectrum of break sizes that are analyzed. Since the small break LOCA analyses have been analyzed at 109.1% of 2772 MWt, the letdown line break transient will be bounded for the MUR power uprate. Similar to the SG tube rupture analysis, this transient is performed to assess the offsite dose release consequences.

Beginning with Cycle 5, the fuel cycle length was extended, Section 15.A.7.0 of the UFSAR. Dose calculations are not revised for extended fuel cycles for accidents that do not involve in release of activity from the fuel, (i.e., the accident source term is based on the RCS activity) because the original analyses used the RCS activity ~ 5 μci/gm as described in the UFSAR Appendix 15A which are much higher than the Technical Specification 3.4.8 value of 1 μci/gm Dose equivalent I-131. The fission product inventory in the core was evaluated at 102% of 2772 MWt. Consequently, the MUR power uprate will be bounded.

15.4.6 *Major Rupture of Pipes Containing Reactor Coolant up to and Including Double-Ended Rupture of the Largest Pipe in the Reactor Coolant system (Loss-of-Coolant Accident)*

The core cooling aspects of loss of coolant accidents are described in UFSAR Section 6.3. This section of the UFSAR assesses the environmental consequences of the LOCA as well as the maximum hypothetical accident. Beginning with Cycle 5, the fuel cycle length was extended. The fission product inventory in the core was evaluated at 102% of 2772 MWt. The radiation doses for the extended fuel cycles have been included in the UFSAR. Consequently, the MUR power uprate will be bounded.

15.4.7 *Fuel-Handling Accident*

Mechanical damage to a fuel assembly is postulated during refueling operations. The analyses for this accident consider an accident inside containment, outside containment, and a dry fuel storage cask drop. The core power level is only used to determine the activity levels in the fuel-to-clad gap region prior to the accident.

Beginning with Cycle 5, the fuel cycle length was extended. Additional bounding analyses have been performed for high fuel burnup. The dose consequences were evaluated at 102% of 2772 MWt and remain bounding for the MUR.

15.4.8 *Effects of Toxic Material Release on the Control Room*

Toxic materials are not stored in volumes which would affect control room habitability. Further, this accident is independent of core power. The NRC Approval of the original analysis is in Reference II-5.

6.2.1.1.2 *LOCA Mass and Energy Release*

A spectrum of break sizes and locations were analyzed to determine the post-LOCA containment pressure and temperature response. The mass and energy release blowdown data were generated based on a core power level of 109.1 % of 2772 MWt, which conservatively bounds the MUR. The long-term mass and energy release data is calculated as part of the containment pressure and temperature analysis and was based on 102% of 2772 MWt. Since the MUR has no affect on the containment heat structures, free volume, or heat removal systems, and a conservative core power level was used for the mass and energy release data, then the UFSAR analyses will remain bounding for the MUR power uprate.

6.2.1.3.2 and 15.4.4.2.3.3 *MSLB Mass and Energy Release*

The main steam line break containment pressure and temperature response is based on a doubled-ended rupture of the main steam line. The mass and energy release blowdown data were generated based on a core power level of 102 % of 2772 MWt, which conservatively bounds the MUR. Since the MUR has no affect the containment heat structures, free volume, or heat removal systems, and a conservative core power level was used for the mass and energy release data, then the UFSAR analyses will remain bounding for the MUR power uprate.

6.3.3.1.2 and 6.3.3.1.3 *Large Break and Small Break Loss-of-Coolant Accidents*

The 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, containment vessel pressure and temperature, and offsite dose consequence.

A spectrum of LOCAs has been analyzed for Davis-Besse using an initial power level of 109.1% of 2772 MWt. The analyses were performed with NRC approved methods and models (Reference II-8). The analyses bound the MUR power uprate.

5.2.2.3 *Overpressure Protection*

The turbine trip, relative to RCS pressure and temperature response, as described in the UFSAR Section 15.2.7 is bounded by the normal loss of main feedwater transient. However, the turbine trip event is also the limiting overpressure transient for the secondary side of the steam generator. A turbine trip analysis was performed to support plant operability for main steam safety valve testing during the 2002 refueling outage and used NRC approved methods (Reference II-6). This analysis modeled an initial core power level of 3025 MWt (109.1% of 2772 MWt) and used the current installed main steam safety valve capacity. This analysis confirmed that the peak OTSG pressure was less than the ASME code allowable. Consequently, the small increase in pressure due to the MUR power uprate is bounded.

II. References:

II-1 Davis- Besse Updated Final Safety Analysis Report, Revision 25, June 2006.

II-2 Davis- Besse Nuclear Power Station No. 1, Technical Specifications through Amendment 275.

II-3 AREVA NP Document No. 43-10179PAB-06, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses", BAW-10179P-AB, Revision 6, August 2005.

II-4 NRC Letter Albert W. De Agazio, Sr. (NRC) to Donald C. Shelton (Toledo Edison), Serial Number DB-88-062, Amendment No. 128 to Facility Operating License No. NPF-3: High Pressure Reactor Trip Setpoint and PORV Setpoint (TAC No. 66727), dated December 28, 1988.

II-5 Safety Evaluation Report by the Office of Nuclear Reactor Regulation, United States Nuclear Regulatory Commission, in the Matter of Toledo Edison Company and Cleveland Electric Illumination Company, Davis-Besse Nuclear Power Station, Unit 1, Docket No. 50-346, December 1976.

II-6 AREVA NP Document No. 43-10193PA-00, RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurized Water Reactors, BAW-10193P-A, Revision 0, dated January 2000. (SER Stewart N. Bailey (NRC) to J.J. Kelly, TAC No. M93346, dated October 15, 1999.)

II-7 AREVA NP Document No. 43-10164PA-04, RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis, BAW-10164PA, Revision 4, December 2002.

II-8 AREVA NP Document No. 43-10192PA-00, BWNT LOCA – BWNT Loss-of-coolant Accident Evaluation Model for Once-Through Steam Generator, BAW-10192PA, Revision 0, July 1998.

III. Accidents and transients for which the existing analyses of record do not bound plant operation at the proposed uprated power level

1. This section covers the transient and accident analyses that are included in the plant's UFSAR (typically Chapter 14 or 15) and other analyses that are required to be performed by licensees to support licensing of their plants (i.e., radiological consequences, natural circulation cooldown, containment performance, anticipated transient without scrams, station blackout, analyses for determination of environmental qualification parameters, safe shutdown fire analysis, spent fuel pool cooling, flooding).

2. For analyses that are covered by the NRC approved reload methodology for the plant, the licensee should:

   A. Identify the transient/accident that is the subject of the analysis

   B. Provide an explicit commitment to re-analyze the transient/accident, consistent with the reload methodology, prior to implementation of the power uprate

   C. Provide an explicit commitment to submit the analysis for NRC review, prior to operation at the uprated power level, if NRC review is deemed necessary by the criteria in 10 CFR 50.59

   D. Provide a reference to the NRC's approval of the plant's reload methodology

3. For analyses that are not covered by the reload methodology for the plant, the licensee should provide a detailed discussion for each analysis. The discussion should:

   A. Identify the transient or accident that is the subject of the analysis

   B. Identify the important analysis inputs and assumptions (including their values), and explicitly identify those that changed as a result of the power uprate

   C. Confirm that the limiting event determination is still valid for the transient or accident being analyzed

   D. Identify the methodologies used to perform the analyses, and describe any changes in those methodologies

   E. Provide references to staff approvals of the methodologies in Item D. above

   F. Confirm that the analyses were performed in accordance with all limitations and restrictions included in the NRC's approval of the methodology

   G. Describe the sequence of events and explicitly identify those that would change as a result of the power uprate

H. Describe and justify the chosen single-failure assumption

I. Provide plots of important parameters and explicitly identify those that would change as a result of the power uprate

J. Discuss any change in equipment capacities (e.g., water supply volumes, valve relief capacities, pump pumping flow rates, developed head, required and available net positive suction head (NPSH), valve isolation capabilities) required to support the analysis

K. Discuss the results and acceptance criteria for the analysis, including any changes from the previous analysis

Response to III

The review of the UFSAR accidents concluded that all transients are bounded by the current analyses of record for the Davis-Besse plant. A new analysis was performed for the Dropped Control Rod Accident to support the MUR power uprate but it was concluded that the existing UFSAR analysis still remained bounding. Therefore, since all analyses of record for Davis-Besse remained valid, no new accident or transient analyses were required to support the MUR power uprate.

IV. Mechanical/Structural/Material Component Integrity and Design

1. A discussion of the effect of the power uprate on the structural integrity of major plant components. For components that are bounded by existing analyses of record, the discussion should cover the type of confirmatory information identified in Section II, above. For components that are not bounded by existing analyses of record, a detailed discussion should be provided.

   A. This discussion should address the following components:

      i. reactor vessel, nozzles, and supports
      ii. reactor core support structures and vessel internals
      iii. control rod drive mechanisms
      iv. Nuclear Steam Supply System (NSSS) piping, pipe supports, branch nozzles
      v. balance-of-plant (BOP) piping (NSSS interface systems, safety related cooling water systems, and containment systems)
      vi. steam generator tubes, secondary side internal support structures, shell, and nozzles
      vii. reactor coolant pumps
      viii. pressurizer shell, nozzles, and surge line
      ix. safety-related valves

   B. The discussion should identify and evaluate any changes related to the power uprate in the following areas:

      i. stresses
      ii. cumulative usage factors
      iii. flow induced vibration
      iv. changes in temperature (pre- and post-uprate)
      v. changes in pressure (pre- and post-uprate)
      vi. changes in flow rates (pre- and post-uprate)
      vii. high-energy line break locations
      viii. jet impingement and thrust forces

   C. The discussion should also identify any effects of the power uprate on the integrity of the reactor vessel with respect to:

      i. pressurized thermal shock calculations
      ii. fluence evaluation
      iii. heatup and cooldown pressure-temperature limit curves
      iv. low-temperature overpressure protection
      v. upper shelf energy
      vi. surveillance capsule withdrawal schedule

D. The discussion should identify the code of record being used in the associated analyses, and any changes to the code of record.

E. The discussion should identify any changes related to the power uprate with regard to component inspection and testing programs and erosion/corrosion programs, and discuss the significance of these changes. If the changes are insignificant, the licensee should explicitly state so.

F. The discussion should address whether the effect of the power uprate on steam generator tube high cycle fatigue is consistent with NRC Bulletin 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes," February 5, 1988.

RESPONSE to IV

IV.1.A.i Reactor Vessel Structural Evaluation

The revised design conditions were reviewed for impact on the existing design basis analyses for the reactor vessel. The evaluation showed that the temperature changes due to the MUR uprate are bounded by those used in the existing analyses. Therefore the existing loads remain valid and the stresses and fatigue values also remain valid. Thus, the existing stress reports for the reactor vessel remain applicable for the uprated power conditions.

IV.1.A.ii Reactor Vessel Internals Structural Evaluation

The revised design conditions were reviewed for impact on the existing design basis analyses for the reactor vessel internals. The evaluation showed that the temperature changes due to the MUR uprate are bounded by those used in the existing analyses. Therefore the existing loads remain valid and the stresses and fatigue values also remain valid.

IV.1.A.iii Control Rod Drive Mechanism Structural Evaluation

The revised design conditions were reviewed for impact on the existing design basis analyses for the control rod drive mechanisms. The evaluation showed that the temperature changes due to the MUR uprate are bounded by those used in the existing analyses. Therefore the existing loads remain valid and the stresses and fatigue values also remain valid. Thus, the existing stress reports for the control rod drive mechanisms remain applicable for the uprated power conditions.

IV.1.A.iv Reactor Coolant Piping and Supports Structural Evaluation

The revised design conditions were reviewed for impact on the existing design basis analyses for the reactor coolant piping and supports. The evaluation showed that the temperature changes due to the power MUR uprate are bounded by those used in the existing analyses. Therefore the existing loads remain valid and the stresses and fatigue values also remain valid. Thus, the existing stress reports for the reactor coolant piping and supports remain applicable for the uprated power conditions.

IV.1.A.v Balance of Plant (BOP) Piping (NSSS Interface Systems, Safety-Related Cooling Water Systems, and Containment Systems)

The structural analyses of the piping attached to the RCS (decay heat line, makeup and purification line, high and low pressure injection lines) use anchor motions from the RCS structural analyses. As discussed in Section IV.1.A.iv, these anchor motions do not change due to the uprated power conditions.

The revised design conditions for the BOP piping (NSSS Interface Systems, Safety-Related Cooling Water Systems, and Containment Systems) were reviewed for impact on the existing design basis analyses for the reactor coolant system attached piping and supports. The evaluation showed that the temperature changes due to the MUR uprate are bounded by those used in the existing analyses. Therefore the existing loads remain valid and the stresses and fatigue values also remain valid.

IV.1.A.vi Once Through Steam Generator Structural Evaluation

The revised design conditions were reviewed for impact on the existing design basis analyses for the steam generator. The evaluation showed that the temperature changes due to the MUR uprate are bounded by those used in the existing analyses. Therefore the existing loads remain valid and the stresses and fatigue values also remain valid. Thus, the existing stress reports for the steam generator remain applicable for the uprated power conditions.

IV.1.A.vi.1 The steam generator tubes, secondary side internal support structures, shell and nozzles are bounded by the existing analyses of record.

IV.1.A.vi.2 Once Through Steam Generator Tubes Evaluation

Topical report BAW-10146 (Reference IV-3) established the minimum required steam generator tube wall thickness for the B&W 177-FA plants. Tube loads were calculated for normal operating and faulted conditions. Normal operating tube loads were determined using design operating transients and were combined with tube geometry to calculate minimum allowable tube wall thickness that satisfy the acceptance criteria of NRC Draft RG 1.121. Faulted condition tube loads are those arising from a safe shutdown earthquake, a loss of coolant accident, a main steam line break and a feedwater line break. These loads were used to calculate minimum wall thickness based on the limits of NRC Draft RG 1.121 and ASME Code, Section III, Appendix F. The MUR uprate operating conditions were compared with the existing design conditions. The comparison showed that the power uprate will not result in operation outside 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 minimum required tube wall thickness for normal operating conditions will not be affected by the power uprate.

Tube loads for the faulted conditions were calculated for LOCA, MSLB, and FWLB accident conditions considering thermal and pressure loads on the steam generator. The MUR uprate operating temperatures were compared with the existing design temperatures. The comparison showed that the existing design temperatures bound the power uprate temperatures. This means that the existing tube loads due to LOCA, MSLB and FWLB will not change as a result of the power uprate.

In addition, a review of calculations performed which assessed the integrity of tubes containing flaws of various types when subjected to operating and accident loads was conducted. This review ensured that existing structural margins are maintained for the MUR uprate design conditions.

IV.1.A.vii Reactor Coolant Pump Structural Evaluation

The revised design conditions were reviewed for impact on the existing design basis analyses for the reactor coolant pump. The evaluation showed that the temperature changes due to the MUR uprate are bounded by those used in the existing analyses. Therefore the existing loads remain valid and the stresses and fatigue values also remain valid. Thus, the existing stress reports for the reactor coolant pump remain applicable for the uprated power conditions.

IV.1.A.viii Pressurizer Structural Evaluation

The revised design conditions were reviewed for impact on the existing design basis analyses for the pressurizer. The evaluation showed that the temperature changes due to the MUR uprate are bounded by those used in the existing analyses. Therefore the existing loads remain valid and the stresses and fatigue values also remain valid. Thus, the existing stress reports for the pressurizer remain applicable for the uprated power conditions.

IV.1.A.ix Safety Related Valves

The revised design conditions were reviewed for impact on the existing design basis analyses for the safety related valves. The evaluation showed that the temperature changes due to the MUR uprate are bounded by those used in the existing analyses. Safety analysis confirmed the installed capacities and lift setpoints of the RCS and Main Steam relief valves to be valid for the MUR Conditions. Therefore the existing loads remain valid and the stresses and fatigue values also remain valid. Safety related valves were reviewed within the system and program evaluations. None of the safety related valves required a change to their design or operation as a result of the MUR.

IV.1.B.i. Stresses

IV.1.B.i.1. BOP Piping and Support System

The MUR conditions remain bounded by the system design. No changes to the Code of Record (COR) are required. Therefore, it was concluded that the current stress analysis of record will remain applicable for the Davis-Besse plant at the MUR conditions.

IV.1.B.ii. Cumulative Usage Factors

The revised design conditions for the NSSS Components and BOP piping (NSSS Interface Systems, Safety-Related Cooling Water Systems, and Containment Systems) were reviewed for impact on the existing design basis analyses. The evaluation showed that the temperature changes due to the MUR uprate are bounded by those used in the existing analyses. Therefore the existing loads remain valid and the stresses and fatigue values (cumulative usage factors) also remain valid. The changes resulting from the MUR power uprate have negligible effect on the NSSS Components and BOP piping (NSSS

Interface Systems, Safety-Related Cooling Water Systems, and Containment Systems) structural analyses. The design conditions used in the existing analyses for the NSSS Components and BOP piping (NSSS Interface Systems, Safety-Related Cooling Water Systems, and Containment Systems) remain bounding for the MUR power uprate.

The following table is a list of the Original Code of Record for the NSSS Components of interest. Calculated stresses are contained in specific calculation files (i.e., stress report summary documents) maintained by AREVA NP for Davis-Besse. These documents include a tabulation of the currently calculated maximum stress intensities/stress ranges with a comparison to stress allowables, cumulative usage factors, and other special stress limits.

Component	Original Code of Record *
Pressurizer	ASME B&PVC Section III 1968 Edition with Addenda through Summer 1968
Reactor Coolant Piping	ANSI B31.7 Draft February 1968 with Errata June 1968
Reactor Vessel	ASME B&PVC Section III 1968 Edition with Addenda through Summer 1968
Once Through Steam Generator	ASME B&PVC Section III 1968 Edition with Addenda through Summer 1968
Reactor Coolant Pump	ASME B&PVC Section III 1968 Edition with Addenda through Winter 1968
Control Rod Drive Mechanism	ASME B&PVC Section III 1968 Edition with Addenda through Summer 1968 (part length) and Summer 1970 (full length)

* The Original Code of Record comes from Table 5.2-1 of the UFSAR (Reference IV-2). The MUR power uprate does not impact the Code of Record.

IV.1.B.iii. Flow Induced Vibration

IV.1.B.iii.1. Reactor Vessel Internals Flow-Induced Vibration Evaluation

Topical report BAW-10051 (Reference IV-1) presents the design analysis of the RV internals and incore instrument nozzles subjected to operational flow-induced vibration loading for the B&W 177-FA plants.

The maximum RCS volumetric flow rate associated with the MUR uprate is 11% higher than the one considered in the topical report. A comparative analysis was performed to evaluate the effects of the new operational condition. This analysis concluded that the new operational condition after MUR uprate is bounded by the topical report. The RV internals and incore instrument nozzles are structurally adequate for flow-induced vibration.

IV.1.B.iv. Changes in Temperature (Pre- and Post-Uprate)

Calculations were completed to define the RCS and steam generator conditions for the Davis-Besse MUR power uprate. New operating conditions were defined for a 1.63% (1.7 used for conservatism) power increase (conservative relative to the actual 1.63% MUR power uprate). Specific outputs of the calculations include T_{hot} and T_{cold}. T_{ave} (vessel) has been maintained at 582°F in all cases. It was determined that there is approximately a 0.8°F increase in ΔT across the core (T_{hot} increases by approximately 0.4°F and T_{cold} decreases by approximately 0.4°F) due to the MUR power uprate relative to the current operating temperatures at 2772 MWt. Therefore, the normal temperature design criteria used as inputs to the core design and safety analyses were changed to reflect the impact of changes in temperature (pre- and post-uprate) due to the MUR power uprate.

IV.1.B.iv.1. Evaluation of Potential for Thermal Stratification

Thermal stratification in the lines attached to the primary side of the RCS occurs mainly during heatup and cooldown. The 100% power hot and cold leg temperatures that the plant has been operating at are essentially the same as those for the MUR uprate. This means that the effects of thermal stratification will not change as a result of the power uprate.

NRC Bulletin 88-11 addresses the issue of surge line thermal stratification. Thermal stratification in the surge line occurs mainly during plant heatup and cooldown and is driven due to the temperature difference between the hot leg and the pressurizer. The current operating temperature of the hot leg will increase very slightly due to MUR uprate. A higher hot leg temperature gives a lower temperature differential between the hot leg and the pressurizer which in turn lessens the stratification effects. This means that stress and fatigue in the surge line which is attributed to thermal stratification is bounded by the existing analysis.

IV.1.B.v. Changes in Pressure (Pre- and Post-Uprate)

Calculations were completed to define the RCS and steam generator conditions for the Davis-Besse MUR power uprate. New operating conditions were defined for a 1.63 (1.7 used for conservatism)% power increase (conservative relative to the actual 1.63% MUR power uprate). Pressurizer performance (level setpoint and pressure control) were evaluated with regard to the MUR power uprate. It was determined that that the change in pressurizer level that occurs as the result of transients will be essentially the same with MUR power uprate. Therefore, no change in the pressurizer level setpoint is required. It was also determined that primary pressure control will not change with the MUR power uprate. Therefore, the normal pressure design criteria used as inputs to the core design and safety analyses will remain the same.

IV.1.B.vi. Changes in Flow Rates (Pre- and Post-Uprate)

Calculations were completed to define the RCS and steam generator conditions for the Davis-Besse MUR power uprate. New operating conditions were defined for a 1.63% power increase (1.7% was used to be conservative relative to the actual power increase). It was determined that the MUR power uprate does not have an appreciable effect on RCS mass flow (<<0.1%). Therefore, the changes in mass flow rates (pre- and post-uprate) will have a negligible impact on core design and safety analyses.

IV.1.B.vii. High Energy Line Break Locations

IV.1.B.vii.1. Leak Before Break Evaluation

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. Topical report BAW-1847 Rev. 1 (Reference IV-4) presents the LBB evaluation of the RCS primary piping. It 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. 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. An evaluation was performed which determined the impact of the MUR uprate design conditions on the inputs to the LBB analyses is negligible, and the LBB conclusions remain unchanged.

IV.1.B.vii.2 Reactor Coolant System Loss of Coolant Accident Forces Evaluation

Topical report BAW-1621 (Reference IV-5) addresses the RCS components for primary break Loss-of-Coolant Accident (LOCA) loadings. The breaks considered were limited break ruptures of the primary piping. Due to LBB qualification of the hot and cold legs, the RCS was requalified for snubber removal considering breaks in the surge line and steam line. These locations were identified via the stresses and fatigue usage existing in the high energy piping. The MUR uprate 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 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.

IV.1.B.viii. Jet Impingement and Thrust Forces

See Section IV.1.B.vii.2

IV.1.B.viii.1. Reactor Coolant System Loss of Coolant Accident Forces Evaluation

The evaluation showed that the asymmetric cavity pressure forces, thrust loads, and jet impingement loads remain bounded by the values in the existing analyses.

IV.1.C.i 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 considering a 32 EFPY inside surface fluence of 1.124 $x10^{19}$ n/cm^2 (E > 1.0 MeV) based on a 1992 fluence projection plus 5% to account for the MUR power uprate. The 1992 fluence projection bounds a more recent 2006 fluence projection for the 32 EFPY inside surface fluence and provides a more conservative RT_{PTS} value. The screening criterion for this weld metal is 300°F. Therefore, the reactor vessel will remain within its limits for PTS after the MUR power uprate.

IV.1.C.ii Fluence Evaluation

The fluence analysis assumed the following:

- Reactor nominal power for cycles 13-14 = 2772 MWt,
- Reactor nominal power for Cycle 15 = 2817 MWt,
- EFPY at EOC 14 = 17.6174 and
- Maximum inside surface (wetted surface) fluence on limiting weld at EOC12 = 4.63E18 n/cm^2.

The pertinent fluence results [maximum (wetted surface) fluence on limiting weld] are:

- EOC14 = 5.666E+18 n/cm^2,
- Cycle 15 + 16 with power uprate = 1.191E+18 n/cm^2, and
- EOC 16 with power uprate = 6.857E+18 n/cm^2
- The total irradiation time at the EOC14 was 17.6174 EFPY

Conclusions:

Based on the above information, the effect of the power uprate on the fluence on the limiting weld at the end of cycle 16 was an increase of no more than 0.3 %, which is obviously much less than the assumed maximum value of 5 %. Naturally, the effect will be more pronounced for longer irradiation times under uprate conditions. Also, it is noted that the EOC16 fluence of 6.857E+18 n/cm^2 is less than the 7.25E+18 n/cm^2 used in the PT curve basis. The current PT curves (Technical Specification Figures 3.4-2, 4.3-3 and 3.4-4) at 21 EFPY are valid for the power uprate.

IV.1.C.iii Heatup and Cooldown Pressure / Temperature Limit Curves

The current P-T limit curves (Technical Specification Figures 3.4-2, 4.3-3 and 3.4-4) are licensed 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 analyzed inside surface neutron fluence for weld WF-182-1 is 7.25 $x10^{18}$ n/cm^2 (E > 1.0 MeV). Adjusted reference temperature values were calculated in accordance with Regulatory Guide 1.99, Revision 2. Inputs affecting the adjusted reference temperatures and P-T curves remain unchanged under the MUR power uprate, with the exception of neutron fluence. Changes to the core power level will affect neutron flux, which will affect neutron fluence, which will ultimately affect the validity period of the current P-T curves.

The impact of the MUR power uprate on the P-T curves was assessed by performing a revised (2006) neutron fluence projection that evaluated recent core design changes and conservatively assumed a 5% neutron flux increase due to the MUR power uprate starting in cycle 15. The result of the conservative fluence projection is that the limiting fluence of 7.25×10^{18} n/cm^2 (E > 1.0 MeV) on weld WF-182-1 will not be exceeded prior to 21 EFPY. Thus, the Davis-Besse P-T curves and LTOP limits for 21 EFPY remain valid.

IV.1.C.iv Low-Temperature Overpressure Protection (LTOP)

As described in section IV.1.C.iii above, the currently licensed limiting RV fluence will not be reached prior to 21 EFPY. Thus, the current LTOP limits in the 21 EFPY P-T curves do not need to be modified for the MUR.

IV.1.C.v Effect on Low Upper Shelf Energy

Due to the increase in fluence from a power uprate, low upper-shelf toughness was evaluated to ensure compliance with Appendix G to 10 CFR Part 50. If the limiting reactor vessel beltline material's Charpy upper shelf energy (USE) is projected to fall below 50 ft-lb, an equivalent margins assessment must be performed. The limiting reactor vessel beltline material for Davis-Besse is weld WF-182-1, the upper to lower shell circumferential weld.

An equivalent margin assessment was performed for weld WF-182-1 outside of the MUR power uprate to address the possibility that the Charpy USE may fall below 50 ft-lb due to increased neutron fluence. Weld WF-182-1 was evaluated for ASME Levels A, B, C, and D Service Loadings based on the evaluation acceptance criteria of the ASME Code, Section XI, Appendix K.

The analysis demonstrated that the limiting reactor vessel beltline weld at Davis-Besse satisfies the ASME Code requirements of Appendix K for ductile flaw extensions and tensile stability using projected low upper-shelf Charpy impact energy levels for the weld material at 32 EFPY considering an inside surface fluence of 1.02×10^{19} n/cm^2 (E> 1.0 MeV) based on a 2006 fluence projection.

IV.1.C.vi 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.

IV.1.D. Codes of Record

IV.1.D.1. CRDM Motor Tube

The code of record from the D-B UFSAR Chapter 3 (Reference IV-2) for the CRDM Motor Tube (Reactor Coolant Pressure Boundary) is ANSI B31.7, Class I; ASME Section III Class 1, 1971. No changes to the code of record for the CRDM Motor Tube (Reactor Coolant Pressure Boundary) are required for the MUR power uprate.

IV.1.D.2. Steam Generator System

The code of record from the D-B UFSAR Chapter 3 (Reference IV-2) for the Once Through Steam Generator is ASME Section III, Class A, 1968. No changes to the code of record for the Once Through Steam Generator are required for the MUR power uprate.

IV.1.D.3. Makeup and Purification System

The codes of record from the D-B UFSAR Chapter 3 (Reference IV-2) for the Makeup and Purification system are:

Letdown Coolers, ASME Section III, 1977
Valves, ASME Section I, II, & III, 1968, 1971, 1986; ANSI B16.5 N3
Filters, ASME Section III, Class 3, Seal Injection, Class 2
Tanks, ASME Section III, Class C, Winter 1968
Pumps, ASME Pump and Valve Code, Class II, Nov. 1968

No changes to the code of record for the Makeup and Purification system are required for the MUR power uprate.

IV.1.D.4. Chemical Addition System

The codes of record from the D-B UFSAR Chapter 9 (Reference IV-2) for the Chemical Addition system are:

Piping and Valves, ASME Section III, Class 3, 1971; ANSI B31.1.0, 1967; ANSI B16.5 (or MSSS SP-66)
Pumps, ASME Draft Pump and Valve Code, November 1968
Tanks, ASME Section III, 1968

No changes to the code of record for the Chemical Addition system are required for the MUR power uprate.

IV.1.D.5. Auxiliary Feedwater System

The codes of record for the Auxiliary Feedwater System are:

Pumps, ASME Section III, Class 3, 1971
Piping, ASME Section III, Class 3, 1971
Valves, ASME Section III, Class 3, 1971

Tanks, AWWA (American Water Works Association)

No changes to the code of record for the Auxiliary Feedwater System are required for the MUR power uprate.

IV.1.D.6. Containment Isolation System

The codes of record for the containment isolation system are:

Containment Vessel, ASME Section III, Class B, Paragraph M-132 1968 through Summer 1969 Addenda[1]
Equipment Hatch and Locks, ASME Section VIII, Paragraphs UG 27, 29, 33 and UA6[2]
Penetrations (including refueling tubes), ASME Section III, Class 2 Components[1]
Containment Vessel Isolation Valves, ASME Section III, Nuclear Components Class 2[1]
Containment Vessel Relief Valves, ASME Section III, Article 16, Class 2[1]

[1] From Reference IV-2.
[2] From Reference IV-6.

No changes to the code of record for the Containment Isolation System are required for the MUR power uprate.

IV.1.D.7. Core Flooding System

The codes of record for the Core Flooding System are:

Tanks, ASME Section III, C, 1968
Piping, ASME Section III Class 2, 1971

No changes to the code of record for the Core Flooding System are required for the MUR power uprate.

IV.1.D.8. High Pressure Injection System

The codes of record for the High Pressure Injection System are:

Valves, ASME Section III, 1968
Pump, ASME Pump and Valve Code, Class II, Nov., 1968
Piping, ASME Section III, Class 2, 1971

No changes to the code of record are required for the High Pressure Injection system for the MUR power uprate.

IV.1.D.9. Containment Vessel Emergency Sump Strainer, Trash Racks, and Recirculation Lines

The code of record for the Containment Vessel Emergency Sump strainer, trash racks, and recirculation lines is ASME Section III, Class 2.

No changes to the code of record are required for the Containment Vessel Emergency Sump strainer, trash racks, and recirculation lines for the MUR power uprate.

IV.1.D.10. Decay Heat Removal/Low Pressure Injection System

The codes of record for the decay heat removal/low pressure injection system components are:

Piping, ASME Section III, Class 2, 1971
DH Coolers, ASME Section III, Class 2, 1968
Pumps, ASME Section II
Valves, ASME Section III, Class 2, 1971

No changes to the code of record for the decay heat removal/low pressure injection system are required for the MUR power uprate.

IV.1.D.11. Containment Spray System

The codes of record for the containment spray system components are:

Piping, Valves and Nozzles: ASME Section III, Class 2, 1971
Pumps: ASME Draft Pump and Valve Code November 1968

No changes to the codes of record for the containment spray system components are required for the MUR power uprate.

IV.1.D.12. Spent Fuel Pool Cooling System

The codes of record for the spent fuel pool cooling system components are:

Pumps, ASME Section III, Class 3, 1971
Filters, ASME Section III, Class 3, 1980
Demineralizer Tanks, ASME Section III, Class 3, 1971
Piping, ASME Section III, Class 3, 1971
Pressure Safety Valve, ASME Section III, Class 3, 1974
Heat Exchangers, (shell) ASME Section VIII, 1971, (tube) ASME Section III, Class 3, 1971

No changes to the codes of record for the spent fuel pool cooling system components are required for the MUR power uprate.

IV.1.D.13. BOP Piping and Supports System

The codes of record for the BOP piping and supports system components are:

<u>Piping:</u>
ASME Section, 1971 edition
USAS B31.1, Power Piping Code, 1967 edition and its addenda
<u>Pipe supports</u>:
MSS Standards SP-58 and SP-69
AISC Manual of Steel construction, 7^{th} edition

IV.1.E Changes to Component Inspection and Testing Programs

IV.1.E.1. Alloy 600 Primary Water Stress Corrosion Cracking (PWSCC)

The effects of a temperature increase resulting from the power uprate on Alloy 600 PWSCC have 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_{ave}, or the pressure and T_{sat}, the impact is limited to Alloy 600 components and welds operating near T_{hot}. Examination of the AREVA NP 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 to be very limited.

IV.1.E.2. Inservice Testing (IST) Program

10 CFR 50.55a(f), Inservice Testing Requirements, requires the development and implementation of an Inservice Testing (IST) Program. The applicable program requirements are specified in ASME OM Code-1995 Edition, 1996 addenda. Davis-Besse has developed and is implementing an Inservice Testing (IST) Program for Pumps and Valves per these requirements. This evaluation reviewed the impact to the Inservice Testing Program as part of the MUR uprate conditions up to the original licensed reactor thermal power (102% of 2772 MWt) and concluded that the MUR uprate is bounded by current analysis and any changes are insignificant.

IV.1.E.3. Inservice Inspection (ISI) Program

10CFR 50.55a(g), Inservice Inspection Requirements, requires the development and implementation of an Inservice Inspection (ISI) Program. The applicable program requirements are specified in ASME Section XI. Davis-Besse has developed and is implementing an Inservice Inspection (ISI) Program per these requirements. This evaluation evaluated the impact to the Inservice Inspection Program as part of the MUR uprate conditions up to of the original licensed reactor thermal power (102% of 2772 MWt) and concluded that the MUR uprate is bounded by current analysis and no changes are required.

IV.1.E.4 Erosion / Corrosion (FAC) Program

The MUR conditions are bound by the design conditions for the FAC Program. Therefore, the predicted increases in maximum component wear rates and reductions in service lives can be managed by the Davis-Besse FAC program.

IV.1.F Impact of NRC Bulletin 88-02 "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes" and NRC Information Notice 2002-02 (including Supplement 1) "Recent Experience with Plugged Steam Generator Tubes" Upon the D-B MUR Power Uprate

Since the NRC Bulletin 88-02 is not applicable to the OTSG designs, there is no impact from this bulletin upon the MUR power uprate and no action is required to address this bulletin. A more relevant notice for OTSG designs to consider would be the NRC Information Notice 2002-02, Recent Experience with Plugged Steam Generator Tubes, dated January 2002 and July 2002 for Supplement 1. Unlike bulletins, information notices do not constitute NRC requirements; therefore, no specific actions or written response is required. However, D-B did review the information for applicability to their facility and considered taking appropriate actions to avoid similar problems. The EPRI topical report 1008438, Three Mile Island Plugged Tube Severance (A Study of Damage Mechanism), addresses the concerns identified with the Information Notice 2002-02.

The results and findings of the EPRI Report 1008438 concluded that certain types of tube degradation can continue to occur in any steam generator after the tube has been taken out of service. For the B&W OTSGs, it was concluded that the only real vulnerability for tube severance is the growth of circumferential cracks due to high cycle fatigue. However, for a swollen, plugged tube, any degradation mechanism has the potential to provide an initiating site for failure.

In response to the findings, AREVA NP has implemented steam generator plugging and de-plugging maintenance procedures that will prevent such incidences from occurring in the future. D-B has complied with these and all other recommendations to mitigate the consequences of over-pressurized tubes in the OTSGs. To address tubes that were plugged prior to the NRC Information Notice 2002-02 that may be susceptible to tube swelling, D-B has plugged and stabilized all of the adjacent/neighboring tubes.

To address the possibility of circumferential tube cracks eventually severing due to high cycle fatigue, the OTSG stabilization criteria have historically required stabilization of all circumferential crack-like indications regardless of the radial location or elevation. In addition, the OTSG stabilization criteria have historically required stabilization of circumferentially-oriented volumetric indications in regions of high cross flows. Therefore, the findings of the EPRI Report 1008438 have always been employed for these degradation types.

Therefore, there are no FIV concerns related to the tube bundle associated with the MUR power uprate relevant to findings provided by NRC Information Notice 2002-02 or the EPRI Report 1008438 that have not already been evaluated.

IV. References:

IV-1 AREVA NP Document BAW-10051, Design of Reactor Internals and Incore Instrument Nozzles for Flow-Induced Vibration, September 1972.

IV-2 Davis-Besse Update Safety Analysis Report, Rev. 25, June 2006.

IV-3 AREVA NP Document 43-10146-00, BAW-10146, Determination of Minimum Required Tube Wall Thickness for 177-FA Once-Through Steam Generators, November 1980.

IV-4 AREVA NP Document 77-1153295-01, BAW-1847 Rev. 01, The B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSS, September 1985.

IV-5 AREVA NP Document 43-1621-00, BAW-1621, Effects of Asymmetric LOCA Loads – Phase II, July 1980.

IV-6 D-B Design Criteria Manual, Section: Historical Design Bases For The Plant Structures Designed By Bechtel Power Corporation (prior to January 1978), February 13, 1996.

IV-7 NRC Bulletin 2001-01, Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles, August 2001.

V. Electrical Equipment Design

1. A discussion of the effect of the power uprate on electrical equipment. For equipment that is bounded by the existing analyses of record, the discussion should cover the type of confirmatory information identified under Section II, above. For equipment that is not bounded by existing analyses of record, a detailed discussion should be included to identify and evaluate the changes related to the power uprate. Specifically, this discussion should address the following items:

   A. emergency diesel generators

   B. station blackout equipment

   C. environmental qualification of electrical equipment

   D. grid stability

RESPONSE to V

V. Electrical Equipment Design

V.1.A Emergency Diesel Generators

EDG System equipment capacity and capability for plant operation under MUR power uprate conditions are bound by the generator loading tables which are supported by the existing analysis of record. Both the bounding analysis and the generator loading tables demonstrate that the system has adequate capacity and capability to provide onsite standby power sources for safety-related loads on loss of offsite power with or without a concurrent accident.

V.1.B Station Blackout Equipment

SBODG System equipment capacity and capability for plant operation under MUR power uprate conditions are bound by the existing analysis of record for the EDG. The SBODG has a higher rating than the EDG. Therefore, the electrical design of the SBODG is not impacted by the MUR power uprate. No additional station load is required under SBO conditions due to the MUR power uprate.

V.1.C Environmental Qualification of Electrical Equipment

The term environmental qualification (EQ) applies to equipment important-to-safety to assure this equipment remains functional during and following design basis events. The MUR power uprate does not change the accident/post-accident temperature profiles inside containment; therefore, there is no impact on environmentally qualified electrical equipment as a result of the MUR power uprate.
All environmental qualification analyses considered operating at 102% of 2772 MWt, so that existing calculated temperature and pressure profiles bound the MUR uprate.

V.1.D Grid Stability

A load flow and stability analysis was performed by General Electric Energy Services in 2000 (Reference V-1) to evaluate the effects of a 10% increase in gross power output at DB. Even at 3 times the currently proposed increase, this study found that all machines maintained stability for all fault cases which have a reasonable probability of occurring.

In addition, a stability analysis of the entire FirstEnergy West (ECAR) system was performed by General Electric Energy Services in 2005, and a similar conclusion was reached. Although this study considered DB generation at its current level of output, it found "No stability issues for single-element (Category B) contingencies", and for those Category C contingencies studied, "No system-wide stability issues and no unit tripping were observed..."

V.1.E Station Auxiliary Electric Power Distribution System

Station Auxiliary Electric Power Distribution System equipment capacity and capability for plant operation under MUR power uprate conditions are bounded by the existing analysis of record. This analysis demonstrates that the system has adequate capacity and capability to operate the plant equipment.

V. References:

V-1 Enclosure 2 to FENOC Letter to NRC dated February 27, 2002 (NRC Adams Accession Number ML020640288).

VI. System Design

1. A discussion of the effect of the power uprate on major plant systems. For systems that are bounded by existing analyses of record, the discussion should cover the type of confirmatory information identified under Section II, above. For systems that are not bounded by existing analyses of record, a detailed discussion should be included to identify and evaluate the changes related to the power uprate. Specifically, this discussion should address the following systems:

   A. NSSS interface systems for pressurized-water reactors (PWRs) (e.g., main steam, steam dump, condensate, feedwater, auxiliary/emergency feedwater) or boiling-water reactors (BWRs) (e.g., suppression pool cooling), as applicable

   B. containment systems

   C. safety-related cooling water systems

   D. spent fuel pool storage and cooling systems

   E. radioactive waste systems

   F. Engineered safety features (ESF) heating, ventilation, and air conditioning systems

RESPONSE to VI

VI. System Design

A comparison between operating requirements for the 2817 MWt MUR conditions generated by the PEPSE heat balance and the 2772 MWt heat balance conditions demonstrates that the major plant systems that meet the requirements identified in Section VI above have sufficient design and operational margin to accommodate the MUR uprate. The MUR power uprate conditions remain bounded by the design basis of the Davis-Besse UFSAR (Reference VI-1).

The NRC record of review is contained in the Safety Evaluation Report by the Office of Nuclear Reactor Regulation United States Nuclear Regulatory Commission Davis-Besse Nuclear Power Station Unit 1 Docket No. 50-346, December 1976 (Reference VI-2).

A review of the Updated Final Safety Analysis Report (UFSAR) Chapter 15 accidents was performed to determine if the analyses of record for Davis-Besse remained applicable and bounding for the power uprate. These evaluations are provided in Section II above. Changes to technical specifications, protective and emergency system settings are described in Section VIII below. The individual systems that meet the requirements identified in Section VI above are discussed below.

VI.1.A NSSS Interface Systems

VI.1.A.1. Main Steam System

The main steam piping meets the requirements of ASME Section III for Nuclear Class 2 piping from the steam generators through the containment penetrations up to and including the isolation valves. Downstream of the isolation valves, the piping meets the requirements of ANSI B31.1.0 and is upgraded for inspection and documentation requirements. Each 600-pound ANSI rated main steam isolation valve serves to isolate its respective steam generator from the main steam line upon a steam and feedwater rupture control system signal. The valves are designed in accordance with the requirements of Section III, Nuclear Power Plant Components of the ASME Boiler and Pressure Vessel Code. The valves are designed to fail closed. Upon closure due to a downstream break, the valves are designed for zero leakage. Upon a steam line break upstream (reversed flow), the affected valve is signaled to close. Main steam non-return valves serve to close in the event of reversed flow. They are designed, manufactured, examined, tested, and inspected in accordance with ANSI B31.1.0. The steam generators' spring-loaded safety valves discharge to the atmosphere and are in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, 1968. The spring-loaded safety valves are capable of handling the Nuclear Steam Supply System capacity at 2834 MW thermal rating. The Main Steam System analysis of record is not impacted by the MUR power uprate.

VI.1.A.2. Steam Dump

The Davis-Besse equivalent of a steam dump system includes the Atmospheric Vent Valves (AVVs) and Turbine Bypass Valves (TBVs).

VI.1.A.2.a Atmospheric Vent Valves

An AVV is located in each Main Steam Line, upstream of the MSIVs (ICS11B in Loop 1 and ICS11A in Loop 2). The valve function is to provide a controlled path for venting of main steam to the atmosphere. These valves were evaluated for power uprate impact on three functions: (1) close to isolate containment, (2) open and modulate to relieve steam to the atmosphere, and (3) maintain pressure boundary to transport steam to safety and non-safety related loads. There are no changes in function. Power uprate conditions are bounded by existing design. The evaluation concludes the functional performance requirements of the Main Steam AVVs will be unaffected by the power uprate.

VI.1.A.2.b Turbine Bypass Valves

The Turbine Bypass Valves SP13B1, SP13B2, SP13B3 (Loop 1) and SP13A1, SP13A2, SP13A3 (Loop 2) are air-operated venturi cage and ball valves which actuate in response to a hand generated signal or an ICS generated signal. The valves primary function is to maintain stable turbine header pressure during load swing events. The flow rate is not being changed and the function of the TBVs is not being changed. For the power uprate, the ICS control will use the existing TBVs. Power uprate parameters are bounded by existing design conditions. There is no impact on the TBVs for the MUR power uprate.

The MUR power uprate conditions remain bounded by the design basis of the Davis-Besse UFSAR (Reference VI-1).

VI.1.A.3. Condensate System

The analysis of record for the Condensate System includes the design, materials, and details of construction of the feedwater heaters, which are in accordance with the ASME Code, Section VIII, Unfired Pressure Vessels, and the Heat Exchange Institute standards for open and closed Feedwater heaters. All piping meets the requirements of ANSI B31.1.0.

The Condensate System is not Safety Related and the existing analyses of record bounds accident/transient analyses for components/systems for plant operation at the proposed uprate power level. The Condensate System piping analysis for the 101.63% MUR power uprate conditions remain bounded by the design basis of record (Reference VI-1). The condensate storage tank volume calculation was revised in support of the proposed power uprate. The revised calculation used current standards and updated assumptions, resulting in a more conservative value for minimum usable tank volume. The change in volume from 250,000 gallons to 270,300 gallons will ensure that sufficient water is available to maintain the reactor coolant system at HOT STANDBY conditions for 13 hours with steam discharge to atmosphere and to cooldown the Reactor Coolant System to less than 280°F under normal conditions (i.e., no loss of offsite power) which is in accordance with the current licensing basis. Accordingly, this change will have no adverse effect on nuclear safety.

VI.1.A.4. Main Feedwater System

The design, materials, and details of construction of the feedwater heaters in the Main Feedwater System are in accordance with the ASME Code, Section VIII, Unfired Pressure Vessels, and Heat Exchange Institute standards for open and closed Feedwater heaters.

All piping meets the requirements of ANSI B31.1.0 except as follows:

a. All feedwater piping from the main feed pumps to the containment isolation valves is upgraded for critical service.
b. All piping into the containment structure from and including the isolation valves meets the requirements of ASME, Section III, Class 2 Piping.

The Main Feedwater System is not Safety Related and the existing analyses of record bounds accident/transient analyses for components/systems for plant operation at the proposed uprate power level. The Main Feedwater System piping analysis for the 101.63% MUR power uprate conditions remain bounded by the design basis of record (Reference VI-1).

VI.1.A.5. Auxiliary Feedwater System

The Auxiliary Feedwater System design basis of record is described in Section 9.2.7.1 and Table 9.0-1 of the UFSAR (Reference VI-1). The Auxiliary Feedwater System design basis of record is not affected by the proposed MUR power uprate.

VI.1.B Containment Systems

VI.1.B.1 Post LOCA Combustible Gas

The post LOCA combustible gas MUR power uprate conditions remain bounded by the design basis of the Davis-Besse UFSAR (Reference VI-1).

VI.1.C Safety-Related Cooling Water Systems

VI.1.C.1. Component Cooling Water

The Component Cooling Water System design basis of record is described in Section 9.2.2.1 and Table 9.0-1 of the UFSAR (Reference VI-1). The design basis of record is not affected by the proposed MUR power uprate.

The thermal load for the CCW system is bounded by the existing 102% instrument uncertainty and therefore bounds the 101.63% MUR power uprate. The uprate will however extend the normal cool down by 2 hours to 26 hours and will extend a single train cool down by 7 hours to 175 hours. D-B has increased the operating temperature limit from 120 °F to 130 °F providing additional operating margin during the summer months when the Service Water is at the maximum temperature.

VI.1.C.2. Service Water System / Ultimate Heat Sink (SWS/UHS)

The Service Water System is designed to serve two functions during station operation. The first function is to supply cooling water to the Component Cooling Water heat exchangers, the Containment Air Coolers, the ECCS Room Coolers, and the Cooling Water heat exchangers in the turbine building during normal operation. The second function is to provide, through automatic valve sequencing, a redundant supply path to the engineered safety features components during an emergency. Only one path, with one service water pump, is necessary to provide adequate cooling during this mode of operation.

The portion of the system required for emergency operation, including the intake structure, is designed to the ASME Code, Section III, Nuclear Class 3 and Seismic Class I, as applicable. This includes protection from a tornado and tornado missiles. The associated containment penetrations are Nuclear Class 2.

The post-LOCA containment analyses contained in UFSAR section 6.2 are based on a power level of 3025 MWt; therefore, they bound the MUR uprate. The NRC record of review is contained in the Safety Evaluation Report by the Office of Nuclear Reactor Regulation United States Nuclear Regulatory Commission Davis-Besse Nuclear Power Station Unit 1, Docket No. 50-346 December 1976 (Reference VI-2).

VI.1.D Spent Fuel Pool Storage and Cooling Systems

The Spent Fuel Cooling system is bound by existing analyses of record.

VI.1.E Radioactive Waste Systems

The systems handling radioactive wastes are designed such that the estimated releases in effluents comply with the following requirements of 10CFR20 and 10CFR50:

a. The individual radionuclide concentrations in liquid effluents at the site boundary shall not exceed the limits for releases to unrestricted areas given in Appendix B of 10CFR20.

b. The releases of radioactivity from the station shall comply with the "as low as reasonably achievable" standard set forth in 10CFR50.

Davis-Besse is in compliance with the regulatory position of Regulatory Guide 4.15 (Revision 1, February 1979).

The power uprate will not affect the ability of the Liquid Radwaste and Solid Radwaste systems to collect, store, process, or dispose of liquid and solid radwaste generated by the station. System functions will not be impacted by the power uprate.

The power uprate may result in some additional liquid waste production, primarily from processing additional reactor coolant waste due to the higher boron concentration at the beginning of core life. This additional production should be minimal and will not impact the ability of the system to function as designed and currently operated.

Some additional increase in spent resin processing and contaminated solid materials may occur over core life. This increase will not challenge the solid waste processing systems ability to perform as designed.

VI.1.F Engineered Safety Features Heating, Ventilation and Air Conditioning Systems

The ESF Heating, Ventilation, and Air Conditioning Systems remain bounded by the design basis (102% of 2772 MWt) of the Davis-Besse UFSAR (Reference VI-1) for MUR power uprate conditions. There are no expected changes in containment cooling operation at the MUR uprate power level. The containment accident analysis has been performed at a bounding power level with the containment air coolers and fan flow rates and found acceptable. The containment cooling system has adequate margin to cool the containment at MUR conditions.

VI. References:

VI-1 Davis-Besse Updated Final Safety Analysis Report, Rev. 25, June 2006.
VI-2 NUREG-0136, Safety Evaluation Report by the Office of Nuclear Reactor Regulation United States Nuclear Regulatory Commission Davis-Besse Nuclear Power Station Unit 1 Docket No. 50-346, December1976.

VII. Other

1. A statement confirming that the licensee has identified and evaluated operator actions that are sensitive to the power uprate, including any effects of the power uprate on the time available for operator actions.

2. A statement confirming that the licensee has identified all modifications associated with the proposed power uprate, with respect to the following aspects of plant operations that are necessary to ensure that changes in operator actions do not adversely affect defense in depth or safety margins:

   A. emergency and abnormal operating procedures

   B. control room controls, displays (including the safety parameter display system) and alarms

   C. the control room plant reference simulator

   D. the operator training program

3. A statement confirming licensee intent to complete the modifications identified in Item 2 above (including the training of operators), prior to implementation of the power uprate.

4. A statement confirming licensee intent to revise existing plant operating procedures related to temporary operation above "full steady-state licensed power levels" to reduce the magnitude of the allowed deviation from the licensed power level. The magnitude should be reduced from the pre-power uprate value of 2 percent to a lower value corresponding to the uncertainty in power level credited by the proposed power uprate application.

5. A discussion of the 10 CFR 51.22 criteria for categorical exclusion for environmental review including:

   A. A discussion of the effect of the power uprate on the types or amounts of any effluents that may be released offsite and whether or not this effect is bounded by the final environmental statement and previous Environmental Assessments for the plant.

   B. A discussion of the effect of the power uprate on individual or cumulative occupational radiation exposure.

RESPONSE to VII

VII.1 Operator actions that are sensitive to the power uprate, including any effects of the time available for operator actions have been reviewed. Based upon the review performed, the Operator Actions required to support the MUR uprate are bounded and supported by the current analysis.

It is therefore concluded that the power uprate requires no additional operator actions, that the additional time required to perform certain operator actions will have no adverse effects and that the time available for critical operator action has not been reduced.

VII.2.A Emergency and abnormal operating procedures have been reviewed for potential impact on the proposed power uprate. There is no adverse impact on these procedures with this power uprate. Additionally, the report concludes this power uprate is within safety margins.

VII.2.B The following changes/modifications are associated with implementation of the power uprate that affect control room controls:

- OTSG levels will be slightly raised but are within the ability of the Integrated Control System (ICS) to maintain when in full automatic control, thus having no adverse effect on defense in depth or safety margins.
- Power Production Heat Transfer will be increased slightly but within the ability of the operator to maintain prescribed parameters less than the required limits, thus having no adverse effect on defense in depth or safety margins.
- Cooldown time on Decay Heat is increased slightly, which increases the amount of time the operators must control the cooldown of the RCS. The increase in the amount of time Decay Heat is controlled is within the ability of the operators thus, having no adverse effect on defense in depth or safety margins.
- Setpoint changes made in the Reactor Protection System establish trip parameters to ensure safety margins are maintained during normal and transient plant operations. The setpoint changes will not impact the operator's ability to control the plant thus having no adverse effect on defense in depth or safety margins.
- Changes to be made to the calibration of the nuclear instrumentation will not change the performance of the Integrated Control System (ICS) due the range of the instruments remaining the same whether maintaining 2772 MWt or 2817 MWt. The change in nuclear instrumentation calibration will have no effects on any control room controls or the operator's ability to monitor core power production, thus having no adverse effect on defense in depth or safety margins.
- Required changes to the settings of the ICS modules are associated with maintaining the plant within normal operating parameters, thus having no adverse affect on defense in depth or safety margins.

The following modifications are associated with implementation of the power uprate that affect operator displays (including Safety Parameter Display System):

- The Incore Monitoring System safety functions and other functions may require changes to the plant computer software. The software changes will be transparent to the operators; their response to abnormal indications by the software will remain unchanged, thus having no adverse affect on defense in depth or safety margins.

- SPDS displays have been created to trend the performance of the following:
  LEFM Feedwater flows and Venturi Feedwater flows versus time.
  LEFM Feedwater temperature and RTD Feedwater temperatures versus time.
  Velocity Profile of LEFM Meter 1 and Meter 2 versus time.

These trends enable the operator to identify changes in performance of the LEFM inputs to the Heat Balance calculation, thus having no adverse affect on defense in depth or safety margins.

The following modifications are associated with implementation of the power uprate that affect alarms:

- Feedwater Heater Level optimization (an enhancement currently planned for 15RFO) may result in the changing of various Feedwater Heater level alarm and trip setpoints. The changing of the alarm and trip setpoints should improve the performance and efficiency of the secondary plant, and operator response to the alarms will remain unchanged, thus having no adverse affect on defense in depth or safety margins. This is not a restraint to implementation of the MUR power uprate.
- An annunciator alarm 10-4-A, MFW FLOW CALDON SYS TRBL, has been added to alert the operators when the LEFM system has self diagnosed a condition that has resulted in an internal alert or failure. The associated alarm procedure will then direct the operator to follow the TRM Limiting Condition for Operation for the LEFM (3.3.4.1). This action is to ensure the plant is operated within designed safety margins and thus does not adversely affect defense in depth or safety margins.

VII.2.C The Control Room plant reference simulator needed to be modified due to this uprate. While there was minimal impact on plant response due to this uprate, changes were made that needed to be properly modeled on the simulator. For example, the decay heat generated by the core model was modified so that the longer time to cool from 280 °F to 140 °F is properly reflected. The simulator modifications have been completed.

VII.2.D The Operator Training program will need to be modified due to this uprate. While there will be minimal impact due to this uprate, changes are being made that the Operator will need to be properly trained on. For example, the additional two hours needed to cool down from 280°F to 140°F.

VII.3 FENOC will complete any modifications [except feedwater heater level optimization, an enhancement currently planned for 15RFO] identified in Item VII.2 (including the training of operators), prior to implementation of the power uprate.

VII.4 FENOC will revise existing plant operating procedures related to temporary operation above "full steady-state licensed power levels" to reduce the magnitude of the allowed deviation from the licensed power level. The magnitude shall be reduced from the pre-power uprate value of 2 percent to 0.37%, the value corresponding to the uncertainty in power level credited by this proposed power uprate application.

VII.5 10 CFR 51.22 Discussion

VII.5.A. 10 CFR 51.22(c)(9) provides criteria for, and identification of, licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (A.1) involve a significant hazards consideration, (A.2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (B) result in a significant increase in individual or cumulative occupational radiation exposure.

It has been determined that this license amendment request meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the proposed license amendment. The basis for this determination is as follows:

1. The proposed license amendment does not involve a significant hazards consideration as described previously in the no significant hazards evaluation (Enclosure 1 Section 5.1) for this License Amendment Request (LAR).

2. The proposed changes will allow Davis-Besse (D-B) to operate at an uprated power level of 2817 Megawatts Thermal (MWt). This represents an increase of approximately 1.63 percent over the current 100 percent power level of 2772 MWt.

   The proposed changes do not significantly impact installed equipment performance or require significant changes in system operation. Changes in maintenance and operational practices will not impact the release of solid, liquid or gaseous effluents. The specific activity of the primary and secondary coolant is expected to increase by no more than the percentage increase in power level. Therefore, the amount and specific activity of solid waste is not expected to change significantly.

   Gaseous and liquid effluent releases are expected to increase from current values by no more than the percentage increase in power level. Offsite release concentrations and doses will continue to be maintained within the limits of 10 CFR 20 and 10 CFR 50 Appendix I in accordance with the requirements of the D-B Offsite Dose Calculation Manual (ODCM). The ODCM contains offsite dose calculation methodologies, the radioactive effluent controls program, and radiological environmental monitoring activities. The ODCM contains the methodologies and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, the methodologies and parameters used in the calculation of gaseous and liquid effluent monitoring alarm and trip setpoints, and the controls for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable in accordance with 10 CFR 50.36a. The proposed changes will not result in changes in the operation or design of the gaseous, liquid or solid waste systems, and will not create any new or different radiological release pathways.

Therefore, the proposed license amendment will not result in a significant change in the types or increase in the amounts of any effluents that may be released off-site.

VII.5.B. The proposed changes will not cause radiological exposure in excess of the dose criteria for restricted and unrestricted access specified in 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 D-B Radiation Protection Program. Therefore, the proposed license amendment will not result in a significant increase to the individual or cumulative occupational radiation exposure.

VIII. Changes to technical specifications, protection system settings, and emergency system settings

1. A detailed discussion of each change to the plant's technical specifications, protection system settings, and/or emergency system settings needed to support the power uprate:

   A. a description of the change

   B. identification of analyses affected by and/or supporting the change

   C. justification for the change, including the type of information discussed in Section III, above, for any analyses that support and/or are affected by change

RESPONSE to VIII

VIII.1. No changes are required to Technical Specifications, protection system settings or emergency system settings as a part of the MUR uprate for the ATWS or DSS systems.

VIII.2. For RPS, setpoint changes are required for the Technical Specifications as a part of the MUR Uprate.

VIII.2.A. Reactor Protection System setpoint changes are required for:

- High Flux (Overpower trip) changes from 105.1%RTP to 104.9% RTP if the NIs are verified by a heat balance using the LEFM.

- A note is added to Table 2.2-1 to state that when implementing a licensee-controlled requirement due to the Ultrasonic Flow Meter instrumentation being inoperable or not used in the performance of the daily heat balance, the high flux allowable value will be reduced to $\leq$ 103.3% of RTP. Requirements regarding the Ultrasonic Flow Meter will be described in the UFSAR Technical Requirements Manual, a licensee-controlled document.

VIII.2.B. RPS Calculations impacted:

- High Flux (Overpower trip) (C-ICE-058.01-008 Reference VIII-3)

VIII.2.C. Justification for Change:

- High Flux (Overpower trip)
  The existing safety analysis models the high flux setpoint as a function of total power (MWt) not percent of RTP. In order to preserve the maximum allowed power level in MWt, the setpoint, expressed in terms of percent RTP must be reduced for the new uprated power level. The new rated core power level will be 2817 MWt. The new analysis value for the overpower reactor trip setpoint with 4 reactor coolant pumps operating becomes:

Analytical High Flux Setpoint = (112% * 2772 MWt) / (2817 MWt)
= 110.2% (rounded down for conservatism)

As a result of the analysis high flux setpoint and heat balance error being revised, the in-plant high flux technical specification Allowable Value (TSAV) must be changed. The heat balance error is reduced to a value of 0.37% from 2% RTP, Reference VIII-1. The total steady state and transient induced errors remain unchanged, i.e., 4% RTP. Similarly, the bistable error is not changed, 0.84% RTP. The new in-plant (allowable) high flux setpoint is:

TSAV High Flux Setpoint = 110.2% – 0.37% – 4.0% – 0.84%
= 104.9% RTP (rounded down for conservatism)

Note that the Allowable Value for the High Flux setpoint is derived using Method 1 as described in Section 7.3 of ISA RP67.04.02-2000, "Methodologies of the Determination of Setpoints for Nuclear Safety-Related Instrumentation." Uncertainties that are random, normally distributed, and independent are combined by the square-root-sum-of-squares (SSRS) method. Uncertainties that are not random, not normally distributed, or are dependent are combined algebraically. The total uncertainty is then subtracted from the Analytical Limit to establish the Allowable Value. This is further described in Section 7.4 of "Safety Criteria and Methodology for Acceptable Cycle Reload Analysis," AREVA NP Topical Report BAW-10179P-AB. Following this methodology, the trip setpoint will assure a reactor trip 95% of the time at a 95% confidence level.

If the LEFM system cannot be used for the feedwater input to the heat balance, a heat balance using the installed feedwater venturis will be performed. The required actions will be specified in the UFSAR TRM. Whenever power range nuclear instrumentation verification to a heat balance using the venturis is required, power will be reduced to 98.4% RTP (or ≤ 2772 MWt). This preserves the core power level used in the accident analyses. When operating with four reactor coolant pumps at this reduced power, the high flux trip allowable value setpoint will be reduced to 103.3% RTP (or ≤ 2910 MWt). This ensures that when the difference in heat balance error between the LEFM (0.37%) and the main feedwater venturis (2%) is included, the maximum analytical setpoint value of 110.2% RTP will not be exceeded.

TSAV High Flux Setpoint = 110.2% – 2% – 4.0% – 0.84%
= 103.3% RTP (rounded down for conservatism)

VIII.3. Recommended Changes to Plant Technical Specifications and Bases

	Table VIII.3-1 Description of Technical Specification and Bases Changes
Change No.	Change Description
1	**Facility Operating License, Paragraph 2.C. (1)** FENOC is authorized to operate the facility at steady state reactor core power levels not in excess of **2817** megawatts (thermal).
2	**TS Definitions, Item 1.3** RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of **2817** MWt.
3	**TS Section 2.2, Pages 2-5 and 2-6, Table 2.2-1** 2. High Flux ≤ **104.9**% of RATED THERMAL POWER with four pumps operating *# Add Note: # **≤ 103.3% of RATED THERMAL POWER, to be implemented in accordance with licensee controlled requirements for Ultrasonic Flow Meter instrumentation that is inoperable or not used in the performance of the daily heat balance.**
4	**TS B 2.0 Safety Limit Bases, Section 2.1.1 AND 2.1.2, Page B 2-1** "....the maximum possible thermal power of **110.2% of 2817** MWt when reactor..."
5	**TS B 2.0 Safety Limit Bases, Section 2.2.1, second paragraph under "High Flux", Page B 2-4** During normal station operation **except as noted below,** a reactor trip is initiated when the reactor power level reaches the Allowable Value of **≤104.9%** 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,** which was used in the safety analysis.

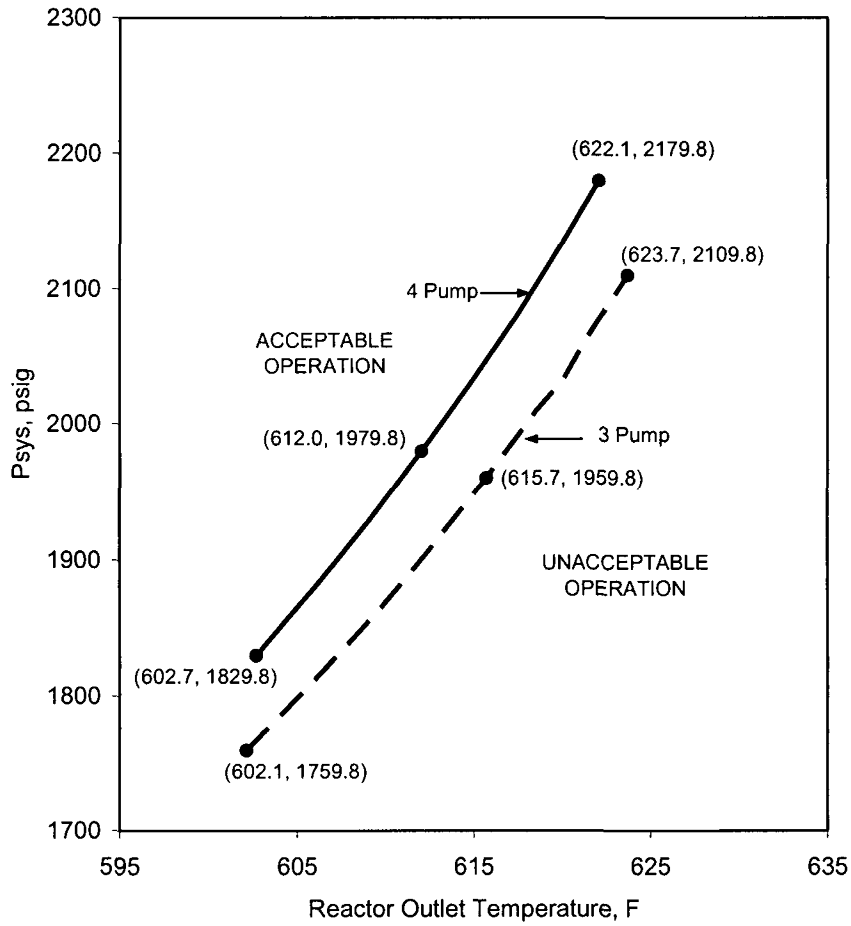
	Table VIII.3-1 Description of Technical Specification and Bases Changes
Change No.	Change Description
	Add as third paragraph: **Requirements regarding the Ultrasonic Flow Meter instrumentation are described in the Technical Requirements Manual (TRM). The TRM includes a required action to adjust the High Flux setpoints in the event the Ultrasonic Flow Meter instrumentation is inoperable or not used in the performance of the daily heat balance. The TRM does not specify the actual Allowable Value; it refers to TS 2.2.1. Under the TRM required action, the Allowable Value is ≤ 103.3% of RATED THERMAL POWER (2910 MWt) when operating with four reactor coolant pumps, thereby maintaining the same level of protection of the analytical limit. No changes to the TS high flux trip setpoints for reduced pump operation are required; the setpoint provided is conservative and bounds operation irrespective of the use of the Ultrasonic Flow Meter for heat balance calculation.**
6	**TS B 2.0 Safety Limit Bases, Section 2.2, Page B 2-8** Bases Figure 2.1 is replaced. The revised Bases Figure 2.1 is contained below this table.
7	**TS 6.9.1.7, Administrative Controls – Core Operating Limits Report** Insert as third paragraph the following: **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, an actual value of 100.37% of RATED THERMAL POWER may be used when the input for reactor thermal power measurement of feedwater mass flow and temperature is from the Ultrasonic Flow Meter. The following NRC approved documents are applicable to the use of the Ultrasonic Flow Meter with a 0.37% measurement uncertainty:** **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.** **Caldon Inc. Engineering Report-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM√™ or LEFM CheckPlus™ System," Revision 5, dated October, 2001.**

	Table VIII.3-1 Description of Technical Specification and Bases Changes
Change No.	Change Description
8	**TS 3.7.1.3** The condensate storage tanks shall be OPERABLE with a minimum **usable** volume of **270,300** gallons of water. **SR 4.7.1.3.1** The condensate storage tanks shall be demonstrated OPERABLE at least once per 12 hours by verifying the **usable** water volume to be within its limits when the tanks are the supply source for the auxiliary feedwater pumps.
9	**TS 3/4.3.1 Reactor Protection System Instrumentation Table 4.3-1** Add a reference to Note 10 to Functional Unit 2, High Flux, Channel Calibration in Table 4.3-1. Note 10 currently reads: (10) - **If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.** **The instrument channel setpoint shall be reset to a value that is within the as-left tolerance of the Limiting Trip Setpoint, or a value that is more conservative than the Limiting Trip Setpoint; otherwise, the channel shall be declared inoperable. The Limiting Trip Setpoint and the methodology used to determine the Limiting Trip Setpoint, the predefined as-found acceptance criteria band, and the as-left setpoint tolerance band are specified in a document incorporated by reference into the Updated Final Safety Analysis Report.**

	Table VIII.3-1 Description of Technical Specification and Bases Changes
Change No.	Change Description
10	**TS B 3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM AND SAFETY SYSTEM INSTRUMENTATION** **Page B 3/4 3-3, second paragraph** For the RPS, SFAS Table 3.3-4 Functional Unit Instrument Strings b, c, d, e, and f, and Interlock Channel a, and SFRCS Table 3.3-12 Functional Unit 2: [Note: Setpoint methodology for RPS Functional **Units 2 and 7 are** discussed on page B 3/4 3-4.] **Page B 3/4 3-4, first paragraph** For RPS Functional **Units 2 and** 7, the Limiting Trip Setpoint is specified in the UFSAR Technical Requirements Manual. The Limiting Trip Setpoint is based on the calculated total loop uncertainty per the plant-specific methodology identified below. The Limiting Trip Setpoint may be established using Method 1 or Method 2 from Section 7 of ISA RP67.04.02-2000, *"Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation."* Additional information is contained in the Technical Requirements Manual. **Page 3/4 3-4, second paragraph** The purpose of a Limiting Safety System Setting is to ensure that protective action is initiated before the process conditions reach the analytical limit, thereby limiting the consequences of a design-basis event to those predicted by the safety analyses. For RPS Functional **Units 2 and** 7, the Limiting Trip Setpoint is the Limiting Safety System Setting required by 10 CFR 50.36. This definition of the LSSS is consistent with the guidance issued to the industry through correspondence with NEI (Reference NRC-NEI Letter dated September 7, 2005). The definition of LSSS values continues to be discussed between the industry and the NRC, and further modifications to these TS Bases will be implemented as guidance is provided. **Page 3/4 3-4, third paragraph** TS Table 4.3-1 Note 10 ensures that unexpected as-found conditions are evaluated prior to returning the channel to service, and that as-left settings provide sufficient margin for uncertainties. Specifically, for RPS Functional **Units 2 and** 7, the following additional requirements are added by TS Table 4.3-1 Note 10:

	Table VIII.3-1 Description of Technical Specification and Bases Changes
Change No.	Change Description
11	**TS B 3/4.7.1.3 CONDENSATE STORAGE TANKS** The OPERABILITY of the Condensate Storage Tanks with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 13 hours with steam discharge to atmosphere and to cooldown the Reactor Coolant System to less than 280°F under normal conditions (i.e., no loss of offsite power). The water volume limit **does not** include water **that is** not usable because of tank discharge line location or other physical characteristics.

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



Pumps	Flow, gpm	2772 MWt Power	2817 MWt Power	Required Measured Flow to Ensure Compliance, gpm
4	380,000	112%	110.2%	389,500
3	283,860	90.1%	89.4%	290,957

The following information provides the supporting justification for the proposed changes described above for the plant Technical Specifications.

For Changes No. 1 and. 2:

The current RTP for Davis-Besse is 2772 MWt. For the MUR power uprate, the increase in power will be 1.63 percent (Reference VIII-1). This is based on a plant specific evaluation of reactor power measurement uncertainty using LEFM CheckPlus™ instrumentation versus the previous mandated 2-percent uncertainty that was formerly required by 10CFR50, Appendix K. Therefore, the new RTP will be:

RTP = 2772 MWt * 1.0163 = 2817 MWt (rounded down for conservatism)

For Change No. 3, 4, 5, and 7:

The existing safety analysis models the nuclear over power setpoint as a function of total power (MWt) not percent of RTP. In order to preserve the maximum allowed power level in MWt, the setpoint, expressed in terms of percent RTP must be reduced for the new uprated power level. The new rated core power level will be 2817 MWt. The new analysis value for the overpower reactor trip setpoint with 4 reactor coolant pumps operating becomes:

Overpower Setpoint = (112% * 2772 MWt) / (2817 MWt)
= 110.2% (rounded down for conservatism)

As a result of the analysis overpower setpoint and heat balance error being revised, the in-plant high flux technical specification Allowable Value must be changed. The heat balance error is reduced to a value of 0.37% from 2% RTP, Reference VIII-1. The total steady state and transient induced errors remain unchanged, i.e., 4% RTP. Similarly, the bistable error is not changed, 0.84% RTP. The new in-plant (allowable) high flux setpoint is:

High Flux Setpoint = 110.2% – 0.37% – 4.0% – 0.84%
= 104.9 % RTP (rounded down for conservatism)

If the LEFM system cannot be used for the feedwater input to the heat balance, a heat balance using the installed feedwater venturis will be performed. The required actions will be specified in the UFSAR TRM. Whenever power range nuclear instrumentation verification to a heat balance using the venturis is required, power will be reduced to 98.4% RTP (or ≤ 2772 MWt). This preserves the core power level used in the UFSAR accident analyses and the initial conditions for DNB as required by the LCO Limits in the COLR with the larger heat balance uncertainty. When operating with four reactor coolant pumps at this reduced power, the high flux trip allowable value setpoint will be reduced to 103.3% RTP (or ≤ 2910 MWt). This ensures that when the difference in heat balance error between the LEFM (0.37%) and the main feedwater venturis (2%) is included, the maximum analytical setpoint value of 110.2% RTP will not be exceeded.

TSAV High Flux Setpoint = 110.2% – 2% – 4.0% – 0.84%

= 103.3% RTP (rounded down for conservatism)

Similar to 4-pump operation, core power must also be reduced to 73.8% RTP (or $\leq$ 75% of 2772 MWt) if the plant is operating with three reactor coolant pumps and the LEFM cannot be used for the heat balance. This preserves the maximum core power for initial condition DNB as required by the LCO Limits in the COLR.

For Change No. 6

Table data for 2817 MWt is added.

For Change No. 8 and 11

Analysis was performed at 102% of 2772 MWt which bounds the MUR uprate and instrument uncertainty to maintain the technical specification bases of maintaining hot standby conditions for 13 hours and cooling down to 280°F in the next 6 hours. The volume was revised to identify the limit based on useable volume.

For Change No. 9 and 10

The proposed change to Functional Unit 2, High Flux, in Table 4.3-1 is the addition of a reference to Note 10. This change provides consistency with the change proposed to Table 2.2-1 and implements NRC guidance on instrument setpoint methodology as established in DBNPS license amendment 274 (Reference VIII-2). Thus, this change will have no adverse effect on nuclear safety.

VIII.4. Recommended Changes to Plant Technical Requirements Manual (TRM)

	Table VIII.4-1 Recommended Changes to Plant Technical Requirements Manual (TRM)
Change No.	Change Description
1	**Page 1-3 Definitions** RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of **2817** MWt.
2	Add new Item for the Ultrasonic Flow Meter Instrumentation and Bases. (See following pages)

The following information provides the supporting justification for the proposed change described above to the plant TRM.

For TRM Change No. 1:

The current RTP for Davis-Besse is 2772 MWt. For the MUR power uprate, the increase in power will be 1.63 percent (Reference VIII-1) in Technical Specification 1.3. This is based on a plant specific evaluation of reactor power measurement uncertainty using LEFM CheckPlus™ instrumentation versus the previous mandated 2-percent uncertainty that was formerly required by 10CFR50, Appendix K. Therefore, the new RTP will be:

RTP = 2772 MWt * 1.0163 = 2817 MWt (rounded down for conservatism)

3/4.3 INSTRUMENTATION

3/4.3.4 ULTRASONIC FLOW METER INSTRUMENTATION

LCO 3.3.4.1 The Ultrasonic Flow Meter instrumentation shall be OPERABLE and used in the performance of the daily heat balance measurement required by Functional Unit 2, "High Flux," of Technical Specification Table 4.3-1, "Reactor Protection System Instrumentation Surveillance Requirements".

APPLICABILITY: **MODE 1, when greater than 50% RATED THERMAL POWER**

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Ultrasonic Flow Meter instrumentation inoperable or not used in the performance of the daily heat balance.	A.1 Reduce thermal power to ≤ 98.4% of RATED THERMAL POWER with four reactor coolant pumps operating or ≤73.8% of RATED THERMAL POWER with three reactor coolant pumps operating. AND	Prior to the completion of the next required daily heat balance measurement
	A.2 Reduce the setpoint for the Reactor Protection System High Flux (Technical Specification Table 2.2-1 Reactor Protection System Instrumentation Trip Setpoints – Functional Unit 2) in accordance with Technical Specification 2.2.1 for operation with the daily heat balance measurement not using the Ultrasonic Flow Meter.	Within 10 hours after the next required daily heat balance measurement

SURVEILLANCE REQUIREMENT

SURVEILLANCE		FREQUENCY
4.3.4.1	The Ultrasonic Flow Meter instrumentation shall be demonstrated OPERABLE by performance of a CHANNEL CHECK.	24 hours

3/4.3 INSTRUMENTATION

BASES

3/4.3.4.1 Ultrasonic Flow Meter Instrumentation

Due to its higher accuracy, the use of OPERABLE Ultrasonic Flow Meter (Leading Edge Flow Meter (LEFM) CheckPlus™ System) instrumentation is preferred for the performance of daily heat balance calculations required by Technical Specification (TS) Surveillance Requirement (SR) 4.3.1.1.1 (Table 4.3-1, Functional Unit 2 - Reactor Protection System High Flux). The use of the LEFM instrumentation for the secondary-side feedwater flow and feedwater temperature inputs into the heat balance calculation provides an uncertainty of 0.37% above 50% of RATED THERMAL POWER (RTP). An uncertainty of 2% is assumed when non-LEFM instrumentation is used for the secondary-side feedwater flow and feedwater temperature inputs into the heat balance calculation. Below 50% of RTP, the heat balance is performed using primary-side instrumentation. Hence, this LCO is only applicable above 50% RTP. In addition, below 75% of RTP, the safety analyses have adequate margin to accommodate a 2% heat balance error either with or without the LEFM being used to perform the daily heat balance calculation.

If the LEFM is not available for use, the heat balance will be performed using inputs from less accurate installed instrumentation. Continued power operation is allowed; however, THERMAL POWER must be limited to ≤98.4% of RTP with four reactor coolant pumps operating, or ≤73.8% of RTP (75% of 2772 MWt) with three reactor coolant pumps operating. Given the larger heat balance uncertainty, these limits preserve the core power used in the UFSAR accident analysis and the initial conditions for DNB as required by the regulating group operating limits in the COLR.

Also, when operating with four reactor coolant pumps at the reduced power, the Reactor Protection System High Flux trip setpoint Allowable Value must be reduced from ≤104.9% to ≤103.3% within ten hours of completion of the heat balance calculation using the less accurate instrumentation, in accordance with the requirements of TS 2.2.1. This reduction ensures that when the increased uncertainty of the instrumentation is considered, the maximum analytical setpoint value of 110.2% will not be exceeded.

Historical comparison of the two feedwater flow measurement systems used for secondary-side heat balance calculations above 50% RTP, LEFM-based and feedwater venturi-based, indicates that the two methods do not diverge significantly during power operations over short periods. The long-term fouling of the venturis results in a more conservative feedwater flow input to the heat balance calculation. Nuclear Instrumentation trend analysis indicates that the NI to heat balance comparison will not drift significantly over a three-week period, and surveillance data indicates essentially no drift of the high flux setpoints. Accordingly, the accuracy and conservatism of the RPS high flux trip is acceptable in the ten hour period provided for setpoint reduction after completion of the non-LEFM-based heat balance calculation.

The LEFM includes a flow meter measurement section in each of the two main feedwater flow headers. Each measurement section consists of sixteen ultrasonic transducers. With any transducer inoperable, the Ultrasonic Flow Meter instrumentation system is considered inoperable and the required actions are to be applied.

The daily CHANNEL CHECK utilizes the on-line verification and self-diagnostic features of the LEFM to ensure the instrumentation is performing as designed.

VIII.5 Recommended Changes to Davis-Besse Updated Final Safety Analysis Report

Recommended changes to the D-B UFSAR are included in Attachment B to AREVA NP Document 51-9004090. These changes are incorporated via 10CFR50.59.

VIII.6 NSSS Operating Conditions

A calculation was prepared to determine the NSSS design operating conditions at the uprated power conditions. The results from Reference VIII-4 were used as input for most of the subsequent tasks.

VIII.6.1 Purpose

The purpose of the NSSS operating conditions calculation was to define the RCS and steam generator conditions for the D-B MUR power uprate. New operating conditions were defined for a bounding 1.63% power increase (1.7% used for conservatism) for both 0% and 20% tube plugging. Specific outputs of the calculations included:

- T_{hot}
- T_{cold}
- RCS Flow Rate (Mass and Volume Based)
- Steam Temperature
- Feedwater (Steam) Mass Flow Rate

In addition, the effects of the uprated conditions were evaluated for their impact on the following:

- Reactor Coolant System Design Conditions (RCS Functional Specification)
- Reactor Vessel
  - ✓ Design Values (RCS Functional Specification)
  - ✓ Core DNBR
  - ✓ Core Lift
  - ✓ RV Internals Flow Induced Vibration
- RCS Piping
  - ✓ Design values (RCS Functional Specification)
  - ✓ Surge and spray line thermal stratification conditions
  - ✓ Attached piping temperatures
- SG Performance
  - ✓ Design values (RCS Functional Specification)
  - ✓ Feedwater flow rates for Flow Induced Vibration (FIV) were defined.
  - ✓ Water Levels
- RCP Performance
  - ✓ Design values (RCS Functional Specification)
  - ✓ Head Capacity, NPSHr, Pump Power

- Pressurizer Performance
  - ✓ Design values (RCS Functional Specification)
  - ✓ Level Setpoint
  - ✓ Pressure Control
- Control Rod Drive Mechanisms
- End-of-Cycle T_{ave} Reduction

Note: 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. It does not address the Safety/LOCA analysis and fuel design aspects of tube plugging. D-B's tube plugging limit is currently 1300 tubes (8.4%) per steam generator.

VIII.6.2 Calculation Assumptions

VIII.6.2.1 Steam Pressure = 930 psia.

VIII.6.2.2 Feedwater Temperature = 455°F for both the current and uprated conditions (at both 0% and 20% tube plugging). The steam generator performance results are relatively insensitive to this value.

VIII.6.3 Calculation Inputs

VIII.6.3.1 Current Core Licensed Power of 2772 MWt, Reference VIII-5.

VIII.6.3.2 The total thermal power is currently 2789 MWt. The contributors to the total thermal power are core power plus other RCS power which consists of: RCS pump heat, the enthalpy difference between hot letdown flow and relatively cold makeup flow, and RCS ambient heat losses (exclusive of the OTSG). A value of 17 MWt has been used by Davis-Besse.

VIII.6.4 Calculation Results

The calculated values at the uprated power are presented in TABLE VIII.6-1. Calculations were performed for 0% and 20% tube plugging to bound the range of RCS temperatures and steam conditions (flow rate and temperature). Inputs to these calculated values were:

Core Thermal Power	=	2819 MWt	Feedwater Temperature	=	455°F
Other RCS Power	=	17 MWt	Steam Pressure	=	930 psia

TABLE VIII.6-1 – Davis-Besse Operating Conditions

	Current Operation	1.7% Uprate		Current Design Values
		No OTSG Tube Plugging	20% OTSG Tube Plugging	
Core Thermal Power (MWt)	2772	2819	2819	2772
Other RCS Power (MWt)(1)	17	17	17	17
Total Thermal Power (MWt)	2789	2836	2836	2789
T_{hot} (°F)	606.1	606.5	607.4	607.5
T_{cold} (°F)	557.9	557.5	556.6	556.5
T_{ave} (°F)	582	582	582	582
RCS Mass Flow Rate (kpph)	146,020	146,077	140,752	137,900
RCS Volumetric Flow Rate (gpm)	393,060	392,990	378,240	369,600
Steam Temperature (°F)	596.2	596.1	584.3	600.0(2)
Terminal Temperature Difference ($T_{hot} - T_{steam}$)	9.9	10.4	23.1	NA
Feedwater/Steam Flow Rate (kpph)	11,650	11,840	11,990	12,240
Steam Pressure (Input) (psia)	930	930	930	1065(2)
Feedwater Temperature (Input) (°F)	455	455	455	470

Notes:

(1) Other RCS Power corresponds to RCP heat less makeup/letdown heat loss and ambient heat loss.
(2) The design steam temperature and pressure values shown are ASME design values rather than maximum design transient values.

VIII. References:

VIII-1 AREVA NP Calculation. 32-5012428-08, Davis-Besse Heat Balance Uncertainty, April 2007 (Enclosure 3).

VIII-2 NRC Letter to FirstEnergy Nuclear Operating Company, "Davis-Besse Nuclear Power Station, Unit 1 – Issuance of Amendment Regarding Framatome Mark B-HTP Fuel Design for Cycle 15 (TAC No. MC6888)," license amendment 274, dated March 2, 2006.

VIII-3 D-B Calculation C-ICE-58.01-008, Rev 3, RPS RX Power Related Field Trip Setpoints

VIII-4 AREVA NP Calculation 32-5011757-001, DB App. K Power Uprate – New Operating Conditions.

VIII-5 Davis-Besse Updated Final Safety Analysis Report, Rev. 25, June 2006.