July 31, 2008

Mr. Barry S. Allen
Site Vice President
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
Mail Stop A-DB-3080
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SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1 – CORRECTION OF ADMINISTRATIVE ERRORS IN AMENDMENT RE: MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NO. MD8326)

Dear Mr. Allen:

On June 30, 2008, the U.S. Nuclear Regulatory Commission (NRC) issued Amendment No. 278 to Facility Operating License No. NPF-3 for the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS). The amendment modified the technical specifications (TSs) in response to your application dated April 12, 2007, as supplemented by letters dated September 18, October 8, October 19, 2007, and January 15 (2 letters), February 14, February 20, March 12, and May 16, 2008.

This amendment made the Operating License and TS changes necessary to allow an increase in the rated thermal power of approximately 1.63 percent, from 2772 megawatts thermal (MWt) to 2817 MWt, based on the use of the Caldon, Inc., Leading Edge Flow Measurement CheckPlus™ System instrumentation, which allows for more accurate measurement of feedwater flow.

Subsequent to issuance, your staff noted that the amendment contained errors. The following errors were noted:

1. In Section 3.6.1.2, a regenerative heat exchanger is discussed in the letdown system. DBNPS does not have a regenerative heat exchanger in the letdown system.

2. Section 3.6.2.1 contains an inaccurate description of the normal operation of the steam generator (SG) blowdown system.

3. Section 3.6.2.2 contains an inaccurate description of the SG blowdown system.

4. In Section 3.7.2.3, there is a reference to an alarm that will be added. DBNPS already has this alarm installed and functional.

5. Section 3.11.1 included reference to atmospheric dump valves. DBNPS has atmospheric vent valves.

6. Section 3.11.1 contains an inaccurate description of the normal operation of the turbine bypass valves.

7. Section B references bounding conditions 10 percent SG tube plugging. This should reference 20 percent.

The NRC staff verified that these errors were administrative in nature and did not affect the NRC's conclusions about the acceptability of this amendment.

The NRC staff has verified the errors have been corrected and is including an updated safety evaluation. The NRC regrets any inconvenience the errors may have caused you.

Sincerely,

/RA/

Cameron S. Goodwin, Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosure: Safety Evaluation

cc w/encl: See next page

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Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-346
Enclosure: Safety Evaluation
cc w/encl: See next page

Distribution: see next page

ADAMS Accession No. ML082040387 NRR-106

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SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1 – CORRECTION OF ADMINISTRATIVE ERRORS IN AMENDMENT RE: MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (TAC NO. MD8326)

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 278

TO FACILITY OPERATING LICENSE NO. NPF-3

FIRSTENERGY NUCLEAR OPERATING COMPANY

FIRSTENERGY NUCLEAR GENERATION CORP.

DAVIS-BESSE NUCLEAR POWER STATION, UNIT NO. 1

DOCKET NO. 50-346

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC, the Commission) dated April 12, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML071030396), as supplemented by letters dated September 18, 2007 (ADAMS Accession No. ML072640612), October 8, 2007 (ADAMS Accession No. ML072830028), October 19, 2007 (ADAMS Accession No. ML072960032), January 15, 2008 (ADAMS Accession Nos. ML080180011 and ML080180012), February 14, 2008 (ADAMS Accession No. ML080510247), February 20, 2008 (ADAMS Accession No. ML080530392), March 12, 2008 (ADAMS Accession No. ML080770169), and May 16, 2008 (ADAMS Accession No. ML081410458), FirstEnergy Nuclear Operating Company, et al. (FENOC, the licensee) requested changes to the technical specifications (TSs) to increase the licensed thermal power level for the Davis-Besse Nuclear Power Station, Unit No. 1 (DBNPS).

Specifically, the proposed changes would make the Operating License and TS changes necessary to allow an increase in the rated thermal power (RTP) of approximately 1.63 percent, from 2772 megawatts thermal (MWt) to 2817 MWt, based on the use of the Caldon, Inc., Leading Edge Flow Measurement (LEFM) CheckPlus™ System instrumentation, which allows for more accurate measurement of feedwater (FW) flow. The licensee developed the license amendment request (LAR) following the guidance of NRC Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications."

The supplements dated September 18, October 8, October 19, 2007, January 15, February 14, February 20, March 12, and May 16, 2008, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

2.0 BACKGROUND

Nuclear power plants are licensed to operate at a specified core thermal power. Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix K, requires licensees to assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level when performing loss-of-coolant (LOCA) and emergency core cooling system (ECCS) analyses. This requirement is included to ensure that instrumentation uncertainties are adequately accounted for in the analyses. Appendix K to 10 CFR Part 50 allows licensees to

assume a power level less than 1.02 times the licensed power level (but not less than the licensed power level), provided the licensee has demonstrated that the proposed value adequately accounts for instrumentation uncertainties. The licensee has proposed to use a power measurement uncertainty of 0.37 percent. To achieve this level of accuracy, the licensee has installed a Caldon LEFM CheckPlus™ ultrasonic flow measurement system for measuring the main FW flow at DBNPS. The Caldon system provides a more accurate measurement of FW flow than the FW flow measurement accuracy assumed during the development of the original 10 CFR Part 50, Appendix K requirements and that of the current method of FW flow measurement used to calculate reactor thermal output. The Caldon system will measure FW mass flow to within plus or minus (±) 0.29 percent for DBNPS. This bounding FW mass flow uncertainty would be used to calculate a total power measurement uncertainty of 0.37 percent.

On the basis of this, the licensee proposed to reduce the power measurement uncertainty required by 10 CFR Part 50, Appendix K to 0.37 percent. The improved power measurement uncertainty would obviate the need for the 2 percent power margin originally required by 10 CFR Part 50, Appendix K, thereby allowing an increase in the reactor power available for electrical generation by 1.63 percent. This accuracy is supported by Caldon Topical Report ER-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check™ System," which, by the safety evaluation (SE) dated March 8, 1999 (ADAMS Accession No. 9903190065 (legacy library)), was approved by the NRC staff for use in justification of measurement uncertainty recapture (MUR) power uprates up to 1 percent. Subsequently, by the SE dated December 20, 2001 (ADAMS Accession No. ML013540256), the NRC staff approved Caldon Topical Report ER-157P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check™ or LEFM CheckPlus™ System," for use in justifying MUR power uprates up to 1.7 percent.

In large part, the basis for acceptability of a proposed MUR power uprate is that the uprated conditions are bounded by the current analyses of record. Historically, the majority of analyses were performed assuming 102 percent core power. Therefore, the analyzed power level, including uncertainty, does not change for the MUR power uprate. The exceptions to this are reviewed in detail by the NRC staff. RIS 2002-03 recommends that, to improve efficiency of the NRC staff's review, licensees requesting an MUR power uprate should identify existing design basis accident (DBA) analyses of record which bound plant operation at the proposed uprated power level. For any existing DBA analyses that do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the reanalysis.

3.0 EVALUATION

3.1 Instrumentation and Controls

Topical Report ER-80P describes the LEFM technology, includes calculations of power measurement uncertainty using an LEFM Check™ System in a typical two-loop pressurized water reactor (PWR) or two-FW-line boiling water reactor (BWR), and provides guidelines and equations for determining the plant-specific power calorimetric uncertainties. The Topical Report ER-157P describes the LEFM CheckPlus™ System and lists non-proprietary results of a typical PWR or BWR thermal measurement uncertainty calculation using LEFM Check™ or LEFM CheckPlus™ Systems. These two reports together provide a generic basis and guidelines for power uprates.

The plant-specific basis for the proposed uprate is provided in Cameron Engineering Report ER-202 Revision 3, "Bounding Uncertainty Analysis for Thermal Power Determination at

Davis-Besse Nuclear Power Station Using LEFM ✔+ System", included as Enclosure 4 in the licensee's response to the NRC staff's request for additional information (RAI) dated September 18, 2007.

The setpoint calculation methodology for the High Flux trip setpoint in TS Table 3.3-1 is provided in the licensee's calculation, NOP-CC-3002-01 Rev. 03, "RPS Reactor Power Related Field Trip Setpoints," included as an enclosure in the licensee's response to the NRC staff's RAI dated February 14, 2008.

3.1.1 Regulatory Evaluation

Nuclear power plants are licensed to operate at a specified core thermal power. In this regard, Appendix K to 10 CFR Part 50 requires LOCA and ECCS analyses to assume "that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level" to allow for instrumentation uncertainties. Alternately, Appendix K allows an assumption of lower than the specified 102 percent, but not less than the licensed thermal power level, "provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error." This allowance provides licensees an option of justifying a power uprate with reduced margin between the licensed power level and the power level assumed in the ECCS analysis by using more accurate instrumentation to calculate the rated thermal power (RTP). Because the maximum power level of a nuclear plant is a licensed limit, a proposal to raise the licensed power level must be reviewed and approved under the license amendment process. The LAR should include a justification for the reduced power measurement uncertainty to support the proposed power uprate.

The Caldon Topical Report ER-80P and its supplement, ER-157P, describe the LEFM CheckPlus™ System for the measurement of FW flow and provide a basis for up to 1.7 percent uprate of the licensed RTP. The NRC staff also considered the guidance of NRC RIS 2002-03 in its review of the licensee's submittals for the proposed power uprate request.

The LEFM CheckPlus™ System does not perform any safety function and is not used to directly control any plant system. However, adjustment of reactor power nuclear instrumentation (NI), which is considered important to safety, is based on the LEFM CheckPlus™ System calorimetric calculations.

3.1.2 Technical Evaluation

Neutron flux instrumentation is calibrated to the core thermal power, which is determined by an automatic or manual calculation of the energy balance around the plant nuclear steam supply system (NSSS). This calculation is called the "secondary calorimetric" for a PWR. The accuracy of this calculation depends primarily upon the accuracy of FW flow and FW net enthalpy measurements. FW flow is the most significant contributor to the core thermal power uncertainty. An accurate measurement of this parameter will result in an accurate determination of core thermal power.

The instrumentation for measuring FW flow rate typically is a venturi. This device generates a differential pressure proportional to the FW velocity in the pipe. Due to the high cost of calibration of the venturi and the need to improve flow instrumentation measurement uncertainty, the industry assessed other flow measurement techniques and found LEFM Check™ and LEFM CheckPlus™ ultrasonic flow meters (UFMs) to be viable alternatives.

Both systems use the transit time methodology to measure fluid velocity. The basis of the transit time methodology to measure fluid velocity and temperature is that ultrasonic pulses transmitted through a fluid stream travel faster in the direction of the fluid flow than opposite the flow. The difference in the upstream and downstream traversing times of the ultrasonic pulse is proportional to the fluid velocity in the pipe, and the temperature is determined using a pre-established correlation between the mean propagation velocity of the ultrasound pulses in the fluid and the fluid pressure.

Both systems use multiple diagonal acoustic paths, instead of a single diagonal path, so that velocities measured along each path can be numerically integrated over the pipe cross section to determine the average fluid velocity in the pipe. This fluid velocity is multiplied by a velocity profile correction factor, the pipe cross section area, and the fluid density to determine the FW mass flow rate in the piping. The mean fluid density may be obtained using the measured pressure and the derived mean fluid temperature as an input to a table of thermodynamic properties of water. The velocity profile correction factor is derived from calibration testing of the LEFM in a plant-specific piping model at a calibration laboratory.

The LEFM Check^{TM} System, as described in Topical Report ER-80P, consists of a spool piece with eight transducers, two on each of the four acoustic paths in a single plane of the spool piece. The velocity measured by any one of the four acoustic paths is the vector sum of the axial and the transverse components of fluid velocity as projected onto the path. The LEFM CheckPlus^{TM} system uses 16 transducers, 8 each in 2 orthogonal planes of the spool piece. As such, the LEFM CheckPlus^{TM} System is a combination of two LEFM Check^{TM} Systems.

In the LEFM CheckPlus^{TM} System, when the fluid velocity measured by an acoustic path in one plane is averaged with the fluid velocity measured by its companion path in the second plane, the transverse components of the two velocities are canceled and the result reflects only the axial velocity of the fluid. This makes the numerical integration of four pairs of averaged axial velocities and computation of volumetric flow inherently more accurate than can be obtained using four acoustic paths in a single plane. Also, since there are twice as many acoustic paths and there are two independent clocks to measure the transit time, errors due to uncertainties in path length and transit time measurements are reduced.

The NRC staff review in the area of Instrumentation & Control (I&C) covers the proposed plant-specific implementation of the FW flow measurement technique and the power increase gained as a result of implementing this technique in accordance with the guidelines (A thru H) provided in Section I of Attachment 1 to RIS 2002-03. The NRC staff review was conducted to confirm that the licensee's implementation of the proposed FW flow measurement device was consistent with the NRC staff-approved Caldon Topical Reports ER-80P and ER-157P, and adequately addressed the four additional requirements listed in the NRC staff SE for ER-157P. The NRC staff also reviewed the power measurement uncertainty calculations to ensure that (1) the conservatively proposed uncertainty value of 0.37 percent correctly accounted for all uncertainties due to power level instrumentation errors and (2) the uncertainty calculations met the relevant requirements of Appendix K to 10 CFR Part 50 as described in Section 3.1.1 of this SE. Additionally, the NRC staff reviewed the proposed limiting conditions of operation (LCO), surveillance requirement (SR), and the limiting safety system setting (LSSS) setpoint changes for compliance with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36.

The licensee's submittals provided the following information regarding the LEFM CheckPlus^{TM} system FW flow measurement technique and its implementation in DBNPS.

The licensee stated that the LEFM CheckPlus™ System at DBNPS consists of an electronic cabinet in the main control room and two measurement sections (spool pieces) located upstream of the existing FW Flow Venturis in the Turbine Building, in each of the two 18-inch main FW flow headers that feed each steam generator (SG). The LEFM flow meters were calibrated at the Alden Research Laboratory (ARL), Inc. facility using the plant's current piping configuration. Each measurement section consists of 16 ultrasonic transducer housings, forming the pressure boundary. Each transducer may be removed at full-power conditions without disturbing the pressure boundary. The licensee further stated that the installation location of these flow elements conforms to the requirements in Topical Reports ER-80P and ER-157P.

The LEFM CheckPlus™ 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'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 FW temperature are displayed on the electronic cabinet and transmitted to the plant process computer for use in the calorimetric measurement. The LEFM indications of FW 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 FW flow and RTD temperature will continue to be used for FW control and other functions that they currently perform. The Caldon panel has outputs for internally generated system trouble alarms, which will be wired into the main control room annunciator and plant process computer.

<u>Items A through C of Section I of Attachment 1 to RIS 2002-03</u>

Items A, B, and C in Section I of Attachment 1 to RIS 2002-03, respectively, require licensees to identify the approved topical reports on the FW flow measurement technique, provide references to the NRC's approval of the measurement technique, and provide a discussion of the plant-specific implementation of the guidelines in the topical report and the NRC staff's approval of the technique.

In the LAR, the licensee identified topical reports ER-80P, Revision 0, and ER-157P, Revision 5, as applicable to the LEFM CheckPlus™ System. The licensee also referenced NRC SEs for ER-80P, dated March 8, 1999, and for ER-157P, dated December 20, 2001.

The licensee stated that the LEFM CheckPlus™ System was installed at DBNPS in accordance with the requirements of topical report ER-80P and ER-157P, and will be used for continuous calorimetric power determination by serial link with the plant process computer. The licensee further stated that hydraulic profile and signal processing requirements are met within its design basis uncertainty analysis.

Based on the NRC staff's review of the licensee's submittals as reflected in the above discussion, the staff finds that the licensee has sufficiently addressed the plant-specific implementation of the LEFM CheckPlus™ System topical report guidelines, and that the licensee's description of the FW flow measurement technique and the power uprate due to implementing this technique adequately addresses the guidance in items A through C of Section I of Attachment 1 to RIS 2002-03.

Items D, G, and H of Section I of Attachment 1 to RIS 2002-03

Items D, G, and H in Section I of Attachment 1 to RIS 2002-03, respectively require licensees to provide dispositions of the four criteria that the NRC stated should be addressed when implementing the FW flow measurement uncertainty technique, provide a proposed allowed outage time (AOT) for the instrument, and propose actions to reduce power if the AOT is exceeded.

The NRC staff's SE on Caldon Topical Report ER-157P included four additional criteria to be addressed by a licensee referencing this topical report to support an MUR power uprate. In its LAR and supplements, the licensee addressed each of the four criteria as follows:

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

Licensee Response:

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 . . . . 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 licensee stated that the Nuclear Instrumentation (NI) indicated power is compared against the heat balance power on a daily basis. Should the LEFM system become unavailable, it must be restored to operable status or the plant power will be reduced to 98.4 percent RTP (≤ 2772 MWt) with four reactor coolant pumps (RCPs) operating, or ≤ 73.8 percent RTP (≤ 75 percent of 2772 MWt) with three RCPs operating, prior to the next NI-to-daily heat balance comparison. The justification for the AOT of the LEFM is that the NIs were compared to the last known good heat balance calculation using the LEFM measurements which do not routinely require adjustments, and thus can continue to be relied upon for power measurement until the next daily comparison.

At most, this AOT would be for a period of 30 hours based on the current requirements for Functional Unit 2, "High Flux," of Technical Specification (TS) Table 4.3-1, which includes taking credit for the 25 percent surveillance interval extension.

Since the NI High Flux trip setpoint is based on the operating power level of the reactor determined by secondary heat balance calculation using the LEFM for the proposed RTP uprate or the existing venturi nozzle for the current RTP, the licensee proposed TS changes to incorporate the required actions, action completion times, and NI trip setpoint allowable

values (AVs) for the LEFM inoperable condition. In addition to the power reduction with an inoperable LEFM, the NI High Flux trip setpoint will be reduced to an AV of ≤ 103.3 percent RTP with four RCPs operating (or ≤ 80.6 percent RTP with three RCPs operating) from the current value of ≤ 104.9 percent RTP within 10 hours after the NI-to-daily heat balance comparison. FENOC stated that NI trend analysis indicates that the NI to heat balance comparison will not drift significantly over a 3-week period, and surveillance data indicates essentially no drift of the high flux setpoints. As such, the expected setpoint drift over 40 hours is insignificant and the NIs will remain calibrated for an extended period of time.

The NI High Flux trip setpoint verification surveillance needed to address the instrument operability is addressed by the two notes applicable to NI channel calibration requirement in the TSs. In its letter to NRC dated February 14, 2008, the licensee provided the NI calibration procedure and Reactor Protection System (RPS) Reactor High Flux trip setpoint calculation in FENOC document NOP-CC-3002-01 Rev. 03, dated February 03, 2006. This document calculated NI High Flux instrumentation limiting trip setpoint based on the total loop uncertainty per the plant specific methodology based on the Method 1 of Instrument Society of America (ISA) ISA-RP67.04.02-2000. The document also calculated as-found and as-left setpoint tolerances of the NI to establish setpoint AVs. The licensee stated that the RPS instrumentation setpoint AVs are based on protecting the analytical limits used in DBNPS safety analysis with the consideration of appropriate uncertainties.

The NRC staff review found the methodology acceptable and the calculated setpoint tolerances to have sufficient margin to the AVs.

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

   <u>Licensee Response</u>:

   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 [its] 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 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.

Based on the foregoing, the NRC staff concludes that DBNPS adequately addressed Criterion 2.

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

Licensee Response:

The LEFM uncertainty calculation is based on the [American Society of Mechanical Engineers] ASME PTC 19.1 methodology . . . and the [Alden] calibration tests . . . . 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.

Based on the foregoing, the NRC staff concludes that DBNPS adequately addressed Criterion 3.

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

Licensee Response:

The calibration factor for the DBNPS spool pieces was established by tests of these spools at [ARL] in October 2001 . . . . These included tests of a full-scale model of the

> DBNPS hydraulic geometry and tests in a straight pipe. An [ARL] data report for these tests . . . and a Caldon engineering report evaluating the test data . . . are on file in the D-B Records Management System . . . .
>
> Final acceptance of the site-specific uncertainty analyses occurred after the completion of the commissioning process. The commissioning process verified bounding calibration test data . . . . This step provided final positive confirmation that actual performance in the field meets the uncertainty bounds established for the instrumentation . . . . Final commissioning was completed in July 2004.

Based on the foregoing, the NRC staff concludes that DBNPS adequately addressed Criterion 4.

The licensee's proposed AOT for the instrument and the proposed actions to reduce power if the AOT is exceeded (Items G and H of RIS 2002-03, Attachment 1, Section I) are described in the evaluation of Item D Criterion 1 above.

The NRC staff's review of the licensee responses found that the licensee had fully addressed the four criteria specified in the NRC staff's SE of topical report ER-157P, and therefore, has adequately addressed the guidance in item D, G, and H of Section I of Attachment 1 to RIS 2002-03.

<u>Items E and F of Section I of Attachment 1 to RIS 2002-03</u>

Items E and F in Section I of Attachment 1 to RIS 2002-03, respectively require licensees to submit a plant-specific total power measurement uncertainty calculation, explicitly identifying all parameters and their individual contribution to the power uncertainty, and to provide information to address the specified aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric.

To address item E of RIS 2002-03, the licensee provided a summary of DBNPS core thermal power measurement uncertainty in a table format based on Cameron Engineering Report ER-202, which provides a detailed calculation of the uncertainties. The licensee stated that the values in the uncertainty column of the table and the total power uncertainty determination are bounding values. The NRC staff's audit of ER-202 found that the calculations determined individual measurement uncertainties of all parameters contributing to the core thermal power measurement uncertainty and those uncertainties were then combined using square root of sum of squares (SRSS) methodology, as described in Regulatory Guide (RG) 1.105 and ISA-S67.04.

The NRC staff finds that the licensee has provided calculations of the total power measurement uncertainty at the plant, explicitly identifying all parameters and their individual contribution to the power uncertainty, and therefore, has adequately addressed the guidance in item E of Section I of Attachment 1 to RIS 2002-03.

In the LAR, the licensee addressed each of the five aspects of the calibration and maintenance procedures listed in item F of RIS 2002-03 related to all instruments that affect the power calorimetric as follows:

i) Maintaining Calibration

Calibration of the LEFM will be ensured by the preventative maintenance activity described in maintenance plan 83956. In addition to the maintenance activities listed in

response to criterion 1, the preventative maintenance activity will verify the calibration of the 5 MHz clock in the acoustic processor unit and power supplies and verify that the wall thickness of the FW spool pieces are within tolerances.
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.

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.

iii) Performing Corrective Actions

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

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. Corrective action procedures, which ensure compliance with the requirements of 10 CFR Part 50, Appendix B, include instructions for notification of deficiencies and error reporting.

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.

The NRC staff review of the licensee's above statements found that the licensee addressed the calibration and maintenance aspects of the LEFM CheckPlus™ system and all other instruments affecting power calorimetric and, thus, complied with the guidance item F of Section I of Attachment 1 to RIS 2002-03.

3.1.3 Summary

The NRC staff review of the licensee's proposed plant-specific implementation of the FW flow measurement device and the power uncertainty calculations determined that the licensee's proposed amendment is consistent with the NRC staff's approved topical report ER-80P and its supplement, ER-157P. The NRC staff has also determined that the licensee adequately accounted for the power level instrumentation uncertainties in the reactor thermal power measurement uncertainty calculations and demonstrated that the calculations meet the relevant requirements of 10 CFR Part 50, Appendix K as described in Section 3.1.1 of this SE. Therefore, the NRC staff finds the I&C aspect of the proposed MUR power uprate acceptable.

3.2 Reactor Systems

3.2.1 Regulatory Evaluation

Early revisions of 10 CFR 50.46, and Appendix K to 10 CFR Part 50, required licensees to base their LOCA analysis on an assumed power level of at least 102 percent of the licensed thermal power level to account for power measurement uncertainty. The NRC later modified this requirement to permit licensees to justify a smaller margin for power measurement uncertainty. Licensees may apply the reduced margin to operate the plant at a level higher than the previously licensed power. The licensee proposed to use a Caldon LEFM CheckPlus™ System to decrease the uncertainty in the measurement of FW flow, thereby decreasing the power level measurement uncertainty from 2.0 percent to 0.37 percent.

The licensee developed its LAR consistent with the guidelines in NRC RIS 2002-03 (Reference 10).

In its April 12, 2007, application (Reference 1), the licensee identified the NRC staff's original evaluations of the Caldon LEFM in References 2 and 3. Following cases of overpower while depending upon UFMs, the NRC staff re-evaluated the hydraulic issues that contributed to the overpower conditions. Reference 4 provided the following information for the LEFMs:

- A theoretical description of LEFM operation. This showed that flatness ratio, defined as the ratio of the measured average axial velocity at the outside chords to the average axial velocity at the inside chords, can be correlated to the LEFM correction factor or calibration coefficient. Note that this does not apply if the CheckPlus™ is located too close to a flow perturbation such as an elbow.

- Substantiation that the uncalibrated CheckPlus™ is typically within a fraction of a percent of the flow rate measured at ARL. The average correction factor for the uncalibrated Seabrook CheckPlus™ for a series of five ARL tests with swirl less than 2.0 percent was +0.28 percent.

- Substantiation that the CheckPlus™ is typically relatively unaffected by flow profile distortion and swirl and, further, that the CheckPlus™ will provide an approximation of the flow profile. Note that this conclusion does not apply if the flow profile consists of multiple individual flow paths such as may exist immediately downstream of a tubular flow straightener or if certain distortion of the flow profiles occurs.

- Flatness ratio can be used for correlation of the calibration coefficient so that reliance on a Reynolds Number extrapolation is not necessary to apply ARL test results to plant applications. Note that this does not apply if the CheckPlus™ is located too close to a flow perturbation such as an elbow.

- Generically, uncertainty associated with the CheckPlus™ calibration coefficient is ±0.25 percent.

- ARL flow rate uncertainty for the Seabrook CheckPlus™ calibration was ±0.088 percent.

- "The NRC staff finds that the hydraulic aspects of Check and CheckPlus systems have been accurately described in applicable Caldon documentation, that there is a firm theoretical and operational understanding of behavior, and, with one exception, there is no further need to re-examine the hydraulic bases for use of the Check and CheckPlus systems in nuclear power plant FW applications. The exception, which should be followed up by Caldon for generic purposes, is to establish the effect of transducer replacement on the Check and CheckPlus system uncertainties."

3.2.2 Technical Evaluation

A. FW FLOW MEASUREMENT DEVICES – CALDON LEFM CHECKPLUS™ SYSTEM

A.1 Installation

In its LAR, the licensee stated that the Caldon LEFM CheckPlus™ System at DBNPS consists of two measurement section/spool pieces located in the Turbine Building in each of the two 18-inch main FW flow headers that feed each SG. The measurement sections are located upstream of the existing FW flow venturis in the turbine building and auxiliary building. The 'A' LEFM is located approximately 10-feet downstream of a 90-degree horizontal-to-vertical elbow and 15-feet upstream of one of the existing venturis. The 'B' LEFM is located approximately 10 feet downstream of a 90-degree vertical-to-horizontal elbow and 11-feet upstream of a 90-degree horizontal-to-vertical elbow. The licensee stated that the LEFM flow meters were calibrated at the ARL facility based on the plant's current piping configuration and variations of the plant's configuration.

Significant variations in piping configuration between the in-situ LEFM installation and the experimental calibration facility could adversely affect the LEFM calibration. Reference 2 discussed the following differences between the plant and experimental configurations: (1) a shorter run of pipe preceding the 90-degree elbow at the experimental facility; and (2) rather than vertical-to-horizontal elbows, the experimental configuration used only horizontal runs of pipe.

The licensee stated that the longer run of straight piping prior to the 90-degree elbows at DBNPS will allow for more flow straightening at the plant than occurred in the test facility, and that this difference "is judged to be insignificant." The NRC staff finds that the increased amount of flow straightening would provide conditions conducive to a more accurate reading using the LEFM, and therefore, that the difference between the in-situ installation and the test configuration results in, if any, conservative effects on the LEFM calibration. For this reason, the NRC staff finds this difference in configuration acceptable.

The licensee stated that the difference between the vertical piping alignment in the in-situ installation versus the horizontal piping alignment at the test facility is judged to be insignificant. The effect of this difference is a potential difference in spool piece alignment relative to the elbow, which could result in local flow distribution differences between the laboratory and in-situ installations. To verify the insignificance of this potential difference, the NRC staff reviewed additional information submitted by the licensee, specifically Caldon Engineering Report 227, "Profile Factor Calculation and Accuracy Assessment for the DBNPS LEFM CheckPlus™ Spool Pieces" (Enclosure to Reference 2). The report documented testing performed, which included testing the LEFM CheckPlus™ spool pieces in both horizontal and vertical configurations. The results of this testing are included in the overall profile factor calculation, which forms the basis for one of the uncertainty terms in the overall LEFM CheckPlus™ thermal power uncertainty. Also, because of the way the spool piece is designed, the LEFM CheckPlus™ system is, to some extent, insensitive to these effects. Because the uncertainty evaluation considered differences in the installation relative to the upstream elbow alignment, and because of the design of the LEFM CheckPlus™ System, the NRC staff agrees that the difference between the vertical in-situ configuration and the horizontal testing configuration is insignificant, and thus acceptable.

Each measurement section consists of 16 ultrasonic transducer housings, forming a pressure boundary. Each transducer may be removed at full-power conditions without disturbing the pressure boundary. The licensee stated that the installation location of these flow elements is in accordance with the requirements in Topical Reports ER-80P and ER-157P (References 2 and 3).

A.2 CheckPlus™ Operating History at DBNPS

The CheckPlus™ UFMs were installed in 2002. Final commissioning was completed in July 2004 to establish that field performance met the uncertainty bounds previously established for the UFMs. Active monitoring has been ongoing since February 2005. A transducer failed in June 2005 and another failed later. All transducers were replaced in June 2006 and again in March 2007. Transducer replacement is addressed in Section A.4, below. The NRC staff's assessment of CheckPlus™ historic operation at DBNPS is provided in Section A.6 below.

A.3 CheckPlus™ Inoperability

To operate above the presently licensed power of 2772 MWt, the licensee proposes that, if the CheckPlus™ is inoperable or not used in performance of the daily heat balance, then it must be restored to operability prior to completion of the next required daily heat balance measurement or thermal power must be reduced to ≤98.4 percent of rated thermal power with four RCPs operating. This requirement is stated in Section 3/4.3.4.1 of the Technical Requirements Manual. The intent is to ensure operation with a 2 percent flowrate uncertainty margin consistent with pre-uprate operation using existing flow measurement instrumentation so that licensing basis accident and operational limits are preserved. The licensee justifies this operation by citing historical comparisons of the CheckPlus™ and venturi-based flow rate instrumentation where there has been no significant divergence during power operation over short periods and that long-term venturi fouling results in more conservative FW flow input to the heat balance calculation. The NRC staff notes that (1) a recalibrated venturi that was fouled during calibration would indicate a non-conservative flow rate if it defouled, an unacceptable condition with the stated response to a non-operable CheckPlus™, and (2) significant perturbations in FW flow rate can induce defouling. These aspects are not applicable to the DBNPS installation because, as noted on Page 17 of 18 in Reference 5, "FENOC plans to always maintain the DBNPS LEFM independent of the venturis." Stated differently, the licensee does not recalibrate its venturis and venturi indication will remain consistent with current operation.

Some licensees have required that thermal power not change by more than 10 percent when operating without an operable CheckPlus™ as part of the justification for several days of operation before reducing thermal power. The licensee has proposed a shorter time without the qualification. The NRC staff finds that the licensee's proposed operation is acceptable since it is shorter and there is no miscalibration concern regarding use of venturis since they are not recalibrated.

A.4 Transducer Replacement

The Reference 4 qualification to establish the effect of transducer replacement on the Check™ and CheckPlus™ System uncertainties has been addressed in References 6 and 7. A number of tests were conducted in which the transducers were removed and replaced for each test. Each of the tests consisted of a statistically meaningful number of individual determinations of the calibration factor. The calibration factors, and uncertainty associated with each calibration factor, were provided and found to be acceptable. The licensee addressed the effect of transducer replacement in Enclosures 3 and 6 to Reference 8 where the bounding uncertainty due to transducer installation variability was reported to increase the total FW mass flow uncertainty by a very small percentage, with no change in the FW temperature uncertainty. This added an uncertainty that was not addressed in the previous documentation (References 2 and 3). Further, there are significant differences between the bounding uncertainties in the older references and in the licensee's submittal. Consequently, the NRC staff assessed these differences and found them to be justified and acceptable.

The licensee has replaced transducers several times. As discussed in Section A.6 below, the process of assessing transducers and the observed changes are consistent with the above conclusion. Therefore, the NRC staff finds that transducer installation variability has been acceptably addressed.

A.5 CheckPlus™ Calibration and Application Considerations

The NRC staff notes the following regarding the CheckPlus™ calibration that was accomplished at ARL (Reference 9):

- Statistically meaningful tests were conducted in a straight pipe and downstream of an elbow that was preceded by a straight pipe.

- The two spool pieces were installed in series in each test configuration. Several tests were conducted at different flow rates and then the spool pieces were reversed and additional tests at different flow rates were conducted.

- Spool pieces were rotated to assess the effect of positioning with respect to the flow profile in the tests downstream of the elbow.

- The elbow test configuration included elbows, a flow straightener, a straight pipe of sufficient length to minimize flow profile distortion, and the spool pieces in series. Tests were also conducted with the flow straightener removed.

- In the elbow test configuration, distance from the upstream elbow to the UFMs bracketed the distance from the upstream elbow in the plant when the UFM positions were reversed.

Regardless of the configuration, the maximum change in calibration factor was a fraction of a percent. As was the case for previous reviews of ARL CheckPlus™ tests, the test temperature was room temperature in contrast to the plant FW temperature of approximately 457 ^{o}F. A correlation factor was used to extrapolate test results to plant operating conditions as was used for some previous CheckPlus™ applications.

It was noted that operation with unequal heating from the FW heaters could potentially introduce thermal stratification in FW passing through the CheckPlus™. However, the CheckPlus™ will provide the average velocity of sound along those paths. The sound velocity will, in turn, provide average path temperatures and an accurate determination of average FW temperature, acceptably addressing any non-uniform FW temperature concerns.

The NRC staff finds the licensee's calibration, application, and FW temperature determination methods acceptable based on the fact that varied configurations were used and, regardless of the configuration or potential thermal stratification, the maximum change in calibration factor was a fraction of a percent.

A.6 CheckPlus™ Operation at DBNPS[1]

CheckPlus™ UFMs were installed at DBNPS in 2002 and the commissioning process was completed in July 2004. The preliminary overall mass flow rate uncertainty estimate was 0.30 percent and as tested, it was 0.26 percent. Preliminary FW temperature uncertainty was 0.60 °F and as tested was 0.58 °F. Conservative consideration of transducer replacement uncertainty increased the 0.26 percent to 0.29 percent, a value that is bounded by the assumed 0.30 percent for FW mass flow rate.

The UFMs were reported to have functioned as designed until a transducer failed in June 2005. Another transducer then failed and all transducers were replaced in June 2006. Flatness ratio, a measure of the velocity profile, increased slowly from May 2005 to June 2006 but remained within the established operational bounds. The calibration change associated with this change was calculated to be less than 0.02 percent. The licensee attributed this change to decreasing signal-to-noise ratio due to transducer aging. All transducers were again replaced in March 2007. The flatness ratio after the change was essentially identical to the May 2005 values and remained unchanged through December 2007.

The UFMs were not initially used for heat balance calculations until April 2007, and the licensee reported a comparison to other plant parameters for the time from April 3, 2007 through December 26, 2007. An increase in flow rate was indicated by the venturis starting in September 2007, while the UFMs may have indicated a slight decrease in flow rate; however, this is not clear from the graphs. The licensee attributed the change in the comparisons to venturi fouling. Comparison to other plant parameters showed an upward trend in first stage turbine pressure, a decrease in condenser pressure, and a slight increase in FW temperature. The comparison data are not sufficient to draw any conclusions other than that there are no gross errors and the venturi behavior is consistent with fouling.

A.7 Caldon Topical Reports ER-80P and ER-157P Safety Evaluation Criteria

The NRC staff reviewed and approved Caldon Topical Reports ER-80P and ER-157P related to the LEFM Check™ and LEFM CheckPlus™ Systems. The criteria contained in the topical reports were also used by the NRC staff in this review. In approving the Caldon Topical reports, the NRC staff established four criteria to be satisfied by each licensee as follows:

Criterion 1

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

The licensee stated that implementation of the power uprate license amendment will include developing the necessary procedures and documents required for operation, maintenance, calibration, testing and training with the incorporation of the LEFM system. To that end, the

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[1] Based on information provided in References 5 and 8.

licensee has developed a preventive maintenance program in accordance with the requirements set forth by Caldon.

In Reference 1, the licensee stated that general LEFM work activities include power supply checks and sensor replacements, which fall within the scope of the Instrumentation and Control Journeyman Qualifications. Additionally, the licensee indicated in its RAI response that training specific to the LEFM has been continually provided to plant personnel since 2002. The licensee also engages in analyses to determine future training and qualification needs relevant to the LEFM.

The licensee also indicated that required preventive maintenance activities are performed in accordance with a specific plant procedure and that these activities are performed every refueling outage.

The licensee's submittal discusses the preventive maintenance activities, and the NRC staff finds that, because the calibration of the LEFM system is verifiable online, and because the preventive maintenance activities are developed in accordance with guidance provided by Caldon, the proposed maintenance and calibration procedures are acceptable.

Criterion 2

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

The licensee reviewed operating experience with the LEFM System since its installation at DBNPS in 2002. The licensee has been actively monitoring the LEFM System since February 2005 and comparing the FW flow and temperature data to the flow venturis and RTD temperature output. According to the licensee, the data comparison showed that the LEFM System measures the FW flow and temperature consistently with the venturis and RTDs.

The licensee's submittal (Reference 1) discussed two transducer failures that have occurred, one of which resulted in an internal alarm that would have resulted in the LEFM system's removal from service. In Reference 8, the licensee clarified that other transducer failures have occurred, that the LEFM vendor is investigating the causes of these failures, and that upon completion of this investigation, the licensee will determine the appropriate corrective actions to take with regard to the transducer failures. The licensee also indicated in the April 12, 2007, submittal that all 32 transducers were replaced in March 2007. The licensee stated in Reference 8 that, should one or more transducer failures occur, the LEFM System Trouble Annunciator will be activated, which will prompt operators to follow an alarm procedure and remove the LEFM system from service.

In consideration of the information discussed above, the NRC staff finds that the licensee's evaluation of operational and maintenance history associated with the LEFM system is acceptable, and that Criterion 2 is satisfied.

Criterion 3

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

The licensee confirmed that FW flow and temperature uncertainties were combined with other plant measurement uncertainties to calculate the overall heat balance uncertainty using accepted plant setpoint methodology. The LEFM uncertainty calculation itself, however, is based on (1) the ASME PTC 19.1 methodology and (2) on calibration tests performed at the ARL. Both the LEFM uncertainty calculation and the accepted setpoint methodology use a SRSS calculation. Section A.5 above discusses the calibration tests in greater detail. The NRC staff finds the calibration tests acceptable.

Criterion 4

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

The licensee stated that Criterion 4 does not apply to DBNPS. The calibration factor for the DBNPS spool pieces was established by tests of these spools at ARL in October 2001. An ARL data report for these tests and a Caldon engineering report evaluating the test data are on file in the DBNPS 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. The site specific uncertainty analysis documents these analyses. The licensee maintains this document on file, as part of the technical basis for the DBNPS MUR uprate.

Based on its review of the licensee's responses, the NRC staff has determined that the licensee has addressed the four criteria contained in Reference 10 acceptable.

B. NSSS PARAMETERS

The NSSS design parameters provide the reactor coolant system (RCS) and secondary system conditions (pressures, temperatures, and flow) that are used as the basis for the design transients and for systems, components, accidents and transient analyses and evaluations. The parameters are established using conservative assumptions to provide bounding conditions to be used in the NSSS analyses. The major operating conditions are as follows:

1. Analyzed core power level of 2819 MWt (2836 MWt NSSS power level). This includes 17 MWt assumed for non-core heat addition, which the licensee clarified is conservative relative to the nominal value of 14-15 MWt.

2. Total RCS volumetric flow rate of 392,990 gallons per minute.

3. SG tube plugging of 0 percent and 20 percent.

4. Full power, normal operating T_{avg} of 582 °F.

5. FW temperature of 455 °F.

6. 15x15 Mark B fuel assemblies.

In all safety analyses except where noted in Table 3.1 below, the assumed initial power level was 102 percent of original licensed thermal power (OLTP).

C. ACCIDENTS AND TRANSIENTS FOR WHICH THE EXISTING ANALYSES OF RECORD BOUND PLANT OPERATION AT THE PROPOSED UPRATE POWER LEVEL

In all analyses, the licensee referenced the current analysis of record, which used previously NRC-approved computer codes and methodologies for each accident and transient analysis. Unless noted, the analyzed core power level was 2828 MWt, 2.0 percent greater than the current licensed core power level of 2772 MWt, and 0.37 percent greater than the MUR core power level of 2817 MWt. The NRC staff reviewed and approved the licensee's transient and accident analyses at 2828 MWt conditions assumed for normal operations, confirming that the acceptance criteria were still met under these conditions.

The results of the NRC staff's review are summarized in Table 3.1 below. Those accidents/transients identified in Table 3.1, where the safety analysis was not performed assuming 102 percent OLTP as an initial condition, are discussed and evaluated in Sections C.1 through C.9 below.

Table 3.1. Accident and Transient Analyses

Accident/Transient	Analyzed Core Power Level	Analysis of Record Bounds MUR Uprate	NRC Staff Conclusion/Discussion
Anticipated Transients Without Scram	100%	N/A	See Section 3.2.2.C.1
Uncontrolled rod cluster control assembly (RCCA) Withdrawal from Subcritical	10^{-7}%	Yes	See Section 3.2.2.C.2
Uncontrolled RCCA Withdrawal at Power	100%	Yes	See Section 3.2.2.C.2
Control Rod Assembly (CRA) Misalignment	100%	No	See Section 3.2.2.C.3
Makeup and Purification System Malfunction	100%	No	See Section 3.2.2.C.4
Total Loss of Forced Reactor Coolant Flow	102%	Yes	Acceptable
Startup of an Inactive RCP	60%	Yes	See Section 3.2.2.C.5
Loss of External Load/Turbine Trip	100%	No	See Section 3.2.2.C.6
Loss of Normal FW Flow	102%	Yes	Acceptable
Station Blackout	100%	No	See Section 3.2.2.C.7
Excessive Heat Removal Due to FW System Malfunction	102%	Yes	Acceptable
Anticipated Variations in the Reactivity of the Reactor	100%	No	See Section 3.2.2.C.8
SG Tube Rupture	102%	Yes	Acceptable
CRA Ejection Accident	102%	Yes	Acceptable
Steam Line Break	102%	Yes	Acceptable
Break in Instrument Lines of Lines from Primary System that Penetrate Containment	102%	Yes	Acceptable
Fuel Handling Accident	102%	Yes	Acceptable
Main steamline break (MSLB) Mass and Energy Release	102%	Yes	Acceptable
Overpressure Protection	100%	No	See Section 3.2.2.C.9
LOCA Radiological Consequences	102%	Yes	Acceptable
Small Break LOCA/ECCS Actuation	109.1%	Yes	Acceptable
Large and Small Break LOCA	109.1%	Yes	Acceptable
LOCA Mass and Energy Release	109.1%	Yes	Acceptable

C.1 Anticipated Transients Without Scram (ATWS)

The licensee stated that the ATWS transients are considered beyond the original design basis of Babcock & Wilcox (B&W) plants. The licensee also confirmed that the current ATWS analysis indicates that the peak predicted reactor pressure is significantly below the maximum pressure criterion of 3200 pounds per square inch – gauge (psig). Reference 8 clarified that the peak pressure criterion is 3250 psig. The licensee further indicated that the DBNPS ATWS response is documented in a September 29, 1989, letter, Thomas V. Wambach, NRC, to Donald C. Shelton, Toledo Edison Company, “Evaluation of the DBNPS Nuclear Power Station Compliance with 10 CFR 50.62 Requirements for Reduction of Risk from Anticipated Transients Without Scram” (Reference 11).

This document cites an NRC-reviewed and accepted generic analysis of B&W plant response to ATWS scenarios. The peak pressures documented in the report ranged from 3621 to 4190 pounds per square inch – atmospheric (psia) for a limiting transient with no reactor trip. The report described a diverse scram system, the purpose of which is to prevent the RCS pressure from exceeding 3250 psig. By crediting the diverse scram system and the ATWS mitigation system actuation circuitry, the analysis indicated that RCS pressure would be maintained below 3250 psig.

During power uprate feasibility studies conducted by the licensee at a power level of 3026 MWt, the licensee concluded that the RCS peak pressure during an ATWS scenario would be less than 2750 psig, which represents a significant margin to the ASME Service Level C limit of 3250 psig. Based on the acceptability of the original B&W ATWS analysis, and the significant margin determined by the feasibility study, the NRC staff finds the plant response to an ATWS at DBNPS acceptable for the proposed MUR power uprate.

C.2 RCCA Withdrawal Accidents; Subcritical and Full Power

These transients are terminated by an assumed TS High Flux Trip Setpoint of 112 percent of OLTP. As a part of the MUR, this analytical limit will be reduced to 110.2 percent of the uprated power level. As such, the analyzed power level at which these transients terminate will remain the same. Although the Uncontrolled RCCA withdrawal at power accident assumes an initial operating power of 2772 MWt, the licensee stated that the same amount of energy would be required to increase the RCS pressure and temperature to the assumed analytical setpoint. Therefore, the current analyses bound the proposed MUR uprate operation, and the NRC staff finds these accident analyses acceptable to support the MUR power uprate.

C.3 CRA Misalignment

Using NRC-approved methods, the licensee re-evaluated this event assuming an initial power level of 2966 MWt, which is greater than 102 percent of the OLTP. The licensee concluded from re-analyzing this transient at a greater power level that the analysis assuming a greater initial power level is bounded by the analysis of record.

In Reference 8, the licensee stated that this transient results in an over-cooling effect that does not result in a significant neutronic effect resulting from the moderator temperature coefficient. In the scoping analysis, the licensee stated, subsequent to dropping a misaligned CRA, the core power returned to approximately 95 percent of its initial value, whereas the analysis of record demonstrated a return to the initial power level. Thus, the licensee concluded, the analysis of record was more conservative based on a higher post-transient power level. The NRC staff agrees with this position.

It should be noted that these results are for end-of-life conditions. In the case of beginning-of-life conditions, the licensee indicated that the smallest worth rod resulted in a reactor trip. The analyzed reactor trip setpoints, both at the currently licensed thermal power level, and at the proposed uprated power level, are higher than those used during operations at the plant in either case. Therefore, the NRC staff concludes that, for beginning-of-life conditions, the analysis of record bounds operation at the uprated conditions. Based on the considerations discussed above, the NRC staff finds that the DBNPS response to a CRA misalignment will be acceptable at the proposed uprated conditions.

C.4 Makeup and Purification System Malfunction

The licensee stated that the current analysis of record, performed at 100 percent of OLTP, is bounded by the rod withdrawal at power accident analysis. The licensee stated that reactivity insertion rates and peak RCS temperature and pressure were all significantly less than the analysis indicated for the rod withdrawal at power accident. The NRC staff reviewed this accident analysis in the licensee's updated final safety analysis report (UFSAR) and concluded that the transients were sufficiently similar in terms of termination because both end with a trip on high power or high pressure.

On the bases that the current analysis remains conservative for uprated conditions, and that the RCCA Withdrawal at Power accident bounds the effects of this accident, the NRC staff accepts the current analysis of record and the licensee's justification that the RCCA withdrawal accidents bound this accident. The staff concludes on these bases that the Makeup and Purification System Malfunction Accident will not be unacceptably affected by the proposed MUR power uprate.

C.5 Startup of an Inactive Reactor Coolant Loop

The licensee stated that this accident was analyzed assuming an initial power of 60 percent OLTP, which is 4.9 percent greater than the maximum allowed thermal power for two-pump (single coolant loop) operation. Therefore, the initial conditions assumed in this analysis bound the power measurement uncertainty, and the NRC staff finds that the current analysis bounds operation at the uprated power level.

C.6 Loss of External Load and/or Turbine Trip

The licensee stated that 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 FW Flow accident. The NRC staff reviewed both transients as analyzed in the UFSAR and determined that the characteristics of a Loss of External Load/Turbine Trip and a Loss of Normal FW Flow result in a diminished capability of the RCS to remove heat from the reactor. The Loss of Normal FW Flow event, however, is more severe than the Loss of External Load/Turbine Trip event such that the Loss of Normal FW event bounds the loss of external load/turbine trip event. Therefore, the NRC staff finds the licensee's conclusion that the Loss of Normal FW Flow event, analyzed at 102 percent OLTP, bounds the Loss of External Load/Turbine Trip analysis acceptable for the purposes of the MUR power uprate.

C.7 Station Blackout (SBO)

The licensee noted that DBNPS has a SBO diesel generator, and described the sequence of events that occur in a SBO. The licensee stated that immediate plant system response to the SBO is a 4-to-0 pump coast down event, which was analyzed at 102 percent of OLTP. Also, the long-term cooling is provided by automatic initiation of the auxiliary FW (AFW) system. The flow rates required of the AFW system are based on the Loss of Normal FW Flow event, which is also analyzed at 102 percent OLTP. Therefore, the licensee concluded, and the NRC staff agrees, that other events bound the SBO analysis such that re-analysis is not necessary for the MUR power uprate.

C.8 Anticipated Variations in Reactivity of the Reactor

The licensee stated that this analysis was originally 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. Because no safety system actuation is required to mitigate this accident, the NRC staff finds that this analysis is bound by the control rod withdrawal at power event.

C.9 Overpressure Protection

The licensee stated that the limiting overpressure transient with respect to SG pressure is the turbine trip. An operability analysis, which was performed using the NRC-approved B&W safety analysis method, assumed an initial core power level of 3025 MWt, and demonstrated that the peak SG pressure was acceptable at this higher power level, which bounds the requested, uprated power level.

For RCS protection, however, the limiting transient is the Loss of Normal FW Flow; at an analyzed core power level of 102 percent current licensed thermal power, this analysis is acceptable for the requested MUR power uprate, as noted in Table 3.1. Based on the limiting nature of the Loss of Normal FW Flow event, and on the SG pressure data discussed by the licensee and in the preceding paragraph, the NRC staff finds the provisions for overpressure protection adequate to support the licensee's proposed MUR power uprate.

D. NI UNCERTAINTY

The licensee will retain an NI calibration tolerance of 2 percent in a non-conservative direction. Thus, if the licensee discovers, based on heat balance calibration, that the NI underpredicts core power level in excess of 2 percent, the NI must be recalibrated.

The NI provides input to the reactor trip system. This input is significant because it affects the maximum power level that the plant can achieve during transients that are assumed to be terminated by a high flux trip. If the NI underpredicts the core power level, then the plant can exceed the limits of its safety analyses.

At DBNPS, the trip setpoint analytic limits incorporate a margin for the nuclear instrumentation calibration. Because this margin is incorporated, the analyses of record demonstrate that those transients terminated by the high flux trip are terminated at an appropriate power level. The proposed uprate does not affect this margin. Therefore, because the high flux trip setpoints incorporate a margin for NI uncertainty that is unchanged, it is acceptable for the licensee to retain its 2 percent NI calibration criterion.

3.2.3 Summary

The NRC staff reviewed the reactor systems and thermal-hydraulic aspects of the proposed LAR in support of implementation of a measurement uncertainty recapture. Based on the considerations discussed above, the NRC staff determined that the results of the licensee's analyses related to these areas continue to meet applicable acceptance criteria following implementation of the MUR. Most of the current analyses of record are based on 2819 MWt that includes 2.0 percent measurement uncertainty. The proposed amendment is based on the use of a Caldon LEFM CheckPlus™ system that would decrease the uncertainty in the FW flow, thereby decreasing the power level measurement uncertainty from 2.0 percent to 0.37 percent.

In these cases, the proposed MUR rated thermal power of 2817 MWt is bounded by the current analyses of record.

As described in Section A above, the NRC staff finds that the hydraulic aspects of the Caldon LEFM CheckPlus™ UFM system have been accurately described in applicable documentation and that there is a firm theoretical and operational understanding of behavior. The NRC staff further finds that the calibration accomplished at ARL is appropriate for CheckPlus™ installation at DBNPS and is acceptable. Therefore, the NRC staff has concluded, based on the considerations discussed above, that the proposed changes are acceptable with respect to the hydraulic aspects of the CheckPlus™ UFM when installed at DBNPS.

As described in Section C above, the NRC reviewed those instances where the licensee's analyses of record were performed at the current licensed thermal power and determined that those analyses were either (1) bounded by the analysis at 100 percent original licensed thermal power or (2) bounded by a more limiting analysis performed at 102 percent original licensed thermal power. In those cases, the appropriate disposition has been noted.

3.3 Electrical Systems

3.3.1 Regulatory Evaluation

The licensee developed the LAR consistent with the guidelines in NRC RIS 2002-03.

The regulatory requirements which the NRC staff applied in its review of the application include:

General Design Criterion (GDC) 17, "Electric power systems," of 10 CFR Part 50, Appendix A requires that an onsite power system and an offsite electrical power system be provided with sufficient capacity and capability to permit functioning of SSCs important to safety.

Section 50.63 of 10 CFR requires that all nuclear plants have the capability to withstand a station blackout, as defined in 10 CFR 50.2, for an established period of time, and to recover therefrom.

Section 50.49 of 10 CFR, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," requires licensees to establish programs to qualify electric equipment important to safety.

3.3.2 Technical Evaluation

The NRC staff reviewed the licensee evaluation of the impact of an MUR power uprate on following electrical systems/components:

- Alternating Current (AC) Distribution System
- Power Block Equipment (Generator, Exciter, Transformers, Isolated-phase ("iso-phase") bus duct, Generator circuit breaker)
- Direct Current (DC) system
- Emergency Diesel Generators (EDGs)
- Switchyard
- Grid Stability
- Station Blackout
- Equipment Qualification Program

3.3.2.1 AC Distribution System

The AC Distribution System is the source of power to non-safety-related buses, and to safety-related emergency buses supplying the redundant engineered safety features loads. It consists of the 13.8 kilovolt (kV) system, the 4.16 kV system (not including the EDGs), the 480 volt (V) system and the 120V system.

The NRC staff requested information by letter dated July 25, 2007 (ADAMS Accession No. ML071790277), on any changes to the AC distribution system as a result of the power uprate.

By letter dated January 15, 2008, the licensee provided additional information regarding impacted system loads. Due to the uprate, the condensate pump motors (3) require an additional 10.6 kW per motor, resulting in an 0.9 percent increase in load. The heater drain pump motors (2) need an additional 4.9 kW per motor, equivalent to a 2 percent increase. The reactor coolant pump motors (4) require an additional 6.2 kW per motor, a 0.1 percent increase. The stator water cooling pump motors (2) require an additional 1.9 kW per motor, a 3 percent increase. As a result, the uprate will require an additional 70.2 kW which is well within the available margin of approximately 4.5 megawatts.

The NRC staff reviewed the licensee's response in their supplemental letter dated January 15, 2008. Based on the above information, the NRC staff finds that the analyses for the AC distribution system bound the MUR power uprate conditions.

3.3.2.2 Power Block Equipment (Generator, Exciter, Transformers, Iso-phase bus duct, Generator Circuit Breaker)

As a result of the power uprate, the rated thermal power will increase to 2817 MWt from the previously analyzed core power level of 2772 MWt.

In its letter dated July 25, 2007, the NRC staff requested information regarding the effect of the power uprate on the main generator, isophase bus, main power transformers, and the unit auxiliary/startup transformers. The licensee discussed the impact of the power uprate on the components in question in its letter dated September 18, 2007.

The licensee stated that all of the components in question will continue to operate below the maximum ratings after the uprate. Specifically, the main generator will continue to operate within the reactive capability curve and below the maximum real power of 1068 megawatt electric (MWe) at a 1.0 power factor (zero MVAR output). The load on the iso-phase bus will remain below the rated value of 25kV. The main transformer will continue to operate below the rated value of 980 megawatt ampere (MVA). For the startup transformers and the auxiliary transformers, the load remains under the maximum ratings of the transformers, 72.8 MVA and 77.653 MVA, respectively. The licensee stated that DBNPS does not have a main generator breaker, but instead has two 345 kV air-blast circuit breakers that connect the generator through the main transformer to the switchyard. Operating the generator such that the output does not exceed the ratings of the main transformer will also prevent the ratings of the 345 kV Air-Blast Circuit from being exceeded.

The NRC staff reviewed the information on generator step-up transformers, auxiliary transformers, startup transformers, iso-phase bus duct, and generator circuit breaker provided by the licensee in its letter dated September 18, 2007. The small increase in generator output does not cause overloading of the iso-phase bus duct or the generator step-up transformer.

Therefore, the NRC staff finds that the ratings of unit auxiliary transformers and startup transformers are not impacted by MUR power uprate conditions.

3.3.2.3 DC System

According to Attachment 2 of the LAR, the DC System is bounded by the existing analyses of record. The LAR states that the analysis demonstrates that the system has adequate capacity and capability to operate the plant equipment.

The station's 250/125 V DC System is comprised of batteries, battery chargers and distribution equipment that supply power to station loads. The nuclear safety-related (Class 1E) portion of the DC System consists of two 250/125 V DC motor control centers, four 125 V DC batteries, six battery chargers, four essential distribution panels, and four 480 V AC/125 V DC rectifiers. It provides the source of power for direct current load groups, vital control and instrumentation, power and control of Class 1E and selected non-Class 1E electrical equipment.

The NRC staff reviewed the LAR and UFSAR. There are no significant changes in the DC system loads. Based on the above information, the NRC staff finds that the analyses for the DC system bound the MUR power uprate conditions.

3.3.2.4 EDGs

In Attachment 2 of the LAR, the licensee states that the EDG System is bounded by generator loading tables, which are supported by the existing analysis of record. Both the bounding analysis and generator loading tables demonstrate that the system has adequate capacity and capability to power the safety-related loads.

The EDG System provides a safety-related source of AC power to sequentially energize and restart loads necessary to shutdown the reactor safely, and to maintain the reactor in a safe shutdown condition. The system is capable of performing this function during a loss of offsite power, with or without a coincident LOCA. There are two redundant EDG sets, each dedicated to one of the essential 4.16 kV buses.

There are no significant changes in the EDG system loads. Based on the above information, the NRC staff finds that the analyses for the EDG System bound the MUR power uprate conditions.

3.3.2.5 Switchyard

The switchyard equipment and associated components are classified as non-safety related. The primary function of the 345 kV switchyard and distribution system is to connect the station electrical system to the transmission grid. The interconnection allows for:

- the normal flow of power out of the station to the grid when the main generator is operating, and;
- the flow of power from the grid to the station auxiliaries when the main generator is shut down.

The small increase in plant output does not significantly impact the switchyard equipment. Based on the above information, the NRC staff finds that the analyses for the switchyard system bound the MUR power uprate conditions.

3.3.2.6 Grid Stability

Grid stability is discussed in Attachment A of the LAR for the MUR power uprate. In the LAR, the licensee referenced "Davis-Besse Stability Study" (ADAMS Accession No. ML020640288), which was performed to evaluate the system impact.

The study evaluated the steady-state and transient performance of the FirstEnergy (FE) System with both the existing DBNPS power level and the uprated power level. For this study, a 10 percent increase in gross power output was assumed. Both power flow and stability analyses were performed.

According to the impact study, the stability analysis was performed under the 1999/2000 summer peak load condition for fourteen contingencies. A 3-phase fault at the Bayshore 345 kV bus, Contingency 4, resulted in unstable system responses for the uprated system but stable conditions for the existing ratings. A 3-phase fault at DBNPS circuit breaker 34564, Contingency 8, resulted in unstable system response. The study states, "If the DBNPS uprate occurs, additional analysis is recommended to determine methods to improve system stability [for Contingencies 4 and 8]." The NRC staff requested information regarding any additional analyses and actions taken to ensure system stability for these two contingencies for a power uprate at 1.63 percent. In its letter dated January 15, 2008, the licensee stated that both Contingencies 4 and 8 are North American Electric Reliability Corporation (NERC) Type D events. A NERC Type D event is a 3 phase fault with a relay failure. According to NERC standard TPL-001, for these types of faults, the transmission owner must evaluate these faults, but there is no requirement for the system to be stable. Thus, the FE system meets all the stability requirements described in the NERC planning standards.

Since the impact study utilizes 1999/2000 summer peak loads, the NRC staff requested the licensee to show that the study, performed in May 2000, bounds current grid conditions. In its response dated January 15, 2008, the licensee stated that a 2005 study examined a selected set of scenarios in the FE - Midwest Independent Transmission System Operator footprint. Additionally, a 2007 study simulated stuck breaker and backup clearing faults on each line emanating from a power plant. In total, 260 faults were simulated with 14 associated with DBNPS and as a result, the FE system met all FE and NERC criteria. Although this study did not specifically model the power uprate, the uprate is within the accuracy of the model, and the results of the study remain valid for the MUR power uprate.

The impact study also addresses MVAR capability. Specifically, with a 10 percent power uprate, the change in power factor reduces the unit's reactive power capability by 67 MVAR. The NRC staff requested information on MVAR support, the MVAR contributions DBNPS is credited by the transmission system operator (TSO), and any compensatory measures taken to compensate for the depletion of MVARs on the system. In its September 18, 2007 letter, the licensee stated that MVAR contributions are not impacted by the power uprate as the generator will continue to operate within its capabilities. Furthermore, no compensatory measures are needed as a result of the power uprate. In regards to bus voltages, DBNPS will be asked to operate up to a 0.95 leading power factor if the grid voltage is higher than the plant's minimum voltage requirements.

The NRC staff reviewed the grid stability study, and finds that the DBNPS MUR power uprate allows for a stable and reliable grid operation.

3.3.2.7 SBO

According to Attachment 2 of the LAR, no additional station load is required under SBO conditions due to the power uprate. 10 CFR 50.63 requires that each light water-cooled nuclear power plant be able to withstand for a specified duration and recover from a SBO, which is defined as loss of all AC power to the essential and non-essential switchgear buses.

A non-class 1E SBO diesel generator (SBODG) is available to provide AC power to all systems required for coping with a SBO. DBNPS' SBO coping duration is four hours. This is based on an evaluation of the offsite power design characteristics, emergency AC power system configuration and EDG reliability in accordance with the evaluation procedure outlined in NUMARC 87-00 and RG 1.155. The offsite power design characteristics include the expected frequency of grid-related loss of offsite power, the estimated frequency of loss of offsite power from severe and extremely severe weather, and the independence of offsite power.

The MUR power uprate does not impact the offsite power design characteristics, modify the emergency AC power system configuration or affect the EDG reliability. Considering this, the NRC staff agrees that the MUR power uprate will have no impact on DBNPS's SBO coping duration. In addition, the SBODG is not impacted by the MUR power uprate. Based on the above information, the NRC staff finds that DBNPS will continue to meet the requirements of 10 CFR 50.63.

3.3.2.8 Equipment Qualification (EQ) Program

According to Attachment 2 of the LAR, the MUR power uprate does not change the accident/post-accident temperature profiles inside containment. Therefore, there is no impact on environmentally qualified equipment. The existing calculated temperature and pressure profiles bound the power uprate of 1.63 percent.

In the LAR, the licensee stated that the EQ of electrical equipment was performed at a core power level of 102 percent of 2772 MWt, which bounds the MUR operating conditions. Considering this, the NRC staff agrees that the MUR power uprate will have no impact on DBNPS's EQ of electrical equipment.

3.3.3 Overall Summary

Based on technical evaluation provided above, the NRC staff agrees that the MUR power uprate will continue to meet the applicable requirements of GDC 17, 10 CFR 50.63, and 10 CFR 50.49. Therefore, with respect to the Electrical Systems, the NRC staff finds that the MUR power uprate is considered acceptable.

3.4 Mechanical and Civil Engineering

3.4.1 Regulatory Evaluation

The NRC staff's review in the area of mechanical engineering covers the structural and pressure boundary integrity of NSSS and balance-of-plant (BOP) systems and components. This review focuses on the impact of the proposed MUR power uprate on (1) NSSS piping, components, and supports; (2) BOP piping, components, and supports; (3) the reactor vessel (RV) and its supports; (4) control rod drive mechanisms (CRDMs); (5) the once-through steam generator (OTSG) and its supports; (6) RCPs and supports; (7) the pressurizer and its supports; (8) reactor internals and core supports; and (9) safety related valves (SRVs). Technical areas covered by this review include stresses, cumulative usage factors (CUFs), flow-induced vibration, high-energy line break locations, jet impingement and thrust forces, and safety-related valve programs.

These piping systems, components and their supports, including core support structures, are designed in accordance with the rules of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section III, and ASA/USAS/ANSI B31.7 & B31.1. The NRC staff's evaluation considered GDC 1, 2, 4, 10, 14, and 15. The NRC staff's review focused on verifying that the licensee has provided reasonable assurance of the structural and functional integrity of piping systems, components, component internals, and their supports under normal and vibratory loadings, including those due to fluid flow, postulated accidents, and natural phenomena such as earthquakes.

The acceptance criteria are based on continued conformance with the requirements of the following regulations: (1) 10 CFR 50.55a, and GDC 1 as they relate to structures, systems and components (SSCs) being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed; (2) GDC 2 as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC 4 as it relates to structures and components important to safety being designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal and accident conditions; (4) GDC 10 as it relates to reactor internals being designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences; (5) GDC 14 as it relates to the reactor coolant pressure boundary being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; and (6) GDC 15 as it relates to the RCS being designed with sufficient margin to ensure that the design conditions are not exceeded.

The specific review areas are contained in the NRC Standard Review Plan (SRP) Section 3.9. The review also includes the plant-specific provisions of Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," and GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," as related to plant-specific program for motor-operated valves, GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," as related to the pressure locking and thermal binding for safety-related gate valves, and the plant-specific evaluation of the GL 96-06, program regarding the over-pressurization of isolated piping segments.

3.4.2 Technical Evaluation

The NRC staff reviewed the DBNPS MUR power uprate amendment request dated April 12, 2007. The review focused on the effects of power uprate on the structural and pressure boundary integrity of piping systems and components, their supports, and reactor vessel and internal components, the CRDMs, and the BOP piping systems.

The proposed 1.63 percent power uprate will increase the RTP level from 2772 MWt to 2817 MWt. The power uprate will be achieved by an increase in steam flow and FW flow, and an increase in the temperature difference across the core. The RCS pressure, RCS average temperature, FW temperature, and OTSG steam pressure will remain the same.

Table VIII.6-1 on Page 338 of Attachment A to the licensee's April 12, 2007, submittal shows the pertinent temperatures, pressures, and flow rates for the current conditions and the uprated conditions. At full power, the hot-leg temperature increases from 606.1 to 607.4 degrees Fahrenheit, the cold-leg temperature decreases from 557.9 to 556.6 ^{o}F, the OTSG pressure remains unchanged at 930.0 psia, the steam flow increases from 11.65 to 11.99 million pounds per hour (Mlbm/hr), the FW temperature remains unchanged at 455.0 ^{o}F, and the FW flow increases from 11.65 to 11.99 Mlbm/hr. The proposed uprate does not change heat up or cool down rates or the number of cycles assumed in the design analyses. In addition, there are no changes in the design transients since the safety analyses were performed at 102 percent of RTP. Thus, the limiting analyses are still bounding.

The design parameters for the RCS and SG are found in Tables 5.1-1a and 5.1-5, respectively, of the DBNPS UFSAR. The RCS components, including the reactor vessel, core support structures, and OTSG, were designed to operate at a core power level of 2827 MWt. The RCS components are designed to 650 ^{o}F (except the pressurizer, which is designed to 670 ^{o}F) and 2,500 psig. The OTSG is designed for a steam flow of 6.12 Mlbm/hr (UFSAR Table 5.1-5). The FW system design temperature is 470 ^{o}F (Licensee Submittal Attachment A, Table VIII.6-1).

3.4.2.1 Reactor Pressure Vessel (RPV) and Internals

The Code of Record for the reactor vessel, nozzles, and supports is ASME Code, Section III, 1968 Edition with addenda through the Summer 1968 Addenda. The licensee compared the expected temperatures and pressures for the proposed power uprate condition against the analysis of record. The licensee confirmed in its submittal that there is no change in RCS design or operating pressure, and the effects of operating temperature changes for cold and hot legs are within design limits. The MUR power uprate conditions are bounded by the design conditions. In addition, the operating transients will not change as a result of the power uprate and no additional transients have been proposed. The existing loads, stresses, and fatigue CUF values for reactor vessel and internals remain valid for the proposed power uprate. Therefore, the NRC staff concurs with the licensee's assessment that the RPV and internals are acceptable for operation at the uprated power level.

3.4.2.2 CRDMs

The Code of Record for the pressure retaining components of the CRDMs is the ASME Code, Section III, 1968 Edition with addenda through Summer 1968 (part length) and Summer 1970 (full length). The licensee confirmed in its submittal that the temperatures and pressures used in the CRDM design analyses continue to bound the conditions at the proposed uprated power level. In addition, the operating transients will not change as a result of the power uprate and no additional transients have been proposed. The existing loads, stresses, and fatigue CUF values for CRDM components remain valid for the proposed power uprate. Therefore, the NRC staff concurs with the licensee's assessment that the CRDMs are acceptable for operation at the uprated power level.

3.4.2.3 Reactor Coolant Piping and Components

The RCS piping was designed to ANSI B31.7 Draft, February 1968 Edition with errata June 1968. The licensee reviewed the revised design conditions for impact on the existing design basis analyses for the reactor coolant piping and supports. The licensee stated that there is no change in RCS design or operating pressure, and the effects of operating temperature changes for cold and hot legs are within design limits. The MUR power uprate conditions are bounded by the design conditions. In addition, the operating transients will not change as a result of the power uprate and no additional transients have been proposed. The existing loads, stresses, and fatigue CUF values for RCS piping and supports remain valid for the proposed power uprate. Therefore, the NRC staff concurs with the licensee's conclusion that the reactor coolant piping and supports are acceptable for operation at MUR power uprate level.

The OTSG was designed to the ASME Code, Section III, 1968 Edition with addenda through Summer 1968. The licensee reviewed the revised design conditions for impact on the existing design basis analyses for the OTSG including the OTSG tubes, secondary side internal support structures, shell and nozzles. There is a negligible change in RCS mass flow, and the RCS temperatures and pressures used in the design continue to bound the uprate conditions. There is an increase in the steam flow and FW flow. The steam and FW pressures and flow rates used in the design of the OTSG continue to bound the expected uprate conditions. In addition, the operating transients will not change as a result of the power uprate and no additional transients have been proposed. Also, since the design of the OTSG included modeling of flow-induced vibration and the steam and FW flow rates remain bounded by the design flow rates, the licensee concluded that the MUR uprate will have no effect on flow-induced vibration.

The licensee stated that the existing tube loads due to LOCA, MSLB, and FW line break will not change as a result of the power uprate. The existing loads, stresses, and fatigue CUF values for the OTSG remain valid for the proposed MUR uprate.

The pressure retaining parts of the RCPs were designed in accordance with the ASME Code, Section III, 1968 Edition with addenda through Winter 1968. The licensee reviewed the revised design conditions for impact on the existing design basis analyses for the RCPs. It was stated that the temperature changes due to the MUR uprate are bounded by those used in the existing analyses. In addition, the operating transients will not change as a result of the power uprate and no additional transients have been proposed. The existing loads, stresses, and fatigue CUF values for RCPs remain valid for the proposed power uprate.

The Code of Record for the pressurizer, including the nozzles, is the ASME B&PVC, Section III, 1968 Edition with addenda through Summer 1968. The licensee reviewed the revised design conditions for impact on the existing design-basis analyses for the pressurizer. It was stated that the temperature changes due to the MUR power uprate are bounded by those used in the existing analyses. In addition, the operating transients will not change as a result of the power uprate and no additional transients have been proposed. The existing loads, stresses, and fatigue CUF values for pressurizer remain valid for the proposed power uprate.

The licensee reviewed the potential for thermal stratification (NRC Bulletin 88-11). The temperature changes and RCS mass flow rate change as a result of MUR are negligible, and the licensee concluded that the effects on thermal stratification will not change as a result of this power uprate.

Based on the above, the NRC staff agrees with the licensee's conclusion that the design of piping, components, including the SGs, RCPs, and pressurizer, and their supports, is adequate to maintain the structural and pressure boundary integrity of the reactor coolant loop because the analyses of record parameters are bounding for the proposed 1.63 percent power uprate condition.

3.4.2.4 High Energy Line Break (HELB) Locations

The licensee stated that an engineering evaluation was performed to determine the impact of power uprate on HELB systems. The HELB evaluations were performed at 2827 MWt (102 percent of RTP) to bound the expected range of operation resulting from the MUR uprate. There are no new line breaks postulated for current HELB systems. The licensee stated that the impact of the MUR uprate on HELB systems remains bounded by the values in existing analyses. Also, there are no new systems that qualify as HELB systems as a result of the uprate. Therefore, the NRC staff agrees with the licensee's conclusion regarding HELB.

3.4.2.5 BOP Piping (NSSS Interface Systems, Safety Related Cooling Water Systems, and Containment Systems) and SRVs

The licensee evaluated the BOP piping systems by comparing the conditions for the proposed power uprate with the analysis of record conditions and the current operating conditions. The BOP piping components were designed to the ASME B&PVC, Section III, 1971 Edition or the USAS B31.1, Power Piping Code, 1967 Edition with addenda. The licensee stated that the revised design conditions were reviewed for impact on the existing design-basis analyses for the BOP Piping. The licensee confirmed in its submittal that there is no change in RCS design or operating pressure, and the effects of operating temperature changes for cold and hot legs are within design limits. The MUR power uprate conditions are bounded by the design conditions. Since the operating transients will not change as a result of the power uprate and no additional transients have been proposed, the licensee concluded that the existing loads, stresses, and fatigue CUF values remain valid.

The licensee stated that the revised design conditions were reviewed for impact on the existing design basis analyses for the SRVs and showed that the temperature changes due to the MUR uprate are bounded by those used in the existing analyses. Safety analyses confirmed that the installed capacities and lift set points of the RCS and MSRVs continue to be valid for the MUR conditions. None of the SRVs required a change to their design or operation as a result of the MUR. The existing loads, stresses, and fatigue CUF values remain valid. The licensee did not identify any changes to the plant-specific provisions of GL 89-10 and GL 96-05, related to motor operated valves, GL 95-07, related to pressure locking and thermal binding of safety-related gate valves, or GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," related to over-pressurization of isolated piping segments. The NRC staff does not anticipate any changes to the analysis of over-pressurization of isolated piping segments because the analysis of record for containment temperature and pressure was performed at 102 percent of current RTP and remains bounding for the uprate conditions. Therefore, the NRC staff does not expect any changes to the plant-specific provisions of GL 89-10, GL 96-05, GL 95-07, or GL 96-06.

The licensee concluded that the DBNPS BOP piping systems remain acceptable for operation at the uprated conditions. Based on the above, the NRC staff agrees with the licensee's conclusion that the proposed 1.63 percent power uprate will not have adverse effects on BOP systems including safety-related valves.

3.4.3 Summary

The NRC staff has reviewed FENOC's assessment of the impact of the proposed MUR power uprate on NSSS and BOP systems and components with regard to stresses, CUFs, and safety related valve programs. On the basis of this review described above, the NRC staff finds that the proposed MUR power uprate will not have an adverse impact on the structural integrity of the piping systems, components, their supports, reactor internals, core support structures, CRDMs, BOP piping, or safety-related valves.

3.5 Component Performance and Testing

3.5.1 Effect of Power Uprate on Major Components

3.5.1.1 CRDM

The licensee reviewed the impact on the existing CRDM design-basis analysis for the MUR conditions. No changes in the RCS design or operating pressure were made as part of MUR uprate. The effect of operating temperature changes (T_{hot}/T_{cold}) are within the design limits. The design conditions in the existing analyses are based on the RCS functional specification. The licensee further stated that the MUR conditions are bounded by the design conditions. Since the operating transients will not change as a result of the MUR power uprate and no additional transients have been developed, the existing loads, stresses, and fatigue values remain valid. Thus, the staff concludes that the existing stress values for the CRDMs remain applicable for the MUR conditions, and that the existing CRDM design-basis analyses support the MUR power uprate.

3.5.1.2 Safety-Related Valves

The NRC staff reviewed the licensee's safety-related valves analysis. The NRC's acceptance criteria for reviewing the safety-related valves analysis are based on the applicable portions of 10 CFR 50.55a. Additional information is also provided by the plant-specific evaluations of GL 95-07, GL 96-06, GL 89-10, and GL 96-05. The licensee reviewed the impact of the proposed MUR conditions on the existing design-basis analyses for the safety-related valves. The evaluation showed that the temperature changes due to MUR uprate are insignificant and bounded by those used in the existing analyses, and no changes in RCS design or operating pressure are made as part of MUR uprate. The licensee's safety analysis also confirmed the installed capacities and lift setpoints of the RCS and MSRVs remain valid for the MUR conditions. Due to the insignificant changes in temperature and operating pressure, none of the SRVs required a change to their design or operation as a result of the MUR uprate. Therefore, the NRC staff finds the performance of existing safety-related valves acceptable with respect to the MUR power uprate.

3.6 Steam Generator Tube Integrity and Chemical Engineering

3.6.1 Makeup and Purification System

3.6.1.1 Regulatory Evaluation

The Makeup and Purification System is the Davis-Besse plant-specific system that corresponds to the chemical and volume control system. The Makeup and Purification System provides a means to (1) control the Reactor Coolant System (RCS) inventory; (2) receive, purify, and recirculate reactor coolant water; (3) maintain the required boron concentration in the RCS; (4) provide seal injection water for the four reactor coolant pumps; (5) accommodate changes in reactor coolant volume due to small temperature changes along with the pressurizer; (6) maintain proper hydrogen and hydrazine concentrations for oxygen control; (7) maintain proper lithium concentration for pH control; (8) provide makeup to the RCS for protection against small breaks in the RCS pressure boundary; and (9) degas the RCS. The staff has reviewed the safety-related functional performance characteristics of the Makeup and Purification System components. The NRC's acceptance criteria are based on (1) GDC 14, "Reactor Coolant Pressure Boundary (RCPB)," as it requires that the RCPB be designed to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture; and

(2) GDC 29, "Protection Against Anticipated Operational Occurrences," as it requires that the reactivity control systems be designed to assure an extremely high probability of accomplishing their functions in the event of condenser in-leakage or primary-to-secondary leakage.

3.6.1.2 Technical Evaluation

During plant operation, reactor coolant is taken from one of the reactor inlet lines and cooled during passage through letdown cooler(s). The reactor coolant passes from the containment vessel through a containment isolation valve, through the letdown flow control station where it is reduced in pressure, and then passed through a purification demineralizer.

Under power uprate conditions, the licensee indicated that the hot-leg and cold-leg temperatures will change by 0.4 ^{o}F. This will result in a slightly lower temperature for the letdown line. The licensee concluded that the slightly lower temperature of the letdown line does not affect the performance of the letdown coolers because they remain bounded by current operation. In addition, the licensee reported that no changes to the Makeup System configuration are required under power uprate conditions.

On the basis of its review, the staff concludes that the Makeup and Purification System is adequate because the proposed MUR power uprate will introduce negligible changes in the Makeup and Purification System's operating parameters, which will not affect satisfactory performance of its intended functions, and it will continue to operate within its design limits under the uprate conditions. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the Makeup and Purification System.

3.6.2 Steam Generator Blowdown System (SGBS)

3.6.2.1 Regulatory Evaluation

Control of secondary-side water chemistry is important for preventing degradation of SG tubes. The SGBS provides a means for removing SG secondary-side impurities and, thus, assists in maintaining acceptable secondary-side water chemistry in the SGs. The design basis of the SGBS includes consideration of expected design flows for all modes of operation. The SGBS for Davis-Besse is not operational above approximately 15 percent reactor power, and is removed from service above that power (operational only during startup and shutdown). The NRC staff reviewed the ability of the SGBS to remove particulate and dissolved impurities from the SG secondary-side at power levels at or below about 15 percent. The NRC's acceptance criteria for the SGBS are based on GDC 14, "Reactor Coolant Pressure Boundary," as it requires that the RCPB be designed so as to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture. Specific review criteria are contained in SRP Section 10.4.8, "Steam Generator Blowdown System (PWR)."

3.6.2.2 Technical Evaluation

The SGBS is designed to extract blowdown water from the secondary side of the SGs as a means of removing particulates and dissolved solids to control water chemistry in steam generators. The Davis-Besse SGBS discharges to the condenser. The SGBS also provides samples of the secondary side water in the SG. These samples are used for monitoring water chemistry and for detecting the amount of radioactive primary coolant leakage through the SG tubes. Proper control of SG secondary side chemistry reduces the probability of secondary side initiated SG tube degradation.

The licensee indicated that the MUR conditions, pressure, temperature, and flow, were bounded by the design conditions used in the DBNPS flow-accelerated corrosion (FAC) program. The plant chemistry requirements are not being revised for the MUR and the potential increase of impurities in the SG water will be accommodated by the plant condensate demineralizer system. Since the power uprate does not affect the pressure at which the SGs are controlled, there will be no effect on the blowdown system.

On the basis of its review, the NRC staff concludes that the SGBS is adequate because the blowdown flow, the SG secondary-water chemistry, and the blowdown pressures and temperatures remain within the original system design. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the SGBS.

3.6.3 FAC Program

3.6.3.1 Regulatory Evaluation

FAC is a corrosion mechanism occurring in carbon steel components exposed to single-phase or two-phase water flow. Components made from stainless steel are immune to FAC, and FAC is significantly reduced in components containing small amounts of chromium or molybdenum. The rates of material loss due to FAC depend on flow velocity, fluid temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and the optimum conditions for minimizing FAC effects, in most cases, cannot be achieved. Loss of material by FAC is, therefore, likely to occur. The NRC staff reviewed the effects of the proposed MUR power uprate on FAC and the adequacy of the licensee's FAC program to predict the rate of loss so that repair or replacement of damaged components could be made before reaching critical minimum thickness.

3.6.3.2 Technical Evaluation

FAC is a corrosion mechanism occurring in carbon steel components exposed to flowing single or two-phase water. FAC results in wall thinning and possible failure of high energy carbon steel pipes in the power conversion system. Components made from stainless steel are immune to FAC, and FAC is significantly reduced in components containing small amounts of chromium or molybdenum. The rates of material loss by FAC depend on velocity of flow, temperature, steam quality, oxygen content, and pH. During plant operation, control of these parameters is limited and the optimum conditions for minimizing FAC effects, in most cases, cannot be achieved. Loss of material by FAC is, therefore, likely to occur. Since undesirable challenges to the plant's safety systems may result from piping system component failure, licensees maintain a FAC-related failure prediction, inspection, and component repair/replacement program.

The licensee evaluated component wear rates and reductions in service lives at a higher power level (~8.8 percent) using the EPRI CHECKWORKS FAC monitoring program. The licensee concluded, based on their evaluation for an 8.8 percent power increase, that the impacts on wear rates and service lives were small and could continue to be managed through the DBNPS

FAC program. In addition, the licensee concluded the increases in pressure, temperature, and flow in the affected systems for the MUR power uprate are bounded by the previous evaluation.

On the basis of its review, the NRC staff concludes that the FAC program is acceptable for operating conditions because the effect of the power uprate on the operating parameters that affect FAC rates is expected to be small and will be adequately managed by the existing FAC program. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the FAC program.

3.6.4 Coatings

3.6.4.1 Regulatory Evaluation

Protective coatings (paints) inside containment are used to protect equipment and structures from corrosion and contamination from radionuclides and also provide wear protection during plant operation and maintenance activities. The coatings are subject to 10 CFR Part 50 Appendix B. The NRC staff reviewed whether the pressure and temperature conditions under the power uprate continue to be bounded by the conditions to which the coatings were qualified.

3.6.4.2 Technical Evaluation

Protective coatings (paints) inside containment are used to protect equipment and structures from corrosion and contamination from radionuclides and also provide wear protection during plant operation and maintenance activities.

The licensee evaluated the design basis accident (DBA) testing for Service I protective coating systems and concluded that the design basis accident temperature and pressure profile under power uprate conditions remain bounded by the design basis accident temperature and pressure profiles.

On the basis of its review, the NRC staff concludes that the coatings will not be adversely impacted by the MUR power uprate and temperature and pressure conditions as they will continue to be bounded by the conditions to which the coatings were qualified. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the coatings.

3.6.5 SG Program

3.6.5.1 Regulatory Evaluation

SG tubes constitute a large part of the RCPB. The NRC staff reviewed the effects of changes in operating parameters (e.g., pressure, temperature, and flow velocities) resulting from the proposed power uprate on the design and operation of the SGs. Specifically, the NRC staff evaluated whether changes to these parameters continue to be bounded by those considered in the plant design and licensing basis (i.e., the TS plugging limits).

3.6.5.2 Technical Evaluation

DBNPS has two B&W OTSGs designated SG “1B” and SG “2A.” Each SG has more than 15,000 Alloy 600 tubes in the mill-annealed condition. The tubes have a nominal outside diameter of 0.625 inches and a nominal wall thickness of 0.037 inches. Both SG “1B” and “2A” contain tubes with sleeves and tubes with shop re-rolls.

The licensee indicated that they reviewed the design and operational functions of the SGs and concluded that the SGs will continue to satisfy all design and operational functions under power uprate conditions. The licensee evaluated the impact of the power uprate conditions on the existing qualification reports and design calculations for the mechanical plugs, welded plugs, tube sleeves, and tube stabilizers. The licensee concluded that the tube repair product qualifications and design remain valid and they maintain functional integrity. The licensee confirmed that the plugging limit continues to be appropriate for power uprate conditions according to the guidance in RG 1.121. The licensee performed an evaluation to address flow-induced vibration (FIV) under power uprate conditions and the FIV impact on the SG tube bundle and installed tube repair hardware. The licensee concluded that the tube bundle will not fail due to high-cycle fatigue, tube-to-tube impacts will not occur over the life of the plant, and that all installed tube repair hardware will maintain functional integrity. The licensee also indicated that tube wear from loose parts is not expected due to power uprate conditions given there is no evidence of any secondary-side loose parts in the SGs.

On the basis of its review, the NRC staff concludes that the power uprate is acceptable from a SG design and inservice inspection perspective because the power uprate is expected to introduce only negligible changes in the SG operating parameters, which will not significantly affect the performance of the SGs, and the SGs will continue to operate within their design limits under uprate conditions. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the SGs.

3.6.6 Overall Summary

In the areas of SGs and Chemical Engineering, the NRC staff concludes that for the MUR power uprate, the licensee has adequately addressed (1) the changes of the reactor coolant and their effect on the Makeup and Purification System, (2) the changes in the system flow and impurity levels and their effects on the SGBS, (3) the changes in the plant operating conditions on the FAC program, (4) the effects on the protective coatings, and (5) the changes in the SGs operating parameters, the effects on the SGs and the determination that the SG tube integrity will continue to be maintained.

3.7 Operator Licensing and Human Performance

3.7.1 Regulatory Evaluation

NRC's human factors reviews address programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC staff's human factors evaluation was conducted to confirm that operator performance would not be adversely affected as a result of system and procedure changes made to implement the proposed MUR power uprate. The scope of the review included changes to operator actions, human-system interfaces, and procedures and training needed for the proposed MUR power uprate. The NRC staff also used the guidance provided in RIS 2002-03, Section VII in their review.

3.7.2 Technical Evaluation

The NRC staff has developed a standard set of questions for human factors reviews (RIS 2002-03, Attachment 1, Section VII, Items 1 through 4) of MURs. The following sections evaluate the licensee's responses to these questions in the LAR and additional clarifications in its RAI response.

3.7.2.1 Operator Actions

The licensee stated in its the submittal and RAI response that the proposed MUR power uprate will not require operators to take additional time in performing most actions credited in the UFSAR. The licensee also stated that the proposed MUR will not require any new operator actions or changes to existing operator actions. The analyses of both the overall single train cooldown and normal cooldown were made to address the effects of the MUR power uprate.

After analyzing the new MUR-based power level of 2817 MWt, the licensee concluded that normal cooldown would be extended from 24 hours to 26 hours using the existing methodology and equipment. For the overall single train cooldown, the licensee concluded that the time to fully cooldown would be extended from 168 hours to 175 hours. These actions are considered to be longer term actions and not time critical to immediate operator actions for UFSAR Chapter 15 events that lead to RCS cooldown. The licensee's review of all other operator actions for DBNPS sensitive to the MUR power uprate concluded that those operator actions will continue to be bounded and supported by the current UFSAR Chapter 15 analyses.

The NRC staff has reviewed the licensee's statements in the original submittal and responses to the RAI relating to any impacts of the MUR power uprate to existing or new operator actions. The NRC staff concludes that the proposed MUR power uprate will not adversely impact operator actions and their response times. The time changes to both cooldown activities are long term operator actions and will not impact the operator's ability to address plant mitigation activities during those times. The NRC staff finds that the statements provided by the licensee are in conformance with Section VII.1 of Attachment 1 to RIS 2002-03.

3.7.2.2 Emergency and Abnormal Operating Procedures

The licensee stated in its submittal that the current emergency operating procedures (EOPs) and abnormal operating procedures (AOPs) were reviewed for potential changes due to the proposed MUR power uprate. The licensee concluded that no changes to operator actions in both the EOPs and AOPs were needed and the available times to perform operator actions would remain unchanged. The revisions to the EOPs and AOPs are being done to reflect the higher power level and minor setpoint changes, which will be made prior to MUR implementation.

The NRC staff has reviewed the licensee's evaluation of the effects of the MUR power uprate on DBNPS's EOPs and AOPs. The NRC staff concludes that the proposed MUR power uprate does not present any adverse impacts on the EOPs and AOPs. This conclusion is based upon: (1) the licensee making revisions to the EOPs and AOPs that will reflect the new power level and revised setpoints, and (2) the minor changes being made to the EOPs and AOPs will be reflected in the operator training program prior to MUR implementation. The NRC staff finds that the statements provided by the licensee are in conformance with Sections VII.2.A, VII.3, and VII.4 of Attachment 1 to RIS 2002-03.

3.7.2.3 Changes to Control Room Controls, Displays, and Alarms

In its submittal, the licensee described changes to control room controls, displays (including the Safety Parameter Display System (SPDS)), and alarms related to the proposed MUR power uprate. The licensee also provided supplemental information related to these changes in the RAI response. Notable proposed changes to controls, displays, and alarms include:

- Setpoint changes will be made in the RPS for establishing trip parameters to ensure that safety margins are maintained during normal and transient plant operations.

- The functions for the Incore Monitoring Detector System will require changes to the plant computer software. These changes will be transparent to the operators and their responses to abnormal indications by the software will remain unchanged.

- The SPDS will be revised to display the trending performance of the Caldon LEFM system parameters.

- 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.

The licensee has stated that all other changes to the control room displays, alarms, and controls will reflect the increased power level, but will not impact the operator's ability to perform operator actions as well as the available times needed to complete certain operator actions during accident scenarios. All changes to the control room, including modifications involving the Caldon LEFM System, will be reflected in the operator training program prior to MUR implementation.

The NRC staff has reviewed the licensee's evaluation and proposed changes to the control room. The NRC staff concludes that the proposed changes do not present any adverse effects to the operators' functions in the control room. The licensee committed to making all modifications to the control room and providing training on these changes prior to MUR power uprate implementation. The NRC staff finds that the statements provided by FENOC are in conformance with Sections VII.2.B and VII.3 of Attachment 1 to RIS 2002-03.

3.7.2.4 Control Room Plant Reference Simulator and Operator Training

The licensee stated that the plant simulator will reflect the control room changes to be made due to the MUR power uprate. This includes the modifications of displays, controls, and alarms that currently mimic the actual control room. The verification and validation of the plant simulator will be performed prior to MUR implementation consistent with the licensee's current practice of simulator configuration control. This includes all engineering changes and simulator work orders to be associated with the MUR power uprate. Specific certification tests will be performed on the plant simulator prior to usage for operator training to verify the fidelity of the actual plant configuration and the plant simulator under MUR power uprate conditions.

The licensee also stated that the changes made to the EOPs and AOPs and the plant cooldown process will be included in the operator training program, along with the control room modifications. These changes, along with the plant simulator modifications, will be made prior to MUR implementation.

The NRC staff has reviewed the licensee's proposed changes to the operator training and plant simulator related to the MUR power uprate. The NRC staff concludes that the changes do not present any adverse effects on the plant simulator or the operator training program. The licensee committed to making all modifications to the plant simulator and incorporating these changes into the operator training program prior to MUR power uprate implementation. The NRC staff finds that the statements provided by the licensee are in conformance with Sections VII.2.C, VII.2.D, and VII.3 of Attachment 1 to RIS 2002-03.

3.7.3 Summary

The NRC staff has completed its human factors review of the licensee's proposed changes and concludes that the licensee has adequately considered the impact of the proposed MUR power uprate on operator actions, EOPs and AOPs, control room components, the plant simulator and operator training programs.

3.8 Vessels and Internals Integrity

The NRC staff's review in the area of RV and RV internals integrity focuses on the impact of the proposed MUR power uprate on pressurized thermal shock (PTS) calculations, fluence evaluations, heatup and cooldown pressure-temperature (P-T) limit curves, low-temperature overpressure protection (LTOP), upper shelf energy (USE), surveillance capsule withdrawal schedules, the pressurize shell, and RV internals. This review is conducted to verify that the results of licensee analyses related to these areas continue to meet the requirements of 10 CFR 50.60, 10 CFR 50.61, 10 CFR 50.55a, and Appendices G and H to Part 50, following implementation of the proposed MUR power uprate. The guidance contained in RIS 2002-03 has also been used by the NRC staff to conduct this review.

3.8.1 Reactor Vessel (RV) Material Surveillance Program

3.8.1.1 Regulatory Evaluation

The RV material surveillance program provides a means for determining and monitoring the fracture toughness of the RV beltline materials to support analyses for ensuring the structural integrity of the ferritic components of the RV. Appendix H of 10 CFR Part 50 provides the NRC staff's requirements for the design and implementation of the RV material surveillance program. The NRC staff's review primarily focused on the effects of the proposed MUR power uprate on the licensee's RV surveillance capsule withdrawal schedule.

3.8.1.2 Technical Evaluation

Regarding the RV surveillance program and capsule withdrawal schedule, the licensee concluded in Section IV.1.C.vi of Enclosure 2 to the LAR that:

> 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.

The licensee's RV material surveillance program is an integrated program designed by the B&W Owners Group, which is now a part of the Pressurized Water Reactor Owners Group (PWROG), for all seven operating B&W-designed 177-fuel assembly plants and six participating Westinghouse plants having B&W-fabricated reactor vessels. The program, which contains capsule withdrawal schedules, is revised periodically. The most recent version, documented in BAW-1543 (NP), Revision 4, Supplement 6, "Supplement to the Master Integrated Reactor Vessel Surveillance Program [MIRVSP]," was approved by the NRC in an SE dated June 28, 2007. This SE stated that the proposed withdrawal schedules satisfy the American Society for Testing and Materials Standard E185-82 for all plants participating in this PWROG MIRVSP, except for Turkey Point, Units 3 and 4. Table IX of Supplement 6 indicated that the peak end-of-license (EOL), i.e., 32 effective full power years (EFPY), inside diameter (ID) fluence for the DBNPS RV is 1.07×10^{19} neutrons per square centimeter (n/cm^2) (E>1.0 MeV). The NRC staff confirmed that this fluence value has not been revised by the licensee since 1992. The corresponding fluence, considering the MUR power uprate, was reported in Section IV.1.C.v of Enclosure 2 of the LAR to be 1.02×10^{19} n/cm^2 (E>1.0 MeV).

3.8.1.3 Summary

Based on the above, the NRC staff determined that the slight change of the EOL ID fluence from 1.07×10^{19} n/cm^2 (E>1.0 MeV) to 1.02×10^{19} n/cm^2 (E>1.0 MeV) will have essentially no impact on the EOL transition temperature shift values and, therefore, on the capsule withdrawal schedule of MIRVSP approved by the NRC on June 28, 2007. The licensee's response dated September 18, 2007, to the NRC staff's RAI confirmed that this ID fluence from 2006 is the latest applicable fluence projection. Therefore, the NRC staff determined that the DBNPS RV surveillance program would continue to meet the requirements of 10 CFR Part 50, Appendix H under the MUR power uprate condition.

3.8.2 P-T Limits and USE

3.8.2.1 Regulatory Evaluation

Part 50 of 10 CFR, Appendix G, provides fracture toughness requirements for ferritic (low alloy steel or carbon steel) materials in the RCPB, including requirements on the USE values used for assessing the safety margins of the RV materials against ductile tearing and for calculating P-T limits for the plant. These P-T limits are established to ensure the structural integrity of the ferritic components of the RCPB during any condition of normal operation, including anticipated operational occurrences and hydrostatic tests. The NRC staff's review of the USE assessments covered the impact of the MUR power uprate on the neutron fluence values for the RV beltline materials and the USE values for the RV materials through the end of the current licensed operating period. The NRC staff's P-T limits review covered the P-T limits methodology and the calculations for the number of the EFPY specified for the proposed MUR power uprate, considering neutron embrittlement effects.

3.8.2.2 Technical Evaluation

Regarding the topic of the RV P-T limits, the licensee concluded in Section IV.1.C.iii of Enclosure 2 to the submittal that:

> The results of the conservative fluence projection [performed in 2006] 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 DBNPS P-T curves and LTOP limits for 21 EFPY remain valid.

The DBNPS TSs contain 21 EFPY P-T limit curves. These curves are based on a then projected peak RV fluence of 7.25×10^{18} n/cm^2 (E>1.0 MeV). The NRC staff confirmed that this 21 EFPY ID fluence is consistent with that in the SE dated July 20, 1995, approving the current P-T limits. The MUR power uprate projected fluence at the end of Cycle 16 is 6.86×10^{18} n/cm^2 (E>1.0 MeV). Since the MUR power uprate fluence is bounded by the current P-T limit fluence at 21 EFPY, the MUR power uprate has no impact on the current P-T limit curves. Hence, the NRC staff confirmed that the DBNPS RV materials would continue to meet the requirements of 10 CFR Part 50, Appendix G, under the MUR power uprate condition.

Regarding the topic of the RPV USE, the licensee concluded in Section IV.1.C.v of Enclosure 2 to the submittal that:

> 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.

The NRC staff has evaluated the information provided by the licensee in the submittal as well as information regarding the equivalent margins analysis contained in BAW-2192-P-A, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Level A & B Service Loads." The BAW-2192-P-A equivalent margins analysis was based on an EOL fluence of 1.07×10^{19} n/cm^2 (E>1.0 MeV) for the limiting RV material, Weld WF-182-1, consistent with that in BAW-1543 (NP), Revision 4, Supplement 6. The MUR power uprate projected EOL fluence for this material is 1.02×10^{19} n/cm^2 (E>1.0 MeV).

Since the MUR power uprate fluence is bounded by the current equivalent margins analysis fluence, the fracture resistance (J-R curve) of the limiting RV material for the MUR power uprate condition is slightly better than those used in the BAW-2192-P-A equivalent margins analysis. Therefore, the NRC staff confirmed that the DBNPS RV materials are bounded by the BAW-2192-P-A equivalent margins analysis and would continue to meet the USE criteria requirements of 10 CFR Part 50, Appendix G under the MUR power uprate condition.

Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the P-T limits and USE.

3.8.3 PTS

3.8.3.1 Regulatory Evaluation

The PTS evaluation provides a means for assessing the susceptibility of PWR RV beltline materials to failure during a PTS event to assure that adequate fracture toughness exists during reactor operation. The NRC staff's requirements, methods of evaluation, and safety criteria for PTS assessments are given in 10 CFR 50.61. The NRC staff's review covered the PTS methodology and the calculations for the reference temperature for pressurized thermal shock (RT_{PTS}) at the expiration of the license, considering neutron embrittlement effects.

3.8.3.2 Technical Evaluation

Regarding the topic of PTS analyses for the DBNPS RV, the licensee provided the RT_{PTS} value for the limiting beltline material of the DBNPS RV in Section IV.1.C.i of Enclosure 2 to the submittal and concluded:

> 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 1.124×10^{19} n/cm^2 (E>1.0 MeV) based on a 1992 fluence projection plus 5 percent 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.

The NRC staff has evaluated this information as well as information contained in the NRC staff's Reactor Vessel Integrity Database (RVID). The RVID EOL fluence of 1.07×10^{19} n/cm^2 (E>1.0 MeV) for the limiting material of the DBNPS RV is identical to that in BAW-2192-P-A. The MUR power uprate projected EOL fluence for this limiting material is 1.124×10^{19} n/cm^2 (E>1.0 MeV). This slightly higher MUR power uprate EOL fluence gives an MUR power uprate RT_{PTS} value of 193.5 °F, significantly below the PTS criterion of 300°F. Therefore, the NRC staff confirmed that the DBNPS RV materials would continue to meet the PTS screening criteria requirements of 10 CFR 50.61.

3.8.4 Pressurizer

The licensee reviewed the revised design conditions to assess the impact of MUR power uprate on the existing basis analyses for the pressurizer. Section IV.1.A.viii of Enclosure 2 to the submittal indicated that the "temperature changes" due to the MUR power uprate are bounded by those used in the existing analyses. Therefore, the licensee concluded that the existing loads remain valid and the stresses and fatigue analyses also remain valid. In an RAI regarding Section IV.1.A.viii, the NRC staff requested a clarification on the nature of these temperature changes and their effects on the most critical transient that was used in the existing pressurizer integrity analyses. The RAI also requested the licensee to confirm that the current pressurizer analyses have considered appropriate pressurizer insurges.

The licensee replied in its September 18, 2007, response that the "temperature change" that was referred to in Section IV.1.A.viii was hot leg temperature, which will increase slightly due to the MUR power uprate and will result in a slightly lower temperature differential between the hot leg and the pressurizer, lessening the effect of the insurges. The RAI response stated further that pressurizer insurges and outsurges have been evaluated for DBNPS, and the critical pressurizer insurge and outsurge transients are not affected by the MUR power uprate. The NRC staff determined that lower temperature differential between the hot leg and the pressurizer would lessen the effect of the insurges and overall thermal loading, including the stratification effects, and agreed with the licensee that the existing stress reports remain applicable to MUR power uprate. As a result, the NRC staff concluded that the pressurizer components will remain adequate for plant operation at the proposed MUR power uprate.

3.8.5 Reactor Vessel Internals and Core Support Materials

3.8.5.1 Regulatory Evaluation

The RV internals and core supports include SSCs that perform safety functions or whose failure could affect safety functions performed by other SSCs. These safety functions include reactivity monitoring and control, core cooling, and fission product confinement (within both the fuel cladding and the RCPB). The NRC staff's review covered the materials' specifications and mechanical properties, welding controls, nondestructive examination procedures, corrosion resistance, and the materials' susceptibility to degradation. The NRC's acceptance criteria for RV internals and core support materials are based on GDC-1 and 10 CFR 50.55a for material specifications, controls on welding, and inspection of RV internals and core supports. Matrix 1 of NRC RS-001, Revision 0, "Review Standard for Extended Power Uprates," provides references to the NRC's approval of the recommended guidelines for RV internals in Topical Reports WCAP-14577, Revision 1-A, "License Renewal Evaluation: Aging Management for Reactor Internals" (March 2001), and BAW-2248-A, "Demonstration of the Management of Aging Effects for the Reactor Vessel Internals" (March 2000).

3.8.5.2 Technical Evaluation

The licensee discussed the impact of the MUR power uprate on the structural integrity of the DBNPS RV internal components in Section IV.1.A.ii of Enclosure 2 to the submittal. The licensee concluded that the temperature changes due to the MUR power 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.

The RV internals of PWR-designed light-water reactors may be susceptible to the following aging effects:

- cracking induced by thermal cycling (fatigue-induced cracking), stress-corrosion cracking (SCC), or irradiation assisted (IA) SCC;

- loss of fracture toughness properties induced by irradiation exposure for all stainless steel grades, or the synergistic effects of irradiation exposure and thermal aging for cast austenitic stainless steel (CASS) grades;

- stress relaxation in bolted, fastened, keyed or pinned RV internal components induced by irradiation exposure and/or exposure to elevated temperatures; and

- void swelling (induced by irradiation exposure).

Table Matrix 1 of NRC RS-001, Revision 0, provides the NRC staff's basis for evaluating the potential for extended power uprates to induce these aging effects. Depending on the magnitude of the RV internals fluence, Table Matrix 1 may be applicable to the current MUR power uprate application. In Note 1 to Table Matrix 1, the NRC staff stated that guidance on the neutron irradiation-related threshold for IA SCC for PWR RV internals are given in BAW-2248-A and WCAP-14577, Revision 1-A. This Table Matrix 1 note further stated that for thermal and neutron embrittlement of CASS, SCC, and void swelling, licensees will need to provide plant-specific degradation management programs or participate in industry programs that investigate degradation effects and determine appropriate management programs. The submittal did not mention any of the above. Therefore, the NRC staff's RAI requested the licensee to address this RS-001, Revision 0 concern, i.e., to discuss its management of the above-mentioned aging effects on RV internals in light of the guidance in BAW-2248-A and WCAP-14577, Revision 1-A and its plant-specific or industry programs to manage thermal and neutron embrittlement of CASS, SCC, and void swelling.

In the RAI response dated September 18, 2007, the licensee replied that FENOC is an active participant of the Electric Power Research Institute (EPRI) Materials Reliability Project (MRP) Issues Task Group, which is addressing the age-related degradation effects on RV internals. In its RAI response dated January 15, 2008, the licensee further clarified that, as appropriate, FENOC commits to incorporate recommendations from EPRI's MRP inspection guidelines into the reactor vessel internals program at DBNPS. This commitment was included in the licensee's official Commitment List. With the commitment, the licensee will manage the degradation of DBNPS RV internals during the remaining period of operation with MUR power uprate using the inspection and evaluation guidelines for assessing each of the aging effects listed above with the MRP guidelines. Therefore, the NRC staff finds the proposed MUR acceptable with respect to reactor internal and core support materials.

3.8.6 Overall Summary

The NRC staff has reviewed the licensee's LAR to increase the rated core thermal power for MUR power uprate by 1.63 percent and has evaluated the impact that the MUR power uprate conditions will have on the structural integrity assessments for the RV, pressurizer, and RV internals. The NRC staff has determined that the changes identified in the proposed LAR will not impact the remaining safety margins required for the following structural integrity assessments: (1) RV surveillance program, (2) RV USE assessment, (3) P-T limits, (4) PTS assessment, (5) pressurizer shell integrity, and (6) the integrity for the stainless steel RV internals, in that the licensee has committed to incorporate recommendations from EPRI's MRP inspection guidelines into the reactor vessel internals program at DBNPS. The NRC staff finds the requested MUR power uprate for DBNPS to be acceptable.

3.9 Fire Protection

3.9.1 Regulatory Evaluation

The purpose of the fire protection program is to provide assurance, through a defense-in-depth design, that a fire will not prevent the performance of necessary plant safe-shutdown functions and will not significantly increase the risk of radioactive releases to the environment. The NRC staff's review focused on the effects of the increased decay heat on the plant's safe-shutdown analysis to ensure that SSCs required for the safe-shutdown of the plant are protected from the effects of the fire and will continue to be able to achieve and maintain safe-shutdown following a fire. The NRC's acceptance criteria for the fire protection program are based on (1) 10 CFR 50.48, "Fire protection" and associated Appendix R to 10 CFR Part 50, insofar as they require the development of a fire protection program to ensure, among other things, the capability to safely shutdown the plant; (2) GDC 3 of Appendix A to 10 CFR Part 50, insofar as it requires that (a) SSCs important to safety be designed and located to minimize the probability and effect of fires, (b) noncombustible and heat resistant materials be used, and (c) fire detection and suppression systems be provided and designed to minimize the adverse effects of fires on SSCs important to safety; and (3) GDC-5 of Appendix A to 10 CFR Part 50, insofar as it requires that SSCs important to safety not be shared among nuclear power units unless it can be shown that sharing will not significantly impair their ability to perform their safety functions.

3.9.2 Technical Evaluation

The licensee developed the LAR consistent with the guidelines in RIS 2002-03. In the LAR, the licensee re-evaluated the applicable SSCs and safety analyses at the proposed MUR core power level of 2817 MWt against the previously analyzed core power level of 2772 MWt. The NRC staff reviewed the LAR, Enclosure 2, Attachment A, "D-B MUR Summary Report," Sections II and III. The NRC staff also reviewed the licensee's commitment to 10 CFR 50.48, "Fire protection" (i.e., approved fire protection program). Specifically, staff reviewed NRC May 30, 1991 SER listed in the DBNPS fire protection License Condition 2.C(4). This SER documents the NRC staff evaluation of the fire protection measures at DBNPS per Appendix R to 10 CFR Part 50.

The review also covers the impact of the proposed MUR power uprate on the results of the safe-shutdown fire analysis as noted in RIS 2002-03, Attachment 1, Sections II and III. The review focused on the effects of the MUR power uprate on the post-fire safe-shutdown capability and increase in decay heat generation following plant trips. In the LAR the licensee stated that each of the current accidents and transients safe-shutdown fire analysis of record for DBNPS was unaffected by the requested power uprate because they were performed assuming a plant output 102 percent of 2772 MWt. Analyses performed at this power are applicable to the requested uprated power of 2817 MWt with a 0.37 percent power measurement uncertainty.

In a letter dated July 25, 2007, the NRC staff issued an RAI, requesting the licensee to clarify whether the MUR power uprate LAR involves changes to the fire protection program or other operating conditions that may adversely impact the post-fire safe-shutdown capability in accordance with Appendix R to 10 CFR Part 50, and to provide the technical justification for whether and, if so, why, existing analyses bound any impact on accidents or transients resulting from any changes.

In a letter to the NRC dated September 18, 2007, the licensee provided additional information in response to the NRC staff's RAIs. In its response the licensee stated that:

> [DBNPS] has performed various analyses to demonstrate safe plant shutdown following a fire. Two calculations to show adequate core cooling were completed to address the two potential results of a fire. Calculation C-NSA-064.02-032, "Davis-Besse Appendix R Overheating Summary Report," analyzed the potential over-heating conditions (i.e., actuations/failures leading to a loss of Feedwater, loss of Makeup/High Pressure Injection (HPI), and spurious opening of Reactor Coolant System (RCS) leak paths). Calculation C-NSA-064.02-003, "Davis-Besse Appendix R Overcooling Summary Report," analyzed the potential over-cooling conditions (i.e., full Auxiliary Feedwater (AFW) flow and spurious opening RCS leak paths.)
>
> The analyses were performed with the objective of updating the analyses to current standards and quantifying available margins by using updated computer models. The updated analyses are used to define time critical operator actions. These calculations will also support an increase in the Davis-Besse rated core power from 2772 to 3014 MWt.
>
> The overcooling analysis was performed using the topically-approved RELAP5/MOD2 computer code with modified boundary conditions from the main steam line break (MSLB). For the MSLB accidents (or overcooling type of events), a nominal core power is used to minimize heat input to the RCS. The results of the analyses using a nominal core power level of 2772 MWt is conservative for a higher core power level because the core decay heat will be minimized. Less core decay heat results in a greater overcooling of the RCS and a greater challenge to maintaining the minimum subcritical margin. The overcooling analysis determined the minimum operator action time to manage RCS makeup flow, core reactivity, and SG overfill concerns.
>
> The overheating analysis was also performed using the RELAPS5/MOD2 computer code with modified boundary conditions form the small break loss of coolant accident (SBLOCA) evaluation model (EM). Successful core cooling was demonstrated for core thermal power levels up to 3025.32 MWt, with flow from only one AFW and ECCS train. That power level bounds the power level proposed by the MUR uprate.

In a letter dated July 25, 2007, the NRC staff requested the licensee to verify whether the following conclusion in the NRC May 30, 1991, SER is valid at an increased reactor power level of 2817 megawatts thermal (MWt), 1.63 percent above the currently licensed power level of 2772 MWt.

> The staff's conclusion is also based on the statements made by the licensee in its letter dated June 6, 1988, that the capability to return the pressurizer level to within prescribed instrument indication range, and to restore other process variables to within the range predicted by a loss of offsite power, will be preserved. In addition, the licensee states that the core will not be uncovered and fission product boundary integrity will not be affected during the postulated transient conditions.

In its response, by letter dated September 18, 2007, the licensee stated the following:

> The conclusion is still valid for the revised power levels. As discussed in the response [above], the [DBNPS] Appendix R Overheating analysis was performed at a reactor power level of 3014 MWt. One of the acceptance criteria of the overheating analysis was

to maintain the core covered. The acceptance criteria (see below) references the June 6, 1988 letter (DBNPS Serial Number 1535) mentioned in the question above. Appendix B of Attachment 1 of the calculation discusses the following acceptance criteria in more detail, including the 1988 letter and 1991 SER.

> The RCS inventory will be permitted to be depleted until an unrecoverable condition is reached. An unrecoverable condition is defined in accordance with Reference 23 [June 6, 1988 letter mentioned above] as "the loss of any shutdown function(s) for such a duration as to ultimately cause the reactor coolant collapsed liquid level to fall below the top of the active fuel height of the core and subsequent breach of the fuel cladding. Maintaining the reactor coolant level above the top of the core ensures adequate core cooling and fission product boundary integrity." This definition of minimum allowable RCS inventory ensures that core cooling will be maintained for the duration of the event.

The licensee evaluated the post-fire safe-shutdown capability following a fire by performing two calculations (potential overheating condition and potential overcooling condition) as result of a fire. These analyses of record show that, for the proposed MUR power uprate, the post-fire safe-shutdown capability continues to meet the requirements of Appendix R to 10 CFR Part 50. The licensee indicated that the LAR involves no changes to the fire protection program or other operating conditions that may adversely impact the post-fire safe-shutdown capability. Further, the licensee indicated that the existing analyses are bounding.

The information provided in the LAR, as supplemented in the response to the NRC staff RAIs, satisfactorily demonstrates the licensee's compliance. The increases in decay heat usually do not affect the elements of a fire protection program related to: (1) administrative controls, (2) fire suppression and detection systems, (3) fire barriers, (4) fire protection responsibilities of plant personnel, and (5) procedures and resources necessary for the repair of systems required to achieve and maintain cold shutdown. In addition, an increase in decay heat will usually not result in an increase in the potential for a radiological release resulting from a fire Further, the licensee indicated that compliance with the fire protection and safe-shutdown program will not be affected because the MUR power uprate evaluation did not identify changes to design or operating conditions that will adversely impact the post-fire safe-shutdown capability. The MUR power uprate evaluation does not change the credited equipment necessary for post-fire safe-shutdown nor does it reroute essential cables or relocate essential components/equipment credited for post-fire safe-shutdown.

The licensee has made no changes to the plant configuration or combustible loading as a result of modifications necessary to implement the MUR power uprate that affect the DBNPS fire protection program.

3.9.3 Summary

The NRC staff has reviewed the licensee's fire-related safe-shutdown assessment and concludes that the licensee has adequately accounted for the effects of the increased decay heat on the ability of the required systems to achieve and maintain post-fire safe-shutdown conditions. The NRC staff further concludes that the fire protection program will continue to meet the requirements of 10 CFR 50.48, Appendix R to 10 CFR Part 50, and GDC 3 following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to fire protection.

3.10 Accident Dose Analyses

3.10.1 Regulatory Evaluation

This SE input addresses the impact of the proposed changes on previously analyzed DBA radiological consequences and the acceptability of the revised analysis results. The regulatory requirements on which the NRC staff based its acceptance are the accident dose guidelines in 10 CFR 100.11, as supplemented by accident-specific criteria in Section 15 of the SRP, and 10 CFR Part 50 Appendix A, GDC 19, "Control Room," as supplemented by Section 6.4 of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants." The NRC staff also considered relevant information in the DBNPS UFSAR and TSs.

The NRC staff's review covers the impact of the proposed MUR power uprate on the results of dose consequence analyses as noted in RIS 2002-03, Attachment 1, Sections II and III.

3.10.2 Technical Evaluation

The NRC staff reviewed the regulatory and technical analyses, as related to the radiological consequences of design basis accidents, performed by DBNPS in support of its proposed license amendment. Information regarding these analyses was provided in Section 4 of the LAR. Only docketed information was relied upon in making this safety finding.

The NRC staff reviewed the impact of the proposed 1.63 percent power uprate on DBA radiological analyses, as documented in Chapter 15 of the UFSAR. In its submittal, the licensee stated that each of the current DBA dose analyses of record for DBNPS were unaffected by the requested power uprate because they were performed assuming 102 percent of 2772 MWt. Analyses performed at this power are applicable to the requested uprated power of 2817 MWt with a 0.37 percent power measurement uncertainty. Using the current DBNPS UFSAR documentation, in addition to information in the April 12, 2007, submittal letter, the NRC staff verified that the existing DBNPS UFSAR Chapter 15 radiological analyses source term and release assumptions bound the conditions for the proposed 1.63 percent power uprate to 2817 MWt, considering the higher accuracy of the FW measurement instrumentation. These analyses of record show that, for the proposed power uprate, the radiological consequences of postulated DBAs continue to meet the dose limits given in 10 CFR 100.11 and 10 CFR Part 50, Appendix A, GDC-19, as well as applicable dose acceptance criteria given in NUREG-0800, Standard Review Plan, Chapter 15, for the radiological consequences of DBAs.

3.10.3 Summary

The NRC staff has reviewed the licensee's assessment of the impact of the proposed 1.63 percent MUR power uprate on dose consequences analyses for the DBNPS. As discussed above, the NRC staff has determined that the results of the licensee's analyses of the radiological consequences of DBAs continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR power uprate acceptable with respect to the dose consequences of DBAs.

3.11 Plant Systems

The NRC staff's review in the area of plant systems covers the impact of the proposed MUR power uprate on NSSS interface systems, containment systems, safety-related cooling water systems, spent fuel pool (SFP) storage and cooling systems, radioactive waste systems, and engineered safety features (ESF) heating, ventilation, and air conditioning (HVAC) systems. The NRC staff's review is based on the guidance in SRP Chapters 3, 6, 9, 10, and 11, and RIS 2002-03, Attachment 1, Sections II, III, and VI. The licensee evaluated the effect of the MUR on the plant systems. This evaluation is reflected in Section VI of Attachment A of the application dated April 12, 2007.

3.11.1 NSSS Interface Systems

The NSSS interface systems include the main steam (MS) system, the atmospheric vent valves (AVVs) and turbine bypass valves (TBVs), the condensate system (CD), and main FW system.

The MS system provides isolation of the steam generators after a steam line failure, provides overpressure relief and/or decay heat removal during accidents, and provides steam to the auxiliary FW system. For the MS system, the licensee stated that the MS system analysis of record is not impacted by the MUR power uprate.

The AVVs provide a controlled path for venting steam to the atmosphere. For the AVVs, the licensee evaluated the valves for their functions to (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. Since the power uprate conditions are bounded by the existing design conditions, the licensee determined that the functional performance of the AVVs will be unaffected by the power uprate.

The TBVs' primary function is to maintain stable turbine header pressure during load swing startup and shutdown. The licensee determined that the steam flow rate is not being changed for the power uprate, and the system parameters are bounded by the existing design conditions. Therefore, the TBVs will be unaffected by the MUR power uprate.

The CD system supplies preheated condensate to the FW system. For the CD system, the licensee evaluated the performance of the system and determined that the existing analysis of record bounds accident/transient analyses for components/systems for plant operation at the proposed uprate power level. The licensee stated that the proposed change in condensate storage tank 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 12 hours with steam discharge to atmosphere and to cooldown the RCS to less than 280 °F under normal conditions, in accordance with the current licensing basis. Therefore, the licensee determined that this change will have no adverse effect on nuclear safety.

The main FW system provides FW during normal operation and isolates during accidents. The licensee determined that the existing analysis of record bounds the accident/transient analyses for components/systems for plant operation at the proposed uprate power level. The licensee also stated that the main FW system piping analysis for the 1.63 percent MUR power uprate conditions remain bounded by the design analysis of record.

The AFW system is designed to provide FW to the SGs when the turbine-driven main FW pumps are not available, or following a loss of normal and reserve electric power. The licensee stated that the AFW system design basis of record is not affected by the proposed MUR power uprate.

The NRC staff reviewed the licensee's evaluation and concurs with the results. The licensee determined that there is no adverse impact on the NSSS interface systems from the MUR power uprate because there is sufficient operating margin to produce an additional 1.63 percent power, and all equipment will be operated within its design limits. The NRC staff does not anticipate that an MUR power uprate will challenge the NSSS interface systems, and all systems have been shown to be operating within design. Therefore, the NRC staff finds that the NSSS systems are acceptable for the MUR uprate.

3.11.2 Containment Systems

The containment systems review consists of evaluating the functional design of the containment, the mass and energy release into the containment from postulated primary and secondary system breaks, subcompartment analyses, combustible gas control, and sump water temperature and ECCS and containment heat removal pump net positive suction head. For each of these, MUR power uprate conditions remain bounded by the design basis of the DBNPS UFSAR. The NRC staff therefore finds the licensee's containment analysis acceptable for MUR conditions.

3.11.3 Safety-Related Cooling Water Systems

The safety-related cooling water systems include the component cooling water (CCW) system and the service water system (SWS)/ultimate heat sink.

The SWS is designed to serve two functions during station operation. The first function is to supply cooling water to the CCW 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 a redundant supply path to the engineered safety features components during an emergency. The licensee stated that the post-LOCA containment analyses contained in UFSAR Section 6.2 are based on 3025 MWt, and therefore bound the MUR power uprate. Therefore, the NRC staff finds that the safety-related cooling water systems will be acceptable for the MUR power uprate conditions.

3.11.4 SFP Storage and Cooling

The SFP storage and cooling systems are described in Section 9 of the DBNPS UFSAR. The principal function is to provide safe storage and cooling of the spent fuel. The licensee determined that the SFP cooling system is bounded by the existing analyses of record. Therefore, the NRC staff finds that the SFP Storage and Cooling Systems will be acceptable for the MUR power uprate conditions.

3.11.5 Radioactive Waste Systems

The DBNPS radioactive waste systems provide the means to sample, collect, process, store/hold, re-use, and/or release gaseous and liquid low-level effluents, and are described in section 11 of the DBNPS UFSAR. The licensee stated that 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.

The licensee stated that the power uprate may result in the production of some additional liquid waste production, primarily from processing additional reactor coolant waste due to the higher boron concentration at the beginning of core life. The licensee further stated that this additional production should be minimal and will not impact the ability of the system to function as designed and currently operated. The licensee also stated that some additional increase in spent resin processing and contaminated solid materials may occur over the core life, however, the increase will not challenge the ability of the solid waste processing systems to perform as designed. Based on the above, the NRC staff finds that the radioactive waste systems will be acceptable for the MUR power uprate conditions.

3.11.6 ESF HVAC Systems

The licensee evaluated the ESF HVAC systems and determined that the systems remain bounded by the design basis (102 percent of 2772 MWt) of the DBNPS UFSAR for MUR power uprate conditions. The licensee stated that 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 licensee further stated that the containment cooling system has adequate margin to cool the containment at MUR conditions. Therefore, the NRC staff finds that the ESF HVAC systems will be acceptable for the MUR power uprate conditions.

3.11.7 Summary

In summary, the licensee reviewed the design and operation of the plant systems and determined that the proposed MUR power uprate does not adversely impact any of the systems. For the reasons noted above, the NRC staff concurs with the licensee's conclusion and finds that the plant systems will be acceptable for the MUR power uprate.

3.12 Changes to Facility Operating License (FOL) and TSs

3.12.1 Regulatory Evaluation

This LAR revises FOL Condition 2.c, the TS definition of RTP, TS 2.2, "Limiting Safety System Settings," TS 3.3.1, "Reactor Protection System (RPS) Instrumentation," and TS 3.7.1.3, "Condensate Storage Tank Level" to reflect the LAR proposed increase in the maximum rated thermal power level.

Section 182a of the Atomic Energy Act (the "Act") requires applicants for nuclear power plant operating licenses to include TS as part of the license. The TS ensure the operational capability of SCCs that are required to protect the health and safety of the public. The regulatory requirements related to the content of the TS are contained in 10 CFR 50.36. Section 50.36 of 10 CFR requires that the TS include items in the following specific categories: (1) Safety limits, LSSSs, and limiting control settings (50.36(d)(1)); (2) LCO (50.36(d)(2)); (3) SRs (50.36(d)(3)); (4) design features (50.34(d)(4)); and (5) Administrative Controls (50.36(d)(5)).

Section 50.36(d)(1)(ii)(A) of 10 CFR defines LSSSs for nuclear reactors as settings for automatic protective devices related to those variables having significant safety functions. Section 50.36(d)(2)(i) of 10 CFR defines limiting conditions for operation as the lowest functional capability or performance levels of equipment required for safe operation of the facility. Section 50.36(d)(2)(ii) of 10 CFR defines the criteria to be used for evaluating items to determine if a limiting condition for operation of a nuclear reactor must be established. Section 50.36(d)(3) of 10 CFR defines SRs as requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the LCOs will be met.

In general, there are two classes of changes to TS: (1) Changes needed to reflect modifications to the design basis (TS are derived from the design basis), and (2) voluntary changes to take advantage of the evolution in policy and guidance as to the required content and preferred format of TS over time. This amendment deals with the first class of changes. In determining the acceptability of revising Definitions (1.0), LSSSs (2.2), RPS Instrumentation (3.3.1), Condensate Storage Tank (3.7.1.3) and Administrative Controls – Core Operating Limits Report (COLR) (6.9.1.7) TSs, the NRC staff used the accumulation of generically approved guidance in NUREG–1430, "Standard Technical Specifications, Revision 3 Babcock and Wilcox Plants," dated June, 2004.

Licensees may revise the TS content provided that plant-specific review supports a finding of continued adequate safety because: (1) the change is editorial, administrative or provides clarification (i.e., no requirements are materially altered), (2) the change is more restrictive than the licensee's current requirement, or (3) the change is less restrictive than the licensee's current requirement, but nonetheless still affords adequate assurance of safety when judged against current regulatory standards. The detailed application of this general framework, and additional specialized guidance, are discussed in this section in the context of the specific proposed changes. Nomenclature specific to the B&W plants is used in the following technical evaluation.

3.12.2 Technical Evaluation

The submittal includes TS requirements that would demonstrate compliance with 10 CFR 50.36, "Technical Specifications," for plant operating conditions related to the requested authorization for a power level increase. The plant modifications will improve the accuracy of the plant power calorimetric measurement based on the Caldon LEFM Checkplus ™ System (ultrasonic flow meter) instrumentation. A discussion of TS changes follows.

3.12.2.1 Definitions – Rated Thermal Power

The licensee proposed to revise TS 1.3, Definitions – RTP to reflect the authorized power level increase. The TS RTP will limit the maximum reactor core heat transfer rate to the reactor coolant to 2817 MWt. The NRC staff finds that this change meets 10 CFR 50.36 and is acceptable because the TS limit for operation is derived from the analyses and evaluation included in the safety analysis report (SAR) as accepted by the SE for the requested power level increase discussed herein.

3.12.2.2 LSSSs - RPS Setpoints

TS 2.2.1 specifies that the RPS instrumentation setpoints shall be set consistent with the allowable values (AVs) shown in Table 2.2-1, "Reactor Protection System Instrumentation Trip Setpoints." As proposed, Table 2.2.1 contains 3 possible AVs, depending on circumstances. The licensee proposed two TS 2.2.1 AV changes. One is related to the requested power level increase when the UFM instrumentation is used in the performance of the daily secondary heat balance calorimetric. The second applies when TS remedial actions are being implemented due to the UFM instrumentation not being used in the performance of the daily secondary heat balance calorimetric.

The licensee proposed to change the AV listed in Table 2.2-1 for Functional Unit 2, High Flux, from ≤ 105.1 percent to ≤ 104.9 percent of RTP with four pumps operating with the secondary heat balance calorimetric based on UFM instrumentation. The safety analysis analytical limit does not support an AV of 105.1 percent RTP unless the UFM instrumentation is functional (UFM instrumentation is not TS-required equipment) and used in the calculation of the daily secondary heat balance calorimetric. The licensee also proposed to require the High Flux Trip AV to be ≤ 103.3 percent of RTP when TS requirements are being implemented due to the UFM instrumentation not being used in the performance of the daily secondary heat balance calorimetric, thereby ensuring TS maintain the same level of protection of the analytical limit.

The licensee stated that no changes to the TS high flux trip setpoints for reduced pump operation (three pumps) are required because the TS three pump high flux setpoint is conservative and bounds operation irrespective of the use of the UFM for heat balance calculation. The proposed High Flux Trip AV preserve assumptions of current accident analyses at the higher thermal power allowed by the proposed amendment, irrespective of the source of daily heat balance calculation input data.

The NRC staff finds the changes to TS 2.2.1 meets 10 CFR 50.36 and are acceptable because the TS limiting conditions for operation are derived from the analyses and evaluation included in the safety analysis report as accepted by the safety evaluation contained herein.

3.12.2.3 RPS Instrumentation

When a TS 3/4.3.1 RPS instrumentation setpoint is less conservative than the value shown in the AV column of Table 2.2-1, the licensee must declare the channel inoperable and apply the applicable Action Statement requirement of TS 3.3.1.1 until the channel is restored to operable status with its trip setpoint set consistent with the AV. The licensee stated that 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 AV of ≤ 104.9 percent of rated power under the new authorized power level. 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 percent of 2817 MWt, which was used in the safety analysis.

The licensee proposed to revise TS 3.3.1 by adding Action 11 to Table 3.3-1, “Reactor Protection System Instrumentation,” for Functional Unit 2, High Flux. The licensee also proposed to revise TS 4.3.1, Note 2 of Table 4.3-1, “Reactor Protection System Surveillance Requirements,” for daily Channel Calibration of Functional Unit 2, High Flux instrumentation channels.

The licensee proposed to add new Action 11 to TS 3.3.1. Action 11.a requires core thermal power to be immediately reduced to ≤ 98.4 percent RTP (2772 MWt) with four RCPs operating or to ≤ 73.8 percent RTP (2079 MWt) with three RCPs operating when the calculated required secondary heat balance is no longer based on the UFM instrumentation. Action 11.b requires the High Flux trip setpoint to be reduced to ≤ 103.3 percent RTP (2910 MWt) within 10 hours with four RCPs operating. Both Action 11.a and 11.b must be met if one or more channels are inoperable due to a daily heat balance calorimetric that is not based on UFM instrumentation. The licensee’s basis for the 10 hour time period is to allow for an orderly reduction of power and implementation of the setpoint change. The licensee indicated that the expected setpoint drift over 10 hours is insignificant and the NI will remain calibrated for that time frame. The licensee will also reduce core power to 2772 MWt (2772 MWt is 98.4 percent of the uprate RTP of 2817 MWt and equal to current RTP) when the high-accuracy calorimetric is not functional, to ensure that the core power level is within the analyzed limit.

If the requirements of Action 11 cannot be met, then the affected high flux channels are inoperable and Action 2 or 10, as applicable, is required. The requirements and timeframe for Action 11 provide remedial action acceptable to the NRC staff for the condition when the calculated required secondary heat balance is no longer based on the UFM instrumentation.
If not met, the TS actions require plant shutdown in accordance with Action 10.

Table 4.3-1 requires a Channel Calibration (heat balance only) every 24 hours for high flux channels when above 15 percent of RTP. The licensee proposed to revise Note 2 for the Functional Unit 2, High Flux reactor trip to include a requirement when reactor power is above 50 percent RTP to verify that the calculated secondary heat balance RTP using the UFM instrumentation is ≤ 2 percent RTP greater than the NI output unless Action 11 of Table 3.3.1 is entered. The 2 percent criterion in the daily calorimetric calibration is not related to the accuracy of secondary heat balance instrumentation; it is an allowed difference that is factored into the safety analyses limits for both four pump and three pump operation. The heat balance calorimetric above 50 percent RTP using the UFM limits the use of the venturi flow orifice to less than 50 percent RTP.

The NRC staff finds that the changes to TS 3.3.1 and TS 4.3.1 meet 10 CFR 50.36 and are acceptable because the TS LCO, remedial actions and SRs are derived from the analyses and evaluation included in the SAR as accepted by the SE contained herein.

3.12.2.4 Condensate Storage Tanks

The licensee proposed that TS 3.7.1.3 be revised to reflect a new minimum volume as follows: The condensate storage tanks shall be OPERABLE with a minimum usable volume of 270,300 gallons of water. SR 4.7.1.3.1 would be revised to match the new minimum volume. In addition, TS 3.7.1.3 would be revised to clarify that the value to be confirmed is the usable water volume of the tanks as compared to the current TS requirement to meet the contained water volume. The NRC staff finds the changes to TS 3.7.1.3 to be acceptable because they meet 10 CFR 50.36 in that the TS LCO is derived from the analyses and evaluation included in the safety analysis report as accepted by the SE contained herein.

3.12.2.5 Administrative Controls - COLR

TS 6.9.1.7 is revised 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 percent of RTP is specified in a previously approved method, an actual value of 100.37 percent of RTP may be used when the input for reactor thermal power measurement of FW mass flow and temperature is from the Ultrasonic Flow Meter.

In addition, TS 6.9.1.7 has been further modified by adding the following NRC staff-approved documents, applicable to the use of the UFM with a 0.37 percent measurement uncertainty, to the section:

> 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, 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.

The NRC staff finds that the changes to TS 6.9.1.7 meet 10 CFR 50.36 and are acceptable because the Administrative Controls TS changes are derived from the analyses and evaluation included in the SAR as accepted by the SE contained herein.

3.12.2.6 TS Bases

The licensee also provided markups to the TS Bases. The NRC staff reviewed the Bases markups and verified that they adequately reflect the bases for the revised TSs. However, the TS Bases are controlled by the licensee's Bases Control Program and, therefore, they are not included in this amendment.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

This amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (72 FR 51861; September 11, 2007). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

7.0 REFERENCES

1. Bezilla, Mark B., "Davis-Besse Nuclear Power Station, License Amendment Application for Measurement Uncertainty Recapture Power Uprate (License Amendment Request No. 05-0007)," Letter to NRC from Vice President-Nuclear, FirstEnergy Nuclear Operating Company, Serial Number 3198, ML071030396, April 12, 2007.

2. "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM Check System," Caldon Topical Report ER-80P, Rev 0, March, 1997.

3. "Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or an LEFM CheckPlus," Caldon Engineering Report ER-157P, Revision 5, October, 2001.

4. Thomas, Brian E., “Evaluation of the Hydraulic Aspects of the Caldon Leading Edge Flow Measurement (LEFM) Check and CheckPlus™ Ultrasonic Flow Meters (UFMs) (TAC No. MC6424),” Letter to Ernest M. Hauser, President, Caldon, Inc., from NRC Acting Deputy Director, ML061700222, July 5, 2006.

5. Bezilla, Mark B., “Response to Request for Additional Information Regarding Measurement Uncertainty Recapture Power Uprate Amendment Application (TAC No. MD5240),” Letter to NRC from Vice President-Nuclear, FirstEnergy Nuclear Operating Company, L-08-094, ML080770169, March 12, 2008.

6. “Caldon Ultrasonics, Engineering Report: ER-551P Rev.1, LEFM✔ + Transducer Installation Sensitivity,” ML071500360 (proprietary, non-proprietary is ML072740228), March, 2007.

7. “Flow Measurement Uncertainty Due To Transducer (Re)placement in Caldon® LEFM Check and CheckPlus Systems,” Caldon Ultrasonics, PR-612P Rev. 0, ML070870441 (Proprietary, non-proprietary is ML070870435), March 15, 2007.

8. Bezilla, Mark B., “Response to Request for Additional Information Regarding Measurement Uncertainty Recapture Power Uprate Amendment Application (TAC No. MD5240),” Letter to NRC from Vice President-Nuclear, FirstEnergy Nuclear Operating Company, Serial Number 3355, ML072640612, September 18, 2007.

9. “Calibration of Two 18” Leading Edge Flow Meters for Caldon, Inc., Purchase Order Number 18350, October 2001 – ARL No. 310-01/C730,” Alden Research Laboratory, Inc, Proprietary, October, 2001.

10. U.S. Nuclear Regulatory Commission, “Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications,” RIS 2002-03, January 31, 2002.

11. Wambach, Thomas V., USNRC, Letter to Donald C. Shelton, Toledo Edison Company, “Evaluation of the Davis-Besse Nuclear Power Station Compliance with 10 CFR 50.62 Requirements for Reduction of Risk from Anticipated Transients Without Scram,” September 29, 1989.

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Table of Acronyms

Acronym	Definition
AC	Alternating Current
ADAMS	Agencywide Documents Access and Management System
AFW	Auxiliary Feedwater
ANSI	American National Standards Institute
AOT	Allowed Outage Time
ARL	Alden Research Laboratory
ASA	American Standards Association
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transients Without Scram
AV	Allowable Value
BOP	Balance of Plant
BWR	Boiling Water Reactor
CASS	Cast Austenitic Stainless Steel
CCW	Component Cooling Water
CD	Condensate System
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CRA	Control Rod Assembly
CRDM	Control Rod Drive Mechanisms
CUF	Cumulative Usage Factors
CVCS	Chemical and Volume Control System
DBA	Design Basis Accident
DBNPS	Davis-Besse Nuclear Power Station,
DC	Direct Current
ECCS	Emergency Core Cooling System
EDG	Emergency Diesel Generator
EFPY	Effective Full Power Years
EOL	End of License
EPRI	Electric Power Research Institute
EQ	Equipment Qualification
ESF	Engineered Safety Features
FAC	Flow-Accelerated Corrosion
FE	FirstEnergy
FENOC	FirstEnergy Nuclear Operating Company
FIV	Flow-Induced Vibration
FOL	Facility Operating License
FR	Federal Register
FW	Feedwater
GDC	General Design Criterion
GL	Generic Letter
HELB	High Energy Line Break
HPI	High Pressure Injection
HVAC	Heating Ventilation and Air Conditioning
IA	Irradiation Assisted
ID	Inside Diameter
ISA	Instrument Society of America

Table of Acronyms

Acronym	Definition
LAR	License Amendment Request
LCO	Limiting Conditions for Operation
LED	Light Emitting Diode
LEFM	Leading Edge Flow Measurement
LOCA	Loss of Coolant Accident
LSSS	Limiting Safety System Setting
LTOP	Low-Temperature Overpressure Protection
MIRVSP	Master Integrated Reactor Vessel Surveillance Program
MRP	Materials Reliability Project
MS	Main Steam
MSLB	Main Steam Line Break
MUR	Measurement Uncertainty Recapture
MVA	Megawatt Ampere
NERC	North American Electric Reliability Corporation
NI	Nuclear Instrumentation
NOP	Normal Operating Pressure
NRC	Nuclear Regulatory Commission
NRR	Nuclear Reactor Regulation
NSSS	Nuclear Steam Supply System
OLTP	Original Licensed Thermal Power
OTSG	Once-Through Steam Generator
PTS	Pressurized Thermal Shock
PWR	Pressurized Water Reactor
PWROG	Pressurized Water Reactor Owners Group
RAI	Request for Additional Information
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RG	Regulatory Guide
RIS	Regulatory Issue Summary
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel
RTD	Resistance Temperature Detector
RTP	Rated Thermal Power
RTPTS	Reference Temperature for Pressurized Thermal Shock
RV	Reactor Vessel
RVID	Reactor Vessel Integrity Database
SAR	Safety Analysis Report
SBLOCA	Small Break Loss Of Coolant Accident
SBO	Station Blackout
SBODG	Station Blackout Diesel Generator
SCC	Stress Corrosion Cracking
SE	Safety Evaluation
SER	Safety Evaluation Report
SFP	Spent Fuel Pool

Table of Acronyms

Acronym	Definition
SG	Steam Generator
SGBS	Steam Generator Blowdown System
SPDS	Safety Parameter Display System
SR	Surveillance Requirement
SRP	Standard Review Plan
SRSS	Square Root of Sum of Squares
SWS	Service Water System
TS	Technical Specification
TSO	Transmission System Operator
UFM	Ultrasonic Flow Meters
UFSAR	Updated Final Safety Analysis Report
USE	Upper Shelf Energy
USNRC	United States Nuclear Regulatory Commission
VCT	Volume Control Tank