UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 21, 2020

Mr. James Barstow
Vice President, Nuclear Regulatory Affairs
and Support Services
Tennessee Valley Authority
1101 Market Street, LP 4A-C
Chattanooga, TN 37402-2801

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 2 - ISSUANCE OF AMENDMENT NO. 42 REGARDING MEASURMENT UNCERTAINTY RECAPTURE POWER UPRATE (EPID L-2019-LLS-0000)

Dear Mr. Barstow:

The U.S. Nuclear Regulatory Commission (NRC, Commission) has issued the enclosed Amendment No. 42 to Facility Operating License No. NPF-96 for the Watts Bar Nuclear Plant, Unit 2. This amendment is in response to your application dated October 10, 2019, as supplemented by letters dated April 29, July 27, and August 28, 2020.

This amendment increases the authorized reactor core power level by approximately 1.4 percent rated thermal power from 3411 megawatts thermal (MWt) to 3459 MWt, based on the use of the existing Caldon® Leading Edge Flow Meter (LEFM®) CheckPlus System. Additionally, it revises operating license Item 2.C.(1) and Technical Specification (TS) 1.1, "Definitions," to reflect the increase in rated thermal power, and revises TS 5.9.5, "Core Operating Limits Report (COLR)," to allow that 100.6 percent rated thermal power may be used only when feedwater flow measurement is provided by the LEFM as described in the NRC-approved Caldon Topical Reports for LEFMs.

A copy of our related safety evaluation is also enclosed. Notice of issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Kimberly J. Green, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-391

Enclosures:
1. Amendment No. 42 to NPF-96
2. Safety Evaluation

cc: Listserv

TENNESSEE VALLEY AUTHORITY

DOCKET NO. 50-391

WATTS BAR NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 42
License No. NPF-96

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Tennessee Valley Authority (TVA, the licensee) dated October 10, 2019, as supplemented by letters dated April 29, 2020, July 27, 2020, and August 28, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2.C.(1) and 2.C.(2) of Facility Operating License No. NPF-96 are hereby amended to read as follows:

   (1) Maximum Power Level

   TVA is authorized to operate the facility at reactor core power levels not in excess of 3459 megawatts thermal.

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A as revised through Amendment No. 42 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented prior to completion of the WBN, Unit 2 fall 2020 refueling outage no later than December 15, 2020.

FOR THE NUCLEAR REGULATORY COMMISSION

Caroline L. Carusone

Digitally signed by Caroline L. Carusone
Date: 2020.10.21 11:32:16 -04'00'

Caroline L. Carusone, Deputy Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Facility Operating License
and Technical Specifications

Date of Issuance: October 21, 2020

ATTACHMENT TO AMENDMENT NO. 42

WATTS BAR NUCLEAR PLANT, UNIT 2

FACILITY OPERATING LICENSE NO. NPF-96

DOCKET NO. 50-391

Replace page 3 of Facility Operating License No. NPF-96 with the attached revised page 3. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
1.1-6	1.1-6
5.0-30	5.0-30
5.0-33	5.0-33

C. The license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act, and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified or incorporated below.

(1) Maximum Power Level

TVA is authorized to operate the facility at reactor core power levels not in excess of 3459 megawatts thermal.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 42 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, are hereby incorporated into this license. TVA shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) TVA shall implement permanent modifications to prevent overtopping of the embankments of the Fort Loudon Dam due to the Probable Maximum Flood by June 30, 2018.

(4) PAD4TCD may be used to establish core operating limits until the WBN Unit 2 steam generators are replaced with steam generators equivalent to the existing steam generators at WBN Unit 1.

(5) By December 31, 2019, the licensee shall report to the NRC that the actions to resolve the issues identified in Bulletin 2012-01, "Design Vulnerability in Electrical Power System," have been implemented.

(6) The licensee shall maintain in effect the provisions of the physical security plan, security personnel training and qualification plan, and safeguards contingency plan, and all amendments made pursuant to the authority of 10 CFR 50.90 and 50.54(p).

(7) TVA shall fully implement and maintain in effect all provisions of the Commission approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The TVA approved CSP was discussed in NUREG-0847, Supplement 28, as amended by changes approved in License Amendment No. 7.

(8) TVA shall implement and maintain in effect all provisions of the approved fire protection program as described in the Fire Protection Report for the facility, as described in NUREG-0847, Supplement 29, subject to the following provision:

1.1 Definitions (continued)

QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3459 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level.
SLAVE RELAY TEST	A SLAVE RELAY TEST shall consist of energizing each slave relay and verifying the OPERABILITY of each slave relay. The SLAVE RELAY TEST shall include, as a minimum, a continuity check of associated testable actuation devices.

(continued)

5.9 Reporting Requirements (continued)

5.9.3 Radioactive Effluent Release Report

---NOTE---
A single submittal may be made for a multiple unit station. The submittal should combine sections common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.9.4 Reserved for Future Use

5.9.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to the initial and each reload cycle, or prior to any remaining portion of a cycle, and shall be documented in the COLR for the following:

LCO 3.1.4	Moderator Temperature Coefficient
LCO 3.1.6	Shutdown Bank Insertion Limits
LCO 3.1.7	Control Bank Insertion Limits
LCO 3.2.1	Heat Flux Hot Channel Factor
LCO 3.2.2	Nuclear Enthalpy Rise Hot Channel Factor
LCO 3.2.3	Axial Flux Difference
LCO 3.9.1	Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102% RTP is specified in a previously approved method, 100.6% RTP may be used only when feedwater flow measurement (used as input for reactor thermal power measurement) is provided by the leading edge flowmeter (LEFM) as described in document number 11 listed below. When feedwater flow measurements from the LEFM are unavailable, the originally approved initial power level of 102% RTP (3411 MWt) shall be used. The approved analytical methods are, specifically those described in the following documents:

(continued)

5.9 Reporting Requirements

5.9.5 CORE OPERATING LIMTS REPORT (COLR) (continued)

11. Caldon, Inc., Engineering Report-80P, “Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System,” Revision 0, March 1997; and Caldon Ultrasonics Engineering Report ER-157P-A, “Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or LEFM CheckPlus System,” Revision 8 and Revision 8 errata.

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

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

ENCLOSURE 2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 42 TO FACILITY OPERATING LICENSE NO. NPF-96

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-391

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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 42 TO FACILITY OPERATING LICENSE NO. NPF-96

TENNESSEE VALLEY AUTHORITY

WATTS BAR NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-391

1.0 INTRODUCTION

By letter dated October 10, 2019 (Reference 1), as supplemented by letters dated April 29, 2020 (Reference 2), July 27, 2020 (Reference 3), and August 28, 2020 (Reference 4), the Tennessee Valley Authority (TVA, the licensee) submitted a request for changes to the Watts Bar Nuclear Plant (WBN), Unit 2, Operating License (OL) and Technical Specifications (TSs). The requested changes would increase the authorized reactor core power level by approximately 1.4 percent rated thermal power from 3411 megawatts thermal (MWt) to 3459 MWt, based on the use of the existing Caldon®[1] Leading Edge Flow Meter (LEFM®) CheckPlus System; revise OL Item 2.C.(1) and TS 1.1, "Definitions," to reflect the increase in rated thermal power; revise TS 5.9.5, "Core Operating Limits Report (COLR)," to allow that 100.6 percent rated thermal power may be used only when feedwater flow measurement is provided by the LEFM as described in the NRC-approved Caldon Topical Reports for LEFMs; and add the NRC-approved Caldon Topical Reports for LEFMs as document number 11 in the list of documents in TS 5.9.5b.

The supplements dated April 29, 2020, July 27, 2020, and August 28, 2020, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC, Commission) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 3, 2020 (85 FR 12581).

2.0 REGULATORY EVALUATION

2.1 MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATES

Nuclear power plants are licensed to operate at a specified maximum core thermal power, often called rated thermal power (RTP). Appendix K, "ECCS [Emergency Core Cooling System]

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[1] Caldon, Inc. (Caldon) is now part of the Measurement Systems Division of Cameron International Corporation (Cameron). Caldon and LEFM are registered trademarks of Cameron. Cameron, operating as Cameron Technologies US, LLC in the U.S., is now a wholly owned subsidiary of Sensia, LLC. This safety evaluation refers to Caldon and Cameron interchangeably.

Evaluation Models," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 formerly required 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 accident (LOCA) and ECCS analyses. This requirement was included to ensure that instrumentation uncertainties were adequately accounted for in the safety analyses. In practice, many of the design basis analyses assumed a 2 percent power uncertainty, consistent with 10 CFR Part 50, Appendix K.

A change to 10 CFR Part 50, Appendix K, was published in the *Federal Register* on June 1, 2000 (65 FR 34913), which became effective July 31, 2000. This change allows licensees to use a power level less than 1.02 times the RTP for the LOCA and ECCS analyses, but not a power level less than the licensed power level, based on the use of state-of-the-art feedwater (FW) flow measurement devices that provide a more accurate calculation of power. Licensees can use a lower uncertainty in the LOCA and ECCS analyses provided that the licensee has demonstrated that the proposed value adequately accounts for instrumentation uncertainties. Substantial conservatism remains in other Appendix K requirements such that sufficient margin to ECCS performance in the event of a LOCA is preserved. However, this change to 10 CFR Part 50, Appendix K, did not authorize increases in licensed power levels for individual nuclear power plants. Therefore, any licensee wishing to increase its licensed power level must request an amendment to its license in accordance with 10 CFR 50.90, "Application for amendment of license, construction permit, or early site permit."

WBN, Unit 2 is currently licensed to operate at a maximum power level of 3411 MWt, with a 2 percent margin in the ECCS evaluation model to allow for uncertainties in RTP measurement. The license amendment request (LAR) would reduce this uncertainty to 0.6 percent.

In order to provide guidance to licensees seeking a measurement uncertainty recapture (MUR) power uprate on the basis of improved FW flow measurement, the NRC issued Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002 (Reference 5). RIS 2002-03 provides guidance to licensees on the scope and detail of the information that should be provided to the NRC staff for MUR power uprate LARs. While RIS 2002-03 does not constitute an NRC requirement, its use aids licensees in the preparation of their MUR power uprate LAR, while also providing guidance to the NRC staff for the conduct of its review. The licensee stated in its LAR that its submittal followed the guidance of RIS 2002-03.

2.2 IMPLEMENTATION OF AN MUR POWER UPRATE AT WBN, UNIT 2

In existing nuclear power plants, the neutron flux instrumentation continuously indicates the RTP. This instrumentation must be periodically calibrated to accommodate the effects of fuel burnup, flux pattern changes, and instrumentation setpoint drift. The RTP generated by a nuclear power plant is determined by steam plant calorimetry, which is the process of performing a heat balance around the nuclear steam supply system (called a calorimetric). The accuracy of this calculation depends primarily upon the accuracy of FW flow rate and FW net enthalpy measurements. As such, an accurate measurement of FW flow rate and temperature is necessary for an accurate calibration of the nuclear instrumentation. Of the two parameters, flow rate and temperature, the most important in terms of calibration sensitivity is the FW flow rate.

The instruments originally installed to measure FW flow rate in existing nuclear power plants were usually a venturi or a flow nozzle, each of which generates a differential pressure proportional to the FW velocity in the pipe. However, errors in the determination of flow rate can be introduced due to venturi fouling and, to a lesser extent, flow nozzle fouling, the transmitter, and the analog-to-digital converter.[2] As a result of the desire to reduce flow instrumentation uncertainty to enable operation of the plant at a higher power while remaining bounded by the accident analyses, the industry assessed alternate flow rate measurement techniques and found that ultrasonic flow meters (UFMs) are a viable alternative. UFMs are based on computer-controlled electronic transducers that do not have differential pressure elements that are susceptible to fouling.

The licensee intends to use UFMs developed by Caldon, specifically, the LEFM CheckPlus System, which provides a more accurate indication of FW flow (and, correspondingly, RTP) than available during the original development of 10 CFR Part 50, Appendix K, requirements, which did not explicitly consider uncertainties and prescribed a required power level of 102 percent for accident analyses. Use of these UFMs to measure FW flow would allow the licensee to operate the plant with a reduced instrument uncertainty margin and an increased power level in comparison to its currently licensed RTP.

The Caldon LEFM CheckPlus System took several years to develop. Caldon submitted a topical report in March of 1997, Engineering Report (ER)-80P, Revision 0, “Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM System,” that describes the LEFM and includes calculations of power measurement uncertainty obtained using a Check system in a typical two-loop pressurized-water reactor (PWR) or a two-FW-line boiling-water reactor (Reference 6). This topical report also provides guidance for determining plant-specific power calorimetric uncertainties. The NRC staff approved the use of this topical report for an exemption to the 2 percent uncertainty requirements in 10 CFR Part 50, Appendix K, in a safety evaluation (SE) dated March 8, 1999, which allowed a 1 percent power uprate using the LEFM (Reference 7). Following the publication of the changes to 10 CFR Part 50, Appendix K, which allowed for an uncertainty less than 2 percent, Caldon submitted Topical Report ER-160P, Revision 0, “Supplement to Engineering Report ER-80P: Basis for a Power Uprate with the LEFM System,” a supplement to ER-80P (Reference 8). The NRC staff approved ER-160P by letter dated January 19, 2001, for use in a power uprate of up to 1.4 percent at Watts Bar Nuclear Plant, Unit 1 (Reference 9). Subsequently, in an SE dated December 20, 2001, the NRC staff approved ER-157P, Revision 5, “Supplement to Engineering Report ER-80P: Basis for a Power Uprate with the LEFM or LEFM CheckPlus System,” for use in a power uprate of up to 1.7 percent using the CheckPlus System (Reference 10). By letter dated August 16, 2010, the NRC staff approved ER-157P, Revision 8, and associated errata (Reference 11). ER-157P, Revision 8, corrects minor errors in Revision 5, provides clarifying text, and incorporates revised analyses of coherent noise, non-fluid delays, and transducer replacement (Reference 12). It also adds two new appendices, Appendix C and Appendix D, which describe the assumptions and data that support the coherent noise and transducer replacement calculations, respectively.

WBN, Unit 2 has a Caldon LEFM CheckPlus ultrasonic multi-path, transit time flow meter installed in the main feedwater (MFW) header as well as feedwater flow venturis, which will be used if the LEFM system is not functional. The LEFM system uses ultrasonic transit time principles to determine fluid velocity and sound velocity. The WBN, Unit 2 LEFM CheckPlus ultrasonic flow meter system consists of an electronic cabinet located in the WBN, Unit 2 auxiliary instrument room, one measurement section/spool piece located in the Turbine

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[2] “Venturi” will generally be used in the remainder of this document to reference both venturi and flow nozzles.

Building, and associated cabling. The LEFM CheckPlus System is further described in SE Section 3.1.2.

2.3 LICENSEE'S PROPOSED CHANGES

The licensee requested the following changes:

- WBN, Unit 2 OL Item 2.C.(1) would be revised to increase the maximum core power level from 3411 MWt to 3459 MWt.

- The definition of RTP in TS 1.1, "Definitions," would be revised to account for the increase in reactor core thermal power level as follows:

  > RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3459 MWt.

- TS 5.9.5, "CORE OPERATING LIMITS REPORT (COLR)," paragraph b, which currently states:

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

  would be revised as follows:

  > The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102% RTP is specified in a previously approved method, 100.6% RTP may be used only when feedwater flow measurement (used as input for reactor thermal power measurement) is provided by the leading edge flowmeter (LEFM) as described in document number 11 listed below. When feedwater flow measurements from the LEFM are unavailable, the originally approved initial power level of 102% RTP (3411 MWt) shall be used. The approved analytical methods are specifically those described in the following documents:

- The NRC-approved Caldon Topical Reports for LEFMs would be added as document number 11 in the list of documents in TS 5.9.5b as follows:

  > 11. Caldon, Inc., Engineering Report-80P, "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM√™ System," Revision 0; and Caldon Ultrasonics Engineering Report ER-157P-A, "Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or LEFM CheckPlus System," Revision 8 and Revision 8 errata.

2.4 REGULATORY REQUIREMENTS AND GUIDANCE

Appendix K to 10 CFR Part 50 requires, in part, that power reactor licensees establish required and acceptable features of evaluation models for heat removal by the ECCS after the blowdown phase of a LOCA.

Pursuant to 10 CFR 50.57, “Issuance of operating license,” the Commission may issue an operating license upon finding, in part, that there is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the NRC’s regulations.

Chapter 3 of the WBN updated final safety analysis report (UFSAR) states the following:

> The Watts Bar Nuclear Power plant was designed to meet the intent of the “Proposed General Design Criteria for Nuclear Power Plant Construction Permits” published in July, 1967. The Watts Bar construction permit was issued in January, 1973. This UFSAR, however, addresses the NRC General Design Criteria (GDC) published as Appendix A to 10 CFR [Part] 50 in July, 1971, including Criterion 4 as amended October 27, 1987.

Therefore, the NRC staff considered the following GDC requirements as part of its review of the WBN, Unit 2 LAR.

Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50:

- GDC 10, “Reactor design,” which requires the reactor core and associated coolant, control, and protection systems to be 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.

- GDC 20, “Protection system functions,” which requires the protection system to be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

RIS 2002-03, “Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications,” provides guidance to licensees on the scope and detail of the information that should be provided to NRC for reviewing MUR power uprate applications.

Regulatory Guide (RG) 1.105, Revision 3, “Setpoints for Safety-Related Instrumentation,” describes a method acceptable to the NRC staff for complying with the NRC regulations for ensuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits (Reference 13). RG 1.105 endorses Part I of Instrument Society of America[3] (ISA)-S67.04-1994, “Setpoints for Nuclear Safety-Related Instrumentation” (Reference 14). The NRC staff used this guide to establish the adequacy of the licensee’s setpoint calculation methodologies and the related plant surveillance procedures. Section 4.4, “Combination of Uncertainty,” of ISA-S67.04-1994 states that the methods in Subsection 4.4.1,

[3] In 2008, the name of this organization was changed to the International Society of Automation.

"Square-Root-Sum-of-Squares (SRSS) method," and in Subsection 4.4.2, "Algebraic methods," are acceptable for combining uncertainties. This section also states that alternate methods, including probabilistic or stochastic modeling, or a combination of SRSS and algebraic methods may also be used.

Additional regulatory requirements and guidance documents are listed within the specific SE sections in which they are applicable below.

3.0 TECHNICAL EVALUATION

3.1 FEEDWATER FLOW MEASUREMENT TECHNIQUE AND POWER MEASUREMENT UNCERTAINTY

3.1.1 System Description and Background

As stated above, early revisions of 10 CFR Part 50, Appendix K, required licensees to base their LOCA and ECCS analyses on an assumed power level of at least 102 percent of the licensed power level to account for power measurement uncertainty. The NRC later amended 10 CFR Part 50, Appendix K, to permit licensees to justify a smaller margin for power measurement uncertainty. Licensees may apply the reduced margin to operate the plant at a power level higher than the previously licensed power level. In its LAR, the licensee proposes to credit the WBN, Unit 2 LEFM CheckPlus ultrasonic flow meter system to revise the licensed power measurement uncertainty to achieve an increase of approximately 1.4 percent in the licensed power level.

3.1.2 Leading Edge Flow Meter Technology and Measurement

The Cameron LEFM CheckPlus System uses a transit time methodology to measure fluid velocity. The feedwater temperature is also measured by the LEFM. The basis of the transit time methodology for measuring fluid velocity and temperature is that ultrasonic pulses transmitted through a fluid stream travel faster in the direction of the fluid flow than through the opposite flow. The difference in the upstream and downstream traversing times of the ultrasonic pulse is proportional to the fluid velocity in the pipe. The temperature is determined using a correlation between the mean propagation velocity of the ultrasound pulses in the fluid and the fluid pressure.

Instead of a single path, the system uses multiple diagonal acoustic paths, allowing velocities measured along each path to 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. The velocity profile correction factor is derived from calibration testing of the LEFM CheckPlus System in a plant-specific piping model at a calibration laboratory.

In the LAR, the licensee stated the following:

> The WBN Unit 2 LEFM CheckPlus ultrasonic flow meter system consists of an electronic cabinet located in the WBN Unit 2 auxiliary instrument room, one measurement section/spool piece located in the Turbine Building, and associated cabling. The measurement section/spool piece is installed in the 32-inch main feedwater header. The measurement spool piece contains 16 ultrasonic, multi-path, transit time transducers grouped into the two planes of eight

transducers each. The LEFM spool piece is located upstream of the existing nozzle venturis, which are located in the feedwater lines to the individual steam generators (SGs).

Validated LEFM data including calculated results, status, and signal process information is sent to the integrated computer system (ICS). The ICS provides an audible and visual alarm upon a change in LEFM system status. The LEFM system instrumentation is not safety-related. The LEFM system was designed and manufactured per Cameron's Quality Assurance Program.

The LEFM CheckPlus System performs automatic continuous self-checking of the transducer signals and the LEFM calculation results. This testing provides verification that the digital circuits are operating correctly, and the system is within its specified accuracy envelope. These processes can identify failure conditions that will cause the LEFM to switch from the NORMAL mode to the MAINTENANCE mode or to the FAIL mode.

The Cameron LEFM CheckPlus System has two operating modes (i.e., NORMAL and MAINTENANCE) and a FAIL mode.

- NORMAL mode: The LEFM CheckPlus System NORMAL status is displayed when the feedwater flow, temperature, and header pressure signals are normal and operating within design limits. The LEFM CheckPlus System measures the average flow of two independent LEFM Check subsystems. Each LEFM Check subsystem consists of four acoustic paths (i.e., a total of eight paths) that comprise the LEFM CheckPlus System. Calculated power level uncertainty associated with the LEFM flow measuring system in this condition is less than 0.6 percent. Per the LAR, the plant will be able to operate at up to 3459 MWt when in the NORMAL mode.

- MAINTENANCE mode: In MAINTENANCE mode, only one of the two LEFM Check subsystems is fully operational, and the uncertainty for that meter is increased. In MAINTENANCE mode, the system must be restored to NORMAL mode within 72 hours or the power level uncertainty reverts to the 2 percent associated with the venturi flow meters and, therefore, power will be required to be reduced to 3411 MWt (i.e., the pre-MUR power uprate level).

- FAIL mode: In FAIL mode, the LEFM CheckPlus System indicates a loss of function. In this case, the power level uncertainty reverts to the 2 percent associated with the venturi flow meters and, therefore, power will be required to be reduced to 3411 MWt (i.e., the pre-MUR power uprate level) within 24 hours if LEFM functionality cannot be restored to the NORMAL mode.

3.1.3 Reason for Proposed Changes

Based on the LEFM instrumentation, the licensee asserts that it is able to determine the core power level with a power measurement uncertainty value (rounded up) of 0.6 percent RTP. The licensee is requesting a license amendment based on the requirements of 10 CFR Part 50, Appendix K, to revise the licensed power uncertainty to achieve an increase of approximately 1.4 percent in the licensed power level. This would increase the licensed power output from

3411 MWt to 3459 MWt for WBN, Unit 2 and align it with the existing power level for WBN, Unit 1.

Because WBN, Unit 2 is only requesting a 1.4 percent power uprate, WBN, Unit 2 can operate at 3459 MWt with only one LEFM Check subsystem operational. However, a 72-hour allowed outage time (or Completion Time (CT)) is currently proposed for when the LEFM CheckPlus System is in MAINTENANCE mode.

The licensee provided WCAP-18419-P, Revision 1, as Enclosure 6 to the LAR to support its request. A non-proprietary version was provided as Enclosure 10. This document describes techniques to address instrumentation uncertainties associated with the LEFM. The techniques include the use of Westinghouse setpoint methodology to calculate the uncertainty components and describes a method for performing daily calorimetric power measurement. In this method, the ICS uses data supplied by the LEFM to calculate operating thermal power level.

The NRC staff reviewed 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 through H) provided in Section I of Attachment 1 to RIS 2002-03, which relates to 10 CFR Part 50, Appendix K. The NRC staff confirmed that the licensee's implementation of the proposed FW flow measurement device is consistent with staff-approved Topical Reports ER-80P and ER-157P and that the licensee adequately addressed the additional requirements as described in SE Section 3.1.5.2.

The NRC staff also reviewed the power measurement uncertainty calculations to ensure that:

(1) The proposed uncertainty value of 0.6 percent correctly accounts for all uncertainties associated with power level instrumentation errors, and

(2) the uncertainty calculations meet the relevant requirements of 10 CFR Part 50, Appendix K, and as described in Section 2.4 of this SE.

3.1.4 <u>Evaluation of Proposed Total Thermal Power Uncertainty Value up to 0.6 Percent</u>

3.1.4.1 Licensee's Methodology

The licensee requested a 1.4 percent power uprate for WBN, Unit 2 based on the use of the LEFM CheckPlus System as referenced in WCAP-18419-P. As described in Section 2, "Methodology," of the WCAP-18419-P, the method statistically combines inputs to determine the overall instrument uncertainty. Channel statistical allowances are calculated for the instrument channels. Dependent parameters are arithmetically combined to form statistically independent groups, which are then added to the SRSS for the remaining random parameters to determine the overall uncertainty. The relationships between the uncertainty components and the instrument channel uncertainty allowance are variations of the basic Westinghouse Setpoint Methodology, and the Channel Statistical Allowance was calculated by Equations 1 – 4 in Enclosure 6 of the LAR.

The vendor's determination of the uncertainty of the Cameron LEFM CheckPlus System is consistent with this methodology, as described in the referenced NRC-approved topical reports. In addition, the NRC staff verified that this setpoint methodology is consistent with RG 1.105, and ISA-S67.04-1994, Section 4.4, “Combination of uncertainties,” and its subsections 4.4.1 and 4.4.2, and, therefore, is acceptable.

3.1.4.2 Instrumentation Uncertainties Evaluation

WCAP-18419-P, Section 3.1.1, “Using Leading Edge Flow Meter (LEFM) Installed in Feedwater Header,” provides that the LEFM, which was installed in the FW header, performs the daily calorimetric power measurements and its overall uncertainty is given as 0.48 percent flow.

WCAP-18419-P, Tables 1, 2, and 3 show the results of the uncertainty calculations and the sensitivities by using the LEFM. Using the power uncertainty values noted in Table 3, the power uncertainty was analyzed and calculated based on four loops and the instrument uncertainties and was proposed to be 0.6 percent.

In the LAR, the licensee indicated that:

- In Cameron report ER-734P, the thermal power uncertainty results of the Westinghouse analysis are conservative and bound the use of the LEFM in both NORMAL and MAINTENANCE modes.
- The proposed power uprate of 1.4 percent will be consistent with the existing power level for WBN, Unit 1.

RIS 2002-03, Attachment 2, states that:

> The term “bounded” is used to refer to areas where the existing analyses of record establish continued acceptability of operation at the proposed uprated power level without the need for re-analysis.

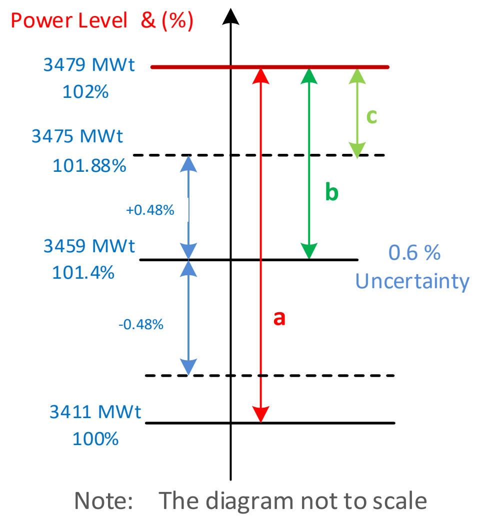
In the LEFM NORMAL mode, the plant would be able to operate at up to 3459 MWt. If an LEFM flow meter is in MAINTENANCE mode, then the uncertainty for that meter is increased and a 72-hour CT would be used prior to reducing power to the pre-MUR power level of 3411 MWt. If an LEFM flow meter is in the FAIL mode, then the power must be reduced to the pre-MUR power level within 24 hours. The 24 and 72 hour CTs are discussed in Section 3.1.5.2 of this SE.

The licensee provided calculations to support an LEFM flow uncertainty of 0.48 percent, based on a bounding uncertainty analysis (Enclosure 6 of LAR). This indicates that the proposed total thermal power uncertainty of 0.6 percent bounds the assumed flow uncertainty value of 0.48 percent.

The NRC staff independently verified that the proposed uncertainty of 0.6 percent is more conservative than the flow uncertainty of 0.48 percent. The proposed power increase of 101.4 percent RTP with an uncertainty of 0.6 percent is consistent with the relevant guidance and satisfies the relevant regulatory requirements as described in Section 2.4 of this SE and, therefore, is acceptable. The WBN, Unit 2 power level relationships are illustrated in Figure 1 below.

Figure 1: WBN, Unit 2 Power Level Relationships

Watts Bar Power Level Relationships



Margin Calculation

Row #		% Uncertainty	(MWt)
1	a	2.00	68.00
2	b	0.60	20.00
3	c = 0.6% - 0.48%	0.12	4.00
			%
4	% Margin = (b/a)*100		29.41
5	% Margin = (c/a)*100		5.88

a: Margin between 102% power and original (100%) power
b : Margin between 102% power and power increase with 0.6% uncertainty
c: Margin between 102% power and power increase with 0.48% uncertainty

The NRC staff, therefore, made the following determinations with respect to the proposed power increase of 101.4 percent RTP (with a proposed total uncertainty of 0.6 percent) for the WBN, Unit 2:

- The value of 3479 MWt, which corresponds to 102 percent of RTP, is greater than 3459 MWt (which corresponds to 101.4 percent of RTP with the uncertainty of 0.6 percent). This indicates that the 102 percent of RTP remains the bounding power level for MUR power uprate conditions when total uncertainty of 0.6 percent is applied.

- The margin percentage between margins “b” and “a” (29.41 percent in Row 4) is adequate. This margin ensures that the total loop uncertainty has been chosen to ensure that a trip or safety actuation will occur significantly before the measured process reaches the power limit (102 percent of RTP).

- This percentage margin is greater than the percentage margin between margins “c” and “a” (5.88 percent in Row 5). That showed that the total loop uncertainty of 0.6 percent is more conservative than the 0.48 percent flow uncertainty since the reactor will be operating below the calculated power level. Therefore, the licensee chose 0.6 percent instead of 0.48 percent for uncertainty.

- In addition, the proposed power level of 3459 MWt, which corresponds to 101.4 percent of RTP with an uncertainty of 0.6 percent, will align the WBN, Unit 2 TS to the WBN, Unit 1 TS.

The NRC staff verified that the proposed total loop uncertainly that has been chosen (0.6 percent) is achieved by the LEMF system operating with the bounds of its uncertainty analysis (0.48 percent). This 0.6 percent uncertainty is consistent with NRC RIS 2002-03 and satisfies the regulatory requirements of 10 CFR 50, Appendix K, and 10 CFR 50.36(c)(1)(ii)(A).

Therefore, the NRC staff concludes that the proposed power increase of 101.4 percent RTP with total uncertainty of 0.6 percent is acceptable.

3.1.5 Evaluation of License Amendment Request Compliance with RIS 2002-03, Attachment 1, Section I, Items A through H

The NRC staff reviewed the requested MUR power uprate based on the LEFM CheckPlus technology and RIS 2002-03, as described below.

3.1.5.1 Items I.1.A through I.1.C, Attachment 1 to RIS 2002-03

Items A, B, and C in Section I of Attachment 1 to RIS 2002-03 guide licensees in identifying the approved topical reports, providing references to the NRC's approval of the measurement technique, and discussing the plant-specific implementation of the guidelines in the topical report and the NRC staff's approval of the FW flow measurement technique, respectively.

In its LAR, the licensee identified Topical Reports ER-80P and ER-157P as applicable to the Cameron LEFM CheckPlus System. The licensee also referenced NRC SEs for Topical Report ER-80P and Topical Report ER-157P (References 7 and 11, respectively).

In its response to item I.1.C, the licensee stated that the LEFM CheckPlus System is installed in WBN, Unit 2 according to the appropriate Cameron installation and testing procedures.

Based on its review of the licensee's submittals as discussed above, the NRC staff determined that the licensee has sufficiently addressed the plant-specific implementation of the Cameron LEFM CheckPlus System using proper topical report guidelines. Therefore, the licensee's description of the FW flow measurement technique and implementation of the power uprate using this technique follows the guidance in Items A through C of Section I of Attachment 1 to RIS 2002-03 and meets the regulatory requirements of 10 CFR Part 50, Appendix K, and 10 CFR 50.36(c)(1)(ii)(A).

3.1.5.2 Item I.1.D, Attachment 1 to RIS 2002-03

Item D in Section I of Attachment 1 to RIS 2002-03 guides licensees in addressing nine criteria that the NRC staff included in its SEs for Topical Reports ER-80P and ER-157P when implementing the FW flow measurement uncertainty technique. The NRC staff's SEs for Topical Reports ER-80P and ER-157P include these nine plant-specific criteria to be addressed by a licensee referencing these topical reports for power uprate (References 7 and 11, respectively). The licensee's submittal addresses each of the criteria as follows:

Criterion 1 from the SE for ER-80P

> 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 instrumentation and the effect on thermal power measurement and plant operation.

The licensee stated that necessary procedures and documents required for maintenance and calibration of the LEFM CheckPlus System have been implemented to ensure that the system is properly maintained and calibrated. Preventive maintenance scope and frequency is based on vendor recommendations and performance data reviews. Transducers are replaced as determined to be necessary by a review of the equipment's performance history by the LEFM system vendor.

For instrumentation other than the LEFM system that contributes to the power calorimetric computation, calibration and maintenance is performed periodically using existing site procedures. Maintenance and test equipment, setting tolerances, calibration frequencies, and instrumentation accuracy were evaluated and accounted for within the thermal power uncertainty calculations.

Contingency plans for plant operation with an inoperable LEFM are discussed in "Criterion 1 from the SE for ER-157P" of this SE below.

Based on its review of the licensee's submittal, the NRC staff concludes that the licensee adequately addressed Criterion 1 from the SE for ER-80P.

Criterion 2 from the SE for ER-80P

> 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 stated that the WBN, Unit 2 LEFM CheckPlus System installed instrumentation is bounded by the analysis and assumptions set forth in Topical Report ER-80P.

The WBN, Unit 2 LEFM system is currently installed and has been highly reliable. The licensee reviewed the maintenance history of the LEFM system since plant startup (October 2016). No significant LEFM system failures or repairs were identified. However, an issue was identified that resulted in the LEFM system being in MAINTENANCE mode for an extended period.

The licensee stated that preventive maintenance scope and frequency is based on vendor recommendations and performance data reviews. Transducers will continue to be replaced as determined to be necessary. The licensee further stated that the operational and maintenance history of these components shows that the system is reliable for FW flow measurement and thermal power calculations.

Based on its review of the licensee's submittal, the NRC staff concludes that the licensee adequately addressed Criterion 2 from the SE for ER-80P.

Criterion 3 from the SE for ER-80P

> 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 alternative methodology is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

The licensee stated that the WBN, Unit 2 LEFM uncertainty calculation methodology is based on the American Society of Mechanical Engineers (ASME) Performance Test Code (PTC) 19.1-1985, "Measurement Uncertainty" (Reference 15); International Society of

Automation (ISA) Recommended Practice (RP) ISA-RP67.04, Part II, "Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation" (Reference 16); and Alden Research Laboratory, Inc. calibration tests. The methodology used to calculate the LEFM uncertainty is described in Cameron reports ER-80P and ER-157P. The ISA-RP67.04, Part II methodology has been used for instrument uncertainty calculations for multiple MUR power uprates that were accepted by the NRC. The WBN, Unit 2 LEFM uncertainty analysis is provided in Cameron report ER-734P (Enclosure 5 to the LAR). A non-proprietary version of this uncertainty analysis, ER-734NP, is provided as Enclosure 9 to the LAR.

Based on its review of the licensee's submittal, the NRC staff concludes that the licensee adequately addressed Criterion 3 from the SE for ER-80P.

<u>Criterion 4 from the SE for ER-80P</u>

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

The license stated that this criterion does not apply to WBN, Unit 2 because the flow elements were tested and calibrated in a full-scale model of the WBN hydraulic geometry at the Alden Research Laboratory. An Alden data report for these tests and a Cameron Meter Factor Calculation and Accuracy Assessment (ER-732P) evaluating the test data were prepared (Enclosure 5 to the LAR). A non-proprietary version of this report, ER-732NP, is provided as Enclosure 9 to the LAR. The Cameron engineering report (ER-734P) includes a bounding calibration factor for the WBN, Unit 2 spool piece that was established by these tests. The piping configuration at WBN, Unit 2 remains bounded by the original LEFM installation and calibration assumptions as analyzed in Cameron engineering reports ER-80P and ER-157P.

In addition, a bounding LEFM uncertainty has been used in the total thermal power uncertainty calculation described in Section I.1.E of Enclosure 2 to the LAR and the site-specific uncertainty analyses are provided in Enclosures 5 and 6 to LAR.

Based on its review of the licensee's calibration data in Cameron Engineering Reports ER-732P and ER-734P, the NRC staff finds that Criterion 4 from the SE for ER-80P is not applicable to the WBN, Unit 2 LAR.

<u>Criterion 1 from the SE for ER-157P</u>

> Continued operation at the pre-failure power level for a pre-determined time and the decrease in power that must occur following that time are plant-specific and must be acceptably justified.

The licensee stated that: "Similar to WBN Unit 1, if a non-functional LEFM for WBN, Unit 2 is not restored to functional status prior to the next performance of TS SR 3.3.1.2,[4] which is performed every 24 hours, then Unit 2 power will be reduced to no more than 3411 MWt (i.e., the current licensed thermal power…)." The licensee also stated that the basis for the proposed CT of "prior to the next performance of TS SR 3.3.1.2," which would potentially allow for a maximum of 24 hours prior to reducing power, is the same CT that is used for WBN, Unit 1.

Additionally, the licensee stated that the redundancy inherent in the two measurement planes of an LEFM CheckPlus System makes the system tolerant to component failures, as compared to the Check system installed for WBN, Unit 1, which only has one measurement plane. Moreover, the licensee is only proposing a 1.4 percent power uprate for WBN, Unit 2 in order to establish consistency with WBN, Unit 1. The licensee also stated that, "WBN Unit 2 can operate at 3459 MWt with only one LEFM Check subsystem operational; however, a 72-hour allowed outage time is currently proposed for when the LEFM CheckPlus System is in MAINTENANCE mode…."

The LEFM CheckPlus System has three modes: NORMAL, MAINTENANCE, and FAIL. As described in Section 3.1.4.2 of this SE, if an LEFM flow meter is in MAINTENANCE mode, the uncertainty for that meter is increased and a 72-hour CT will be used prior to reducing power to the pre-MUR power level of 3411 MWt. When the LEFM flow meter is in FAIL mode, the power level will be reduced to 3411 MWt within 24 hours, which is consistent with WBN, Unit 1 TS SR 3.3.1.2.

The 72-hour CT for the LEFM flow meter being in MAINTENANCE mode prior to restoring it to NORMAL mode and the 24-hour CT for a failed LEFM prior to reducing the power level to 3411 MWt are acceptable because the existing FW flow nozzle-based signals will be calibrated to the last validated data from the LEFM system during this period. Any drift of the FW flow nozzle measurements due to fouling would result in a higher than actual indication of feedwater flow and an overestimation of the calculated calorimetric power level. This is conservative because the reactor will be operating below the calculated power level. A sudden de-fouling event during the 24-hour period is unlikely and any significant sudden de-fouling would be detected by other plant parameters.

Moreover, the NRC has approved a 72-hour CT for previous MUR power uprate applications, which is consistent with the proposed maximum allowed outage time of 72 hours for Shearon Harris Nuclear Power Plant, Unit 1 (Reference 17), and Waterford Steam Electric Station, Unit 3, River Bend Station, Unit 1, and Grand Gulf Nuclear Station, Unit 1 (Reference 10).

Thus, over a 72-hour period, FW flow nozzle instrument drift would have an insignificant effect on the FW flow measurement.

Based on its review of the licensee's submittal, the NRC staff concludes that the licensee adequately addressed Criterion 1 from the SE for ER-157P.

[4] WBN, Unit 1 TS SR 3.3.1.2 states, "Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output," at a frequency in accordance with the Surveillance Frequency Control Program, as revised by Amendment No. 132, issued on February 28, 2020.

Criterion 2 from the SE for ER-157P

> A CheckPlus operating with a single failure is not identical to an LEFM Check. Although the effect on hydraulic behavior is expected to be negligible, this must be acceptably quantified if a licensee wishes to operate using the degraded CheckPlus at an increased uncertainty.

The licensee stated that when the LEFM CheckPlus meter has only one of its two LEFM Check subsystems fully operational, resulting in that meter computing flow from just the remaining operational LEFM Check subsystem, that LEFM flow meter is considered to be in the Maintenance mode. This status is indicated to operators on the ICS in the control room. The total thermal power uncertainties for the three LEFM flow meter conditions have been quantified for the WBN, Unit 2 application in accordance with the methodology of ER-157P. The total thermal power uncertainties for the WBN, Unit 2 LEFM CheckPlus System in NORMAL and MAINTENANCE modes have been quantified in accordance with the Cameron ER-734P, Revision 2.

Based on its review of the licensee's submittals, the NRC staff concludes that the licensee adequately addressed Criterion 2 from the SE for ER-157P.

Criterion 3 from the SE for ER-157P

> An applicant with a comparable geometry can reference the above Section 3.2.1 [of Reference 11] finding to support a conclusion that downstream geometry does not have a significant influence on CheckPlus calibration. However, CheckPlus test results do not apply to a Check and downstream effects with use of a CheckPlus with disabled components that make the CheckPlus comparable to a Check must be addressed. An acceptable method is to conduct applicable Alden Laboratory tests.

In the NRC letter to McGuire Nuclear Station, the NRC staff determined that for conditions in which the CheckPlus System is operating with one or more transducers out of service, the effect of downstream piping should be addressed if the separation distance from the meter transducers to the downstream piping change is less than five pipe diameters (Reference 18). At WBN, Unit 2, the LEFM flow meters are installed upstream of the FW flow nozzles, and the distance from meter transducers to downstream piping changes, i.e., venture contraction, is greater than five pipe diameters in each FW line. Therefore, the downstream geometries for WBN, Unit 2 do not have a significant influence on CheckPlus calibration. This criterion is, therefore, not applicable to the WBN, Unit 2 application.

Criterion 4 from the SE for ER-157P

> An applicant that requests a MUR with the upstream flow straightener configuration discussed in Section 3.2.2 [of Reference 11] should provide justification for claimed CheckPlus uncertainty that extends the justification provided in [Reference 19]. Since the [Reference 19] evaluation does not apply to the Check, a comparable evaluation must be accomplished if a Check is to be installed downstream of a tubular flow straightener.

The installed configuration of the WBN, Unit 2 LEFM flow meters does not include an upstream flow straightener. This criterion is therefore not applicable to the WBN, Unit 2 application.

Criterion 5 from the SE for ER-157P

> An applicant assuming large uncertainties in steam moisture content should have an engineering basis for the distribution of the uncertainties or, alternatively, should ensure that their calculations provide margin sufficient to cover the differences shown in Figure 1 of [Reference 20].

The licensee designed the internal moisture separation equipment to ensure that the moisture carryover does not exceed 0.25 percent by weight (wt%) for the WBN, Unit 2 Westinghouse Model D3-2 SGs. In the WCAP-18419-P (Enclosure 6 to the LAR), this value was used to calculate the power measurement sensitivity.

The licensee does not assume large uncertainties in steam moisture content. Therefore, no engineering basis for the distribution of the uncertainties in steam moisture is required and this criterion does not apply to the WBN, Unit 2 application.

Based on its review of the licensee's submittals as discussed above, the NRC staff finds that the licensee adequately addressed the guidance in Item D of Section I of Attachment 1 to RIS 2002-03 and meets the regulatory requirements of 10 CFR Part 50, Appendix K, and 10 CFR 50.36(c)(1)(ii)(A).

3.1.5.3 Item I.1.E, Attachment 1 to RIS 2002-03

Item E in Section I of Attachment 1 to RIS 2002-03 guides licensees to submit a plant-specific total power measurement uncertainty calculation, explicitly identifying all parameters and their individual contributions to the power uncertainty.

To address Item E of RIS 2002-03, the licensee provided Cameron Documents ER-734P, "Bounding Uncertainty Analysis for Thermal Power Determination at Watts Bar Unit 2 Using the LEFM CheckPlus System," and ER-732P, "Meter Factor Calculation and Accuracy Assessment for Tennessee Valley Authority, Watts Bar Nuclear Power Plant Unit 2" (Enclosure 5 to the LAR), and WCAP-18419-P, "Westinghouse Leading Edge Flow Meter (LEFM) Power Measurement Uncertainty for the Watts Bar Unit 2 MUR Program" (Enclosure 6 to the LAR). The licensee calculated the total thermal power uncertainty at WBN, Unit 2 due to the LEFM system with assumed bounding power uncertainty of 0.48 percent to be 0.6 percent.

As described in Section 3.1.4.1 of this SE, the NRC staff evaluated the methodology used to calculate the total thermal power uncertainty and reviewed the results of these calculations. This method statistically combines inputs to determine the overall uncertainty. Channel statistical allowances are calculated for the instrument channels. Dependent parameters are arithmetically combined to form statistically independent groups, which are then added to the SRSS for the remaining random parameters to determine the overall uncertainty. The vendor's determination of the uncertainty of the Cameron LEFM CheckPlus System is consistent with this methodology, as described in the referenced NRC-approved topical reports. In Section 3.1.4.2 of this SE, the NRC staff determined that the licensee properly identified all parameters associated with the thermal power measurement uncertainty, provided individual measurement uncertainties, and calculated the overall thermal power uncertainty.

Therefore, the NRC staff finds that the licensee adequately addressed the guidance in Item E of Section I of Attachment 1 to RIS 2002-03 and meets the regulatory requirements of 10 CFR Part 50, Appendix K, and 10 CFR 50.36(c)(1)(ii)(A).

3.1.5.4 Item I.1.F, Attachment 1 to RIS 2002-03

Item F in Section I of Attachment 1 to RIS 2002-03 guides licensees in providing information to address the specified aspects of the calibration and maintenance procedures related to all instruments that affect the plant thermal heat balance calculation.

The licensee addressed each of the five aspects of the calibration and maintenance procedures listed in Item F of Attachment 1 to RIS 2002-03, as follows:

Maintaining Calibration

LEFM hardware and instrumentation calibration and maintenance are performed using procedures that are based on Cameron LEFM CheckPlus System maintenance and calibration requirements, thus ensuring that the LEFM remains bounded by the Topical Report ER-80P analysis and assumptions. Calibration and maintenance for instrumentation that contributes to the thermal power heat balance computation other than the LEFM system, is performed periodically using existing site procedures.

Controlling Software and Hardware Configuration

The licensee stated that LEFM system software and digital assets are maintained using existing TVA procedures, which include verification and validation of changes to software configuration. Configuration of the hardware associated with the LEFM system and the instrumentation is maintained in accordance with TVA configuration control procedures.

The licensee also stated, “Instruments that affect the power calorimetric, including the Caldon LEFM CheckPlus System inputs, are monitored and maintained. Equipment issues for plant systems, including the Caldon LEFM CheckPlus System equipment, fall under site work control processes.”

Performing Corrective Actions

The licensee stated that problems with WBN, Unit 2 plant instrumentation are documented in the WBN, Unit 2 corrective action program and necessary corrective actions are identified and implemented. Deficiencies associated with the vendor’s processes or equipment are reported to Cameron to support corrective actions. Therefore, the corrective actions are monitored and performed in accordance with the TVA Corrective Action Program.

Reporting Deficiencies to the Manufacturer and Receiving and Addressing Manufacturer Deficiency Reports

The licensee stated that, regarding 10 CFR Part 21, “Reporting of Defects and Noncompliance,” the reporting of deficiencies to the manufacturer will be performed in accordance with the TVA procedures. The manufacturer deficiency reports also will be received and addressed in accordance with the TVA procedures.

After reviewing the above statements, the NRC staff finds that the licensee has addressed the calibration and maintenance aspects of the Cameron LEFM CheckPlus System and all other instruments affecting the power calorimetric. Therefore, the NRC staff finds that the licensee adequately addressed the guidance in Item F of Section I of Attachment 1 to RIS 2002-03 and meets the regulatory requirements of 10 CFR Part 50, Appendix K, and 10 CFR 50.36(c)(1)(ii)(A).

3.1.5.5 Items I.1.G and I.1.H, Attachment 1 to RIS 2002-03

Items I.1.G and I.1.H in Section I of Attachment 1 to RIS 2002-03 guide licensees to provide a proposed allowed outage time (AOT) for the instrument and to propose actions to reduce power if the AOT is exceeded. Items G and H refer to the Criterion 1 from ER-157P. The term AOT in the LAR is also referred to as Completion Time or CT.

Criterion 1 from the SE for ER-157P is addressed in Section 3.1.5.2 in this SE. The NRC staff found the proposed 72-hour CT for the LEFM being in MAINTENANCE mode prior to restoring it to NORMAL mode and the proposed 24-hour CT for a failed LEFM prior to reducing the power level to 3411 MWt acceptable.

Based on the above discussion and its review of the licensee's LAR and the Cameron engineering reports, the NRC staff finds that the licensee has provided sufficient justification for the proposed CT and the proposed power reduction actions if the CT is exceeded. Therefore, the NRC staff finds that the licensee adequately addressed the guidance in Items G and H of Section I of Attachment 1 to RIS 2002-03 and meets the regulatory requirements of 10 CFR Part 50, Appendix K, and 10 CFR 50.36(c)(1)(ii)(A).

3.1.6 <u>Conclusion</u>

The NRC staff's evaluation of the identified instrumentation for new power level for WBN, Unit 2 is based on the analytical limits documented by the licensee in the submitted application. Based on its review of the licensee's LAR, and Cameron engineering reports (including uncertainty calculations and referenced topical reports), the NRC staff considers that the licensee provided sufficient justification for the proposed TS changes. The licensee's proposed amendment is consistent with the NRC-approved Cameron Topical Report ER-80P and its supplement, Topical Report ER-157P. The licensee has followed the guidance in Items A through H in Section I of Attachment 1 to RIS 2002-03 and has, therefore, met the regulatory requirements and guidance in Section 2.4 of this SE.

Therefore, the NRC staff concludes that the instrumentation and controls aspect of the proposed MUR thermal power uprate of 1.4 percent RTP is acceptable.

3.2 <u>ACCIDENT ANALYSES</u>

The WBN, Unit 2 is a four-loop PWR with a NSSS furnished by Westinghouse Electric Corporation. The Unit 2 reactor core is rated at 3411 MWt and, at this core power, the NSSS will operate at 3427 MWt. The additional 16 MWt is due to the contribution of heat to the primary coolant system from nonreactor sources, primarily reactor coolant pump heat. The reactor core has an Engineered Safeguards design rating of 3582 MWt, and the NSSS has a design rating of 3596 MWt.

3.2.1 Regulatory Evaluation

The NRC staff performed this safety evaluation based on the following regulations:

GDC 10, “Reactor design,” which requires the reactor core and associated coolant, control, and protection systems to be designed with appropriate margin to assure that specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences (AOOs).

GDC 15, “Reactor coolant system design,” which requires the reactor coolant system and associated auxiliary, control, and protection systems to be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded during any condition of normal operation, including the effects of AOOs.

10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactor,” which requires, in part, the emergency core cooling systems to be designed with sufficient margin to assure that the design safety limits specified in 10 CFR 50.46(b) are met following loss-of-coolant accidents.

10 CFR 50.62, “Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants,” which provides the requirements for reduction of risk from ATWS events for light-water-cooled nuclear power plants.

3.2.2 Technical Evaluation

In Enclosure 2 to the LAR, TVA provided the WBN, Unit 2 specific information for each item outlined in RIS 2002-03, Attachment 1 to support the WBN, Unit 2 MUR power uprate application. Item II.1 of Enclosure 2 addresses the applicable analyses for transients and accidents included in Chapter 15 of the WBN UFSAR and other analyses that TVA is required to perform to support licensing of its plants. Specifically, Events 1 through 30 in Item II.1.D.iii of Enclosure 2 contain the information related to the UFSAR Chapter 15 transient and accident analyses.

In Section II.1.D.ii of Enclosure 2 to the LAR, TVA stated that for events that are departure from nucleate boiling (DNB) limited or when the Revised Thermal Design Procedure (RTDP) is used, the transient analyses assumed 3475 MWt as initial power level (representing the nominal MUR uprated power of 3459 MWt (101.4 percent of 3411 MWt) plus a reactor coolant pump (RCP) net heat input of 16 MWt). For events that are not DNB limited or for which the RTDP was not applied, the analyses were performed at initial conditions obtained by adding the bounding steady-state errors to nominal values in such a manner to maximize the impact on the limiting parameter.

The NRC staff identified some apparent inconsistencies in the stated analysis of record (AOR) power levels for some events when compared to the LAR and, therefore, issued a request for additional information (RAI) (Reference 21). The staff reviewed the responses to the RAI (Reference 2) and other available references. The staff determined that the discrepancies are most likely due to different interpretations of which power level is being referenced (nominal core power level, nominal NSSS power level, or some other bounding power level). However, the NRC staff determined that the information necessary to reconcile the inconsistencies was not necessary to make a safety finding. In all cases where two different power levels are provided in different documents, the lower power level is appropriate for the event in question.

In the unlikely event that the higher power level is correct due to an updated AOR, the results would be conservative. Therefore, the NRC staff finds that TVA’s selection of the initial power levels for the transient and accident analyses is consistent with the methods used for the Westinghouse design plants, which includes WBN, Unit 2, and finds that it is acceptable.

In Enclosure 2 to the LAR, TVA described each analysis briefly for the UFSAR Chapter 15 related events and assessed the AORs of the relevant events to determine the impact of the proposed MUR power uprate levels on the analyses. TVA stated that the AORs for UFSAR Chapter 15 events were performed at power levels equal to or greater than the MUR uprated power level and that acceptable results were obtained. Therefore, the AORs reflected in the WBN UFSAR were unaffected by the MUR power level and remained acceptable for the WBN, Unit 2 MUR power uprate to meet the requirements of GDC 10, GDC 15, 10 CFR 50.46, and 10 CFR 50.62.

Based on the licensee’s information of impact assessment of the UFSAR Chapter 15 events, the NRC staff classified these events into the following categories, with the exception of Event 20, “Waste Gas Decay Tank Rupture”; Event 23, “Major Rupture of a Main Steam Line”; Event 26, “Steam Generator Tube Rupture”; and Event 28, “Fuel Handling Accident,” which are dose release related analyses that are evaluated in SE 3.6.2.2.2. The containment performance analyses are evaluated in SE Section 3.2.2.6; the spent fuel pool (loss of cooling) accidents are evaluated in SE Section 3.2.2.7; and the spent fuel pool criticality event is evaluated in SE Section 3.2.2.8.

1. Category 1: Events that are not credible;
2. Category 2: Events that are not affected by the MUR power uprate;
3. Category 3: Events for which consequences were bounded by other Chapter 15 events;
4. Category 4: Events that were analyzed at a power level equal to the MUR uprated power level;
5. Category 5: Events that were analyzed at a power level that bounds the MUR uprated power level.

For the events in the five categories discussed below, TVA determined that the AORs remain valid for the relevant events at the proposed MUR uprate power level and provided rationales for each event supporting its determination for why an AOR is valid. The NRC staff discusses its evaluation of each event of Category 1 through 5 in Sections 3.2.2.1 to 3.2.2.5 below.

3.2.2.1 Category 1: Events that Are Not Credible

TVA indicates that the following event is not credible at WBN, Unit 2:

- UFSAR Section 15.2.6 - Startup of an Inactive Reactor Coolant Loop at an Incorrect Temperature

The reactor coolant system (RCS) at WBN, Unit 2 consists of four loops, with one RCP in each loop, connected parallel to the reactor vessel. The inactive reactor coolant loop startup event is defined as the startup assuming that one RCP in that loop was idle. The NRC staff finds that this mode of operation is prohibited by the WBN, Unit 2 TS 3.4.4, “RCS Loops - MODES 1 and 2,” which requires four RCS loops to be OPERABLE and in operation while in MODES 1 and 2. Therefore, the NRC staff concludes that this event is not credible at WBN, Unit 2.

3.2.2.2 Category 2: Events that Are Not Affected by the MUR Power Uprate

TVA indicates that the AORs of the following events are not affected by the MUR power uprate, since the analyses were performed at hot zero power (HZP) conditions.

3.2.2.2.1 UFSAR Section 15.2.1 - Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical Condition

UFSAR Section 15.2.1 documents the analysis of an uncontrolled addition of reactivity to the reactor core caused by withdrawal of rod cluster control assembly (RCCA) from a subcritical condition (Reference 22). This event could be caused by a malfunction of the reactor control or rod control systems. The AOR for this UFSAR Section 15.2.1 event was performed at HZP conditions and acceptable results were obtained. Therefore, the NRC staff concludes that the AOR remains valid for the WBN, Unit 2 MUR application because the MUR power uprate has no impact on the analyses performed at HZP conditions.

3.2.2.2.2 UFSAR Section 15.2.4 - Uncontrolled Boron Dilution during Modes 3 through 5

In a letter dated January 12, 2012, TVA supplied analyses of the boron dilution event, addressing all modes of plant operation, indicating the time that would be available for operator action following receipt of a qualified alarm for WBN, Unit 2 (Reference 23). The analyses of the event during Modes 3 through 5 were previously approved by the NRC (Reference 24). The AORs for the boron dilution events during Modes 3 through 5 at WBN, Unit 2 were performed at HZP conditions and acceptable results were obtained. Therefore, the NRC staff concludes that the AOR remains valid for the WBN, Unit 2 MUR application because the MUR power uprate has no impact on the analyses performed at HZP conditions. The NRC staff's evaluation of uncontrolled boron dilution while at power is presented in SE Section 3.2.2.4.3.

3.2.2.2.3 UFSAR Section 15.4.2.1 - Major Rupture of a Main Steam Line

UFSAR Section 15.4.2.1 contains the analysis of a complete severance of a main steam line break (SLB) with the plant initially at HZP conditions (Reference 22). The AOR for the SLB event was performed at HZP conditions and acceptable results were obtained. Therefore, the NRC staff concludes that the AOR remains valid for the WBN, Unit 2 MUR application because the MUR power uprate has no impact on the analyses performed at HZP conditions.

3.2.2.2.4 Conclusion for Category 2 Events

For the above Category 2 events, the NRC staff reviewed the information related to the methodologies, including computer codes, in WBN UFSAR Table 15.1-2 and applicable sections of WBN UFSAR Chapter 15 and found that the methodologies by which the AORs for the Category 2 events were performed were previously reviewed and approved by the NRC. The AORs were performed at HZP conditions and the results of the AORs meet the GDC 10 requirement, insofar as it relates to the SAFDLs, and the GDC 15 requirement, insofar as it relates to the RCPB pressure limits. Therefore, the NRC staff concludes that AORs for the Category 2 events are not affected by the MUR power uprate and remain valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.3 Category 3: Events for which Consequences Are Bounded by Other Chapter 15 Events

TVA indicates the AORs for the following events are not affected by the MUR power uprate, since the analyses are bounded by other Chapter 15 events.

3.2.2.3.1 UFSAR Section 15.2.9 - Coincident Loss of Onsite and External (Offsite) Alternating Current (AC) Power to the Station – Loss of Offsite Power to the Station Auxiliaries

UFSAR Section 15.2.9 describes the results of the analysis of a loss of offsite power (LOOP) event (Reference 22). A complete loss of all offsite power may result in the loss of all power to the plant auxiliaries. The loss of power may be caused by a complete loss of the offsite grid, accompanied by a turbine generator trip at the station, or by a loss of the onsite ac distribution system. Because the AOR of this LOOP event is bounded by AORs of UFSAR Section 15.2.7 event, “Loss of External Electrical Load and/or Turbine Trip”; UFSAR Section 15.2.8 event, “Loss of Normal Feedwater”; and UFSAR Section 15.3.4 event, “Complete Loss of Forced Reactor Coolant Flow,” the NRC staff finds that a separate analysis of the LOOP event is not needed for the WBN, Unit 2 MUR application.

3.2.2.3.2 UFSAR Section 15.3.2 - Minor Secondary System Pipe Breaks

UFSAR Section 15.3.2 discusses analyses of the minor secondary side pipe break (Reference 22). Because the results of this minor secondary side pipe break event are bounded by the AOR of the Section 15.4.2.1 event, “Major Rupture of a Main Steam Line,” the NRC staff finds that a separate analysis of this event is not needed for the WBN, Unit 2 MUR application.

3.2.2.4 Category 4: Events that Were Analyzed at a Power Level Equal to the MUR Uprated Power Level

TVA indicates that the AORs for the following 14 events remain valid, since the analyses were performed at a power level equal to the MUR uprated power level of 3459 MWt and acceptable results were obtained.

3.2.2.4.1 UFSAR Section 15.2.2 - Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at MUR Power Level of 3459 MWt

UFSAR Section 15.2.2 discusses the analysis of an uncontrolled RCCA bank withdrawal at power (Reference 22). This withdrawal will uncontrollably add positive reactivity to the reactor core, resulting in a power excursion. The AOR for this event was performed at a power level equal to the MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.4.2 UFSAR Section 15.2.3 - Rod Cluster Control Assembly Misalignment

UFSAR Section 15.2.3 discusses RCCA misalignment accidents including: (1) one or more dropped RCCAs within the same group; (2) a dropped RCCA bank; and (3) statically misaligned RCCA (Reference 22). The AOR for this event was performed at a power level equal to the

MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.4.3 UFSAR Section 15.2.4 - Uncontrolled Boron Dilution at Power

TVA supplied for WBN, Unit 2, in its letter dated January 12, 2012 (Reference 23), analyses of the boron dilution event, which addressed all modes of plant operation and indicated the time that would be available for operator action following receipt of a qualified alarm. The analyses of the event at power were previously approved by the NRC (References 24 and 25). The AOR for this boron dilution at power event was performed at a power level equal to the MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.4.4 UFSAR Section 15.2.5 - Partial Loss of Forced Reactor Coolant Flow

UFSAR Section 15.2.5 contains the analysis of a partial loss of reactor coolant flow involving the loss of one RCP with four loops in operation (Reference 22). The AOR for this partial loss of flow event was performed at a power level equal to the MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.4.5 UFSAR Section 15.2.7 - Loss of External Electrical Load and/or Turbine Trip

UFSAR Section 15.2.7 documents the analysis of a major load loss on the plant (loss of load), which can result from loss of external electrical load or from a turbine trip (Reference 22). The AOR for this loss of load event was performed at a power level equal to the MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.4.6 UFSAR Section 15.2.14 - Inadvertent Operation of Emergency Core Cooling System During Power Operation

UFSAR Section 15.2.14 provides TVA's analysis of an inadvertent operation of the ECCS. The inadvertent operation of the ECCS could be caused by operator error or by a spurious safety injection actuation signal (Reference 22). The AOR for this event was performed at a power level equal to the MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.4.7 UFSAR Section 15.2.15 - Chemical and Volume Control System Malfunction During Power Operation

UFSAR Section 15.2.15 provides TVA's analysis of an inadvertent operation of the chemical and volume control system (CVCS) (Reference 22). The inadvertent operation of the CVCS could be caused by operator error or by control signal malfunction. The AOR for this CVCS malfunction event was performed at a power level equal to the MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.4.8 Steam Line Break with Coincident Rod Withdrawal at Power

In a letter dated October 11, 2018, TVA provided to the NRC an analysis of a SLB coincident with rod withdrawal at power (Reference 26). The analysis was performed in response to an NRC request regarding an analysis of a SLB to confirm that the overpower delta temperature (OPΔT) reactor trip function provides adequate protection from unacceptable consequences during a steam line break event from hot full power conditions, including consideration of the thermal conductivity degradation (TCD) effects. The analysis was performed using the computer code PAD4TCD for evaluating TCD effects. The NRC allows TVA to use PAD4TCD for WBN, Unit 2 licensing applications until the WBN, Unit 2 SGs are replaced with SGs equivalent to the existing SGs at WBN, Unit 1 (Reference 27). The analysis determined that the peak fuel linear heat generation rate remained below a value that would cause fuel melting, and that the minimum departure from nuclear boiling ratio remained above the applicable safety analysis limit value when accounting for the effects of TCD. The AOR for this SLB coincident with rod withdrawal at power event was performed using the acceptable codes, including PAD4TCD, and an initial power level equal to the MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid and acceptable for the WBN, Unit 2 MUR uprated power level.

3.2.2.4.9 UFSAR Section 15.3.3 - Inadvertent Loading of a Fuel Assembly into an Improper Position

UFSAR Section 15.3.3 describes fuel and core loading errors that may occur from the inadvertent loading of one or more fuel assemblies into improper positions, loading a fuel rod during manufacture with one or more pellets of the wrong enrichment, or the loading of a full fuel assembly during manufacture with pellets of the wrong enrichment (Reference 22). The fuel assembly misloading event will lead to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. Also included among possible core loading errors is the inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods. The AOR for this fuel assembly misloading event was performed at a power level equal to the MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.4.10 UFSAR Section 15.3.4 - Complete Loss of Forced Reactor Coolant Flow

UFSAR Section 15.3.4 documents the analysis of a complete loss of forced reactor coolant flow that may result from a simultaneous loss of electrical supplies to the RCPs (Reference 22). The AOR for this complete loss of flow event was performed at a power level equal to the MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.4.11 UFSAR Section 15.3.6 - Single Rod Cluster Control Assembly Withdrawal at Full Power

UFSAR Section 15.3.6 presents the analysis of the case of the worst rod withdrawn from bank D inserted at the insertion limit, with the reactor initially at full power (Reference 22). This event is assumed to occur at beginning-of-life because this results in the minimum value of moderator temperature coefficient, which, in turn, maximizes the power rise and minimizes the tendency of increased moderator temperature to flatten the power distribution. The AOR for this single RCCA withdrawal from full power event was performed at a power level equal to the MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.4.12 UFSAR Section 15.4.2.2 - Major Rupture of a Main Feedwater Pipe

UFSAR Section 15.4.2.2 describes the results of TVA's analysis of FW system pipe breaks (FW line break) inside and outside containment (Reference 22). The AOR of the FW line break assumed that the flow from the break is mostly water. The effect of the break flow of water upon the RCS is an undercooling of the core. The RCS temperature and pressure increase and reach elevated levels until all the decay heat can be removed by the auxiliary feedwater system. The AOR for this event was performed at a power level equal to the MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.4.13 UFSAR 15.4.3 - Steam Generator Tube Rupture

UFSAR Section 15.4.3 discusses the analysis of the complete severance of a single SG tube rupture (SGTR) (Reference 22). The SGTR event leads to an increase in contamination of the secondary system due to leakage of radioactive coolant from the RCS. In the event of a coincident loss of offsite power, or failure of the condenser dump system, discharge of radioactivity to the atmosphere takes place via the SG power-operated relief valves (and safety valves if their setpoint is reached). TVA evaluated the impacts of the MUR power uprate on the SGTR analyses (margin to overfill and input to dose calculation) to determine if they remain bounding for the MUR power uprate. The AOR for this SGTR event was performed at a power level equal to the MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.4.14 UFSAR Section 15.4.4 - Single Reactor Coolant Pump Locked Rotor

UFSAR Section 15.4.4 presents the analysis of an instantaneous seizure of an RCP rotor (locked rotor) (Reference 22). Flow through the affected reactor coolant loop is rapidly reduced, leading to initiation of a reactor trip on a low flow signal. The AOR for this locked rotor event was performed at a power level equal to the MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.4.15 Conclusion for Category 4 Events

For the above Category 4 events, the NRC staff reviewed the information related to the methodologies, including computer codes, in WBN UFSAR Table 15.1-2 and in applicable sections of WBN UFSAR Chapter 15, and found that the methodologies by which the AORs for the Category 4 events were performed were previously reviewed and approved by the NRC. The AORs were performed at a power level equal to the MUR uprated power level of 3459 MWt and the results of the related AORs meet the GDC 10 requirement, insofar as it relates to the SAFDLs, and the GDC 15 requirement, insofar as it relates to the RCPB pressure limits. Therefore, the NRC staff concludes that the AORs for the Category 4 events remain valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.5 Category 5: Events that Were Analyzed at a Power Level that Bounds the MUR Uprated Power Level

TVA indicates that AORs for the following nine events remain valid, since the analyses were performed at a power level that bounds the MUR uprated power level of 3459 MWt and acceptable results were obtained:

3.2.2.5.1 UFSAR Section 15.2.8 - Loss of Normal Feedwater

UFSAR Section 15.2.8 presents the results of the analysis of a loss of normal feedwater (Reference 22). The AOR for this event was performed at a power level greater than the MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.5.2 UFSAR Section 15.2.10 - Excessive Heat Removal Due to Feedwater System Malfunctions

UFSAR Section 15.2.10 describes the results of TVA's analysis of excessive heat removal caused by feedwater system malfunctions (Reference 22). A decrease in feedwater temperature and an increase in feedwater flow are two examples of feedwater system malfunctions that can cause an increase in heat removal rate. The AOR for this excessive heat removal event was performed at a power level greater than the MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.5.3 UFSAR Section 15.2.11 - Excessive Load Increase Incident

UFSAR Section 15.2.11 discusses the analysis of excessive load increase event (Reference 22). The AOR for this excessive load increase event was performed at a power level greater than the MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.5.4 UFSAR Section 15.2.12 - Accidental Depressurization of the Reactor Coolant System

UFSAR Section 15.2.12 documents the analysis of an accidental depressurization of the RCS due to an inadvertent opening of a pressurizer safety valve (Reference 22). The AOR for this RCS depressurization event was performed at a power level greater than the MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.5.5 UFSAR Section 15.2.13 - Accidental Depressurization of the Main Steam System

UFSAR Section 15.2.12 documents the analysis of an accidental depressurization of the main steam system associated with an inadvertent opening of a single steam dump, relief, or safety valve (Reference 22). The AOR for this depressurization of the main steam system event was performed at a power level greater than the MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.5.6 UFSAR Section 15.4.6 - Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

UFSAR Section 15.4.6 includes the analysis of the mechanical failure of a control rod mechanism pressure housing resulting in the ejection of an RCCA and drive shaft (rod ejection) (Reference 22). The consequence of this rod ejection event is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. Regarding an updated analysis of the hot spot rod ejection event, TVA previously submitted to the NRC the results of a reanalysis of the rod ejection cases for end-of-life conditions, assuming both the hot full power and hot zero power cases, including consideration of the TCD effects (Reference 26). The analysis was performed using the computer code PAD4TCD for evaluating TCD effects. The NRC allows TVA to use PAD4TCD for WBN, Unit 2 licensing applications until the Watts Bar Unit 2 steam generators are replaced with SGs equivalent to the existing SGs at Watts Bar Unit 1 (Reference 27). The analysis determined that the licensing basis acceptance criteria continue to be satisfied when accounting for the effects of TCD. Therefore, the effects of fuel TCD can be accommodated for the rod ejection event for WBN, Unit 2. The AOR for this rod ejection event was performed using acceptable codes, including PAD4TCD, and an initial power level at a power level greater than the MUR uprated power level of 3459 MWt, and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.5.7 UFSAR Section 15.3.1 - Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes Which Actuate the Emergency Core Cooling System

UFSAR Section 15.3.1 documents a rupture of RCS piping (Reference 22). A break sizes scoping study identified that the four-inch break was the limiting break size for WBN, Unit 2 in terms of the highest peak cladding temperature. The small break LOCA AOR and the subsequent evaluations have explicitly analyzed a core power of 3459 MWt plus a 0.6 percent power uncertainty for a total core power of 3480 MWt in accordance with Appendix K to 10 CFR Part 50, and acceptable results were obtained. Therefore, the NRC staff finds that the small break LOCA AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.5.8 UFSAR Section 15.4.1 Major Reactor Coolant System Pipe Ruptures (Loss of Coolant Accident)

UFSAR Section 15.4.1 documents the analysis for large break LOCA, which is a double-ended, guillotine rupture of the cold leg (Reference 22). The best estimate large break LOCA analysis, which is documented in WCAP-17093-P, Revision 1, performed for WBN, Unit 2 uses the NRC-approved method, the Automated Statistical Treatment of Uncertainty Method (ASTRUM) (Reference 28). The best estimate large break LOCA analysis was performed at an assumed core power of 3479.8 MWt, which was specifically done to bound operation considering MUR power uprate conditions. The NRC-approved results in UFSAR Table 15.4-18b show that the analysis meets the 10 CFR 50.46 requirements in terms of the limits of the peak cladding temperature, maximum local oxidation, and maximum hydrogen generation (Reference 22).

The post-LOCA long-term core cooling AOR addresses three concerns relative to managing the post-LOCA recovery: subcriticality, boric acid precipitation control (i.e., hot-leg switchover), and decay heat removal. The NRC-approved post-LOCA long-term core cooling AOR was performed at a power level of 3479 MWt (102 percent of 3411 MWt).

The AOR for the large break LOCA and its associated long-term core cooling AOR were performed at a power level greater than the MUR uprated power level of 3459 MWt and acceptable results were obtained. Therefore, the NRC staff finds that the AOR remains valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.5.9 Anticipated Transients Without Scram

The ATWS events are not considered as design-basis events, and the ATWS analysis is not included in WBN UFSAR Chapter 15. The ATWS rule (i.e., 10 CFR 50.62) requires that Westinghouse-designed plants install ATWS mitigation system circuitry (AMSAC) to initiate a turbine trip and actuate auxiliary feedwater flow independent of the reactor protection system. The generic ATWS analyses documented in NS-TMA-2182 (Reference 29) support the analytical basis for the NRC-approved generic AMSAC designs. The results of the generic analyses applicable to WBN, Unit 2 indicate that the peak RCS pressure predicted for ATWS events is 2780 pounds per square inch absolute (psia), which is below the ASME Service Level C limit of the weakest component in the RCS (3200 pounds per square inch gauge (psig)). The sensitivity analysis in the generic analysis also determined that the peak RCS pressure with the 2 percent increase in power remains below 3200 psig. The proposed increase in power of 1.4 percent for WBN, Unit 2 is within the applicable range of the 2 percent increase in power assumed in the sensitivity analysis. In addition, the NRC previously approved the AMSAC design including the analytical basis at WBN, Unit 2 for meeting the requirements of 10 CFR 50.62(c) (Reference 24). Therefore, the NRC staff finds that operation of WBN, Unit 2 at the MUR uprated power level remains within the bounds of the generic Westinghouse ATWS analysis documented in NS-TMA-2182 and will remain in compliance with 10 CFR 50.62(c).

3.2.2.5.10 Conclusion for Category 5 Events

For the above Category 5 events, the NRC staff reviewed the information related to the methodologies, including computer codes, in WBN UFSAR Table 15.1-2 and applicable sections of WBN UFSAR Chapter 15 (Reference 22) and finds that the methodologies by which the AORs for the Category 5 events were performed were previously reviewed and approved by the NRC. The AORs were performed at a power level that is greater than the MUR uprated power level and the results of the related AORs meet: (1) the GDC 10 requirement, insofar as it relates to the SAFDLs; (2) the GDC 15 requirement, insofar as it relates to the RCPB pressure limits; (3) 10 CFR 50.46 requirements, insofar as they relate to the performance criteria of the emergency core coolant system; and (4) 10 CFR 50.62(c) requirements, insofar as they relate to requirements for reduction of risk from ATWS events. Therefore, the NRC staff concludes that the AORs for the Category 5 events remain valid for the WBN, Unit 2 MUR uprated power level.

3.2.2.6 Containment Performance Analyses

The containment for WBN, Unit 2 consists of a free-standing steel vessel with an ice condenser and separate reinforced concrete Shield Building. The free-standing steel vessel and the concrete shield building were designed by TVA and the ice condenser was designed and furnished by the Westinghouse Electric Corporation.

3.2.2.6.1 Regulatory Evaluation

The NRC staff performed this safety evaluation based on the following regulations and guidance:

GDC 16, “Containment design,” which requires the containment and associated systems to be designed to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

GDC 38, “Containment heat removal,” which requires a containment heat removal system to be provided with the safety function to rapidly reduce the containment pressure and temperature following a LOCA and maintain them at acceptably low levels.

GDC 50, “Containment design basis,” which requires the containment and its associated heat removal systems to be designed so that the containment structure can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

NUREG-0800 (referred to as the Standard Review Plan (SRP), Sections 6.2.1, “Containment Functional Design”; Section 6.2.1.1.B, “Ice Condenser Containments”; Section 6.2.1.2, “Subcompartment Analysis”; Section 6.2.1.3, “Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents (LOCAs)”; and Section 6.2.1.4, “Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures,” contain guidance for the NRC staff’s evaluation of the functional capability of the reactor containment, including its ability to withstand, without loss of function, the pressure and temperature conditions resulting from postulated loss-of-coolant, steam line, or feedwater line break accidents (References 30, 31, 32, and 33, respectively).

3.2.2.6.2 Technical Evaluation

The NRC staff reviewed the following areas of containment design and analysis as described in the LAR Enclosure 2, Section II.1.D.iii.31: long-term LOCA containment integrity analysis; short-term LOCA mass and energy release analysis; loop subcompartment analysis; reactor cavity analysis; pressurizer enclosure analysis; maximum reverse pressure differential analysis; and SLB mass and energy release evaluation (long-term SLB mass and energy release inside containment or outside of containment; short-term SLB mass and energy release inside containment or outside of containment). The above review areas were developed based on the RIS 2002-03 guidance (Reference 5).

3.2.2.6.3 Long-Term LOCA Containment Integrity Analysis

The purpose of this analysis is to demonstrate the ability of the containment safeguard systems to mitigate the consequences of a large break LOCA. WBN UFSAR Section 6.2.1.3 has a discussion of the long-term containment analysis (Reference 34). The methodology for the licensing basis analysis as stated in the LAR is contained in WCAP-17721-P-A (Reference 35). Based on this methodology, the AOR presently assumes a core thermal power of 3479 MWt, which is the same as the proposed MUR uprated core thermal power of 3459 MWt plus the calorimetric uncertainty of 0.6 percent (20 MWt).

Because the long-term LOCA containment integrity analysis was performed using the NRC-approved methodology with power level initiated from the same power as the proposed MUR uprated power level plus the required calorimetric uncertainty (0.6 percent), the long-term LOCA mass and energy release are not impacted by the MUR power uprate and will not change due to the MUR power uprate. The MUR power uprate will have no impact on the long-term post LOCA containment response and containment integrity analysis for the current licensing basis analysis and the conclusions discussed in UFSAR Section 6.2.1.3.

Based on the above, the NRC staff finds that the licensee's evaluation conforms with SRP 6.2.1.3 and will continue to meet the GDC 16, GDC 38, and GDC 50 requirements and, therefore, is acceptable.

3.2.2.6.4 Short-Term LOCA Mass and Energy Release Analysis

The purpose of this analysis is to provide the data for the containment and subcompartment functional design. WBN UFSAR Section 6.2.1.3.4 discusses the short-term LOCA mass and energy analysis. WBN UFSAR Tables 6.2.1-23, 6.2.1-24, 6.2.1-28, and 6.2.1-30 present the LOCA mass and energy release data that were used to perform the containment and subcompartment analyses.

In a response to RAI SNSB-Containment-1 (Reference 2), the licensee stated that the mass and energy release is unique to each pipe break and location for each short-term subcompartment analysis. The power level assumed for the generation of the LOCA mass and energy release used for the compartment response for the current design basis analyses are provided in Table RAI SNSB-Cont-1, which is reproduced below.

Table RAI SNSB-Cont-1 – Analyzed Power Level (includes uncertainty)

Analysis	**Current Analysis of Record**
Loop Compartment	102% of 3411 MWt = 3479 MWt
Reactor Cavity	102% of 3411 MWt = 3479 MWt
Pressurizer Enclosure	102% of 3411 MWt = 3479 MWt
Maximum Reverse Pressure Differential	102% of 3570 MWt = 3641 MWt

The WBN, Unit 2 reactor core power is currently rated at 3411 MWt. The proposed MUR uprated core power of 3459 MWt is equal to a power uprate of 1.4 percent from current rated core power of 3411 MWt (i.e., 3459 = 3411 x 101.4). By taking the calorimetric uncertainty of 0.6 percent (i.e., 3411 x 0.006 = 20 MWt) into the core power used for the MUR compartment pressurization analysis due to line break (i.e., 3479 = 3459 + 20), the existing AOR analysis results appear to remain valid for the MUR uprated power level.

However, the short-term blowdown transients as evaluated below (i.e., Loop Compartment Line Break, Reactor Cavity Line Break, and Pressurizer Enclosure Spray Line Break), last only for approximately three seconds (see UFSAR Tables 6.2.1-23, 6.2.1-24, 6.2.1-28, and 6.2.1-30). The analysis results for these kinds of short-term transients will be more affected by the initial RCS fluid temperatures than the reactor core power.

The short-term blowdown transients are characterized by a peak mass and energy release rate that occurs during a subcooled condition. The Zaloudek correlation is part of the NRC-approved methodology (Reference 36) for calculating the short-term critical mass flux from the break.

This correlation shows that the use of lower RCS water temperatures will make the critical mass flux higher.

The RCS temperatures used for the following evaluation are provided in Table RAI SNSB-Cont-2, which is reproduced below.

Table RAI SNSB-Cont-2		
Analysis	**Current Analysis of Record Conditions**	**Conditions Proposed for 1.4% MUR**
Loop Compartment Breaks		
T_{hot}	617.1°F	619.1°F
T_{cold}	552.2°F	557.3°F
Reactor Cavity Breaks and Pressurizer Spray Line Break		
T_{hot}	624°F	619.1°F
T_{cold}	559°F	557.3°F

3.2.2.6.5 Loop Subcompartment Analysis

The analysis, described in UFSAR Section 6.2.1.3.4, is performed to ensure that the walls of the loop subcompartments, including the lower crane wall, upper crane wall, operating deck, and the containment shell, can maintain their structural integrity during the short pressure pulse (generally, less than three seconds) that accompanies a LOCA. Additionally, this analysis verifies the adequacy of the ice condenser performance.

By the Zaloudek correlation as discussed in Section 3.2.2.6.4 of this SE and the RCS water temperature shown in Table RAI SNSB-Cont-2, the critical mass flux from the loop compartment break for the current AOR will bound that for the 1.4 percent MUR condition. Therefore, the NRC staff finds that the current WBN, Unit 2 licensing basis mass and energy release from compartment break and associated loop subcompartment analyses remain bounding for the 1.4 percent MUR power uprate.

3.2.2.6.6 Reactor Cavity Analysis

The purpose of this analysis is to ensure that the walls in immediate proximity of the reactor vessel can maintain their structural integrity during the short pressure pulse that accompanies a LOCA within the reactor cavity region.

UFSAR Section 6.2.1.3.9 discusses the reactor cavity analysis that was performed using the NRC-approved transient mass distribution (TMD) computer code (References 37 and 38). For the short-term LOCA in the reactor cavity region, its pressurization can be primarily determined by the critical mass flux as calculated by the Zaloudek correlation and the break size. Based on the RCS water temperature shown in Table RAI SNSB-Cont-2, the critical mass flux from the RCS line break inside the reactor cavity region will be higher by 3.6 percent for the 1.4 percent MUR condition than that for the AOR condition. The NRC staff confirmed a similar amount of increase (i.e., approximately 4 percent) based on the calculation using steam table and speed of sound in subcooled water. The licensee stated that the much smaller as-built potential break size compared to the break size used in AOR analysis can offset the increase in mass flux.

The current licensing basis reactor vessel inlet break size (i.e., AOR analysis size) is 127 in^2. The licensee stated that the as-built potential break size of 45 in^2 (Reference 39), which was confirmed by the NRC staff during audit (Reference 40), would reduce the actual mass and energy release rate by a factor of 2.6. The licensee further stated that this reduction in break size would more than offset the 3.6 percent increase in the critical mass flux and still maintain enough analysis margin for the peak differential pressure across the structures.

Because the reactor cavity analysis was performed using the NRC-approved TMD computer code and methodology with the analysis break size bounding the actual break size by an amount sufficient to offset the increase in the critical mass flux due to the proposed MUR uprated power level, the NRC staff finds that the licensee's evaluation conforms with SRP 6.2.1.3 for mass and energy release applied to reactor cavity analysis and will continue to meet the GDC 16, GDC 38, and GDC 50 requirements; therefore, the change is acceptable.

3.2.2.6.7 Pressurizer Enclosure Analysis

The purpose of this analysis is to ensure that the walls in the immediate proximity of the pressurizer enclosure can maintain their structural integrity.

UFSAR Section 6.2.1.3.9 discusses the pressurizer enclosure analysis that was performed using the NRC-approved TMD computer code (References 37 and 38). For the short-term LOCA in the pressurizer enclosure region, the pressurization can be primarily determined by the critical mass flux as calculated by the Zaloudek correlation and the break size. Based on the RCS water temperature shown in Table RAI SNSB-Cont-2, the critical mass flux from the RCS line break inside the pressurizer enclosure region will be higher by 3.6 percent for the 1.4 percent MUR power uprate condition than that for the AOR condition. The NRC staff confirmed a similar amount of increase (i.e., approximately 4 percent) based on the calculation using steam table and speed of sound in subcooled water. The licensee stated that the much smaller as-built potential break size compared to the break size used in AOR analysis can offset the increase in mass flux.

The current licensing basis for pressurizer spray line break is a double-ended rupture of the 6-inch (an approximate area of 0.1963 ft^2) spray line. The rupture is assumed to occur at the top of the enclosure. However, the licensee stated that the as-built potential break will be around either the cold leg spray nozzle or the pressurizer spray nozzle with an area of 0.0645 ft^2 or 0.08727 ft^2, respectively. In its review of the response to RAI SNSB-Containment-3, the NRC staff confirmed that the stated as-built break area will be 0.0645 ft^2 or 0.08727 ft^2 for potential breaks in the pressurized enclosure area. This reduction in break sizes from the analyzed break size area of 0.1963 ft^2 to the as-built size areas of 0.0645 ft^2 or 0.08727 ft^2 would more than offset the 3.6 percent increase in the critical mass flux and still maintain enough analysis margin for the peak differential pressure across the structures.

Because the pressurizer enclosure analysis was performed using the NRC-approved TMD computer code and methodology with the analysis break size bounding the actual break size by an amount sufficient to offset the increase in the critical mass flux due to the MUR uprated power level, the NRC staff finds that the licensee's evaluation conforms with SRP 6.2.1.3 for pressurizer enclosure related analysis and will continue to meet the GDC 16, GDC 38, and GDC 50 requirements and, therefore, is acceptable.

3.2.2.6.8 Maximum Reverse Pressure Differential Analysis

Following a LOCA, the pressure and temperature in the lower compartment of containment increases, which forces the air in the lower compartment into the upper compartment and increases the pressure in the upper compartment. As the temperature in the lower compartment decreases with time, the pressure in the lower compartment also decreases. Eventually the pressure in the lower compartment becomes less than the pressure in the upper compartment, which creates a reverse differential pressure across the operating deck. This analysis is used to predict this reverse differential pressure and to ensure the structural adequacy of the operating deck.

UFSAR Section 6.2.1.3.11 discusses the maximum reverse pressure differential analysis that was performed using the NRC-approved computer codes TMD and LOTIC (References 37 and 38).

UFSAR Table 6.2.1-37 shows that the calculated maximum reverse pressure differential across the operating deck was 0.65 psi. The licensee stated that the design reverse pressure differential across the operating deck is 6.8 psi. During the audit, the NRC staff confirmed that the analysis of record, WCAP-18398-P, Section 6.4.3.3, documented the design reverse differential pressure of 6.8 psi and calculated a differential pressure 0.65 psi (Reference 40).

The AOR is a generic and conservative analysis discussed in UFSAR Section 6.2.1.3.11. The dead-ended compartments adjacent to the lower compartment are assumed to be swept of air during the initial blowdown. This is a conservative assumption because this will maximize the air forced into the upper ice bed and upper compartment thus raising the compression pressure for the operating deck. In addition, it will minimize the non-condensables in the lower compartment.

The mass and energy release utilized serves only as a vehicle to initiate the event and to purge the lower and the dead-ended compartment air. Any increases in release during the post-blowdown period would result in the lower compartment pressure remaining at a higher value, and thus would reduce the reverse differential pressure. The mass and energy release is extracted from a model used to maximize the LOCA peak cladding temperature and not from a model used to maximize the peak containment pressure. It is judged that the RCS temperature changes and the resulting effects would not affect the results of the maximum reverse pressure differential calculation.

The AOR shows that significant margin exists between a calculated reverse differential pressure of 0.65 psi and the design reverse differential pressure value of 6.8 psi across the operating deck. Thus, the 1.4 percent MUR power uprate will have a minimal impact, if any, on the analysis and there is significant analysis margin available. Therefore, the NRC staff finds that the current AOR remains bounding for the 1.4 percent MUR power uprate.

The NRC staff finds that the licensee's evaluation is acceptable based on the following:

1. There exists significant margin between a calculated reverse differential pressure of 0.65 psi and the design reverse differential pressure value of 6.8 psi across the operating deck.

2. For the current AOR (i.e., prior to the MUR power uprate), the calculated maximum reverse pressure differential of 0.65 psi was obtained by using the NRC-approved computer codes TMD and LOTIC and methodology.

The NRC staff concludes that the licensee's evaluation conforms with SRP 6.2.1.3 for mass and energy release as applied to the maximum reverse pressure differential analysis and will continue to meet the GDC 16, GDC 38, and GDC 50 requirements; therefore, it is acceptable.

3.2.2.6.9 SLB Mass and Energy Release Evaluation

Following a steam line break in the lower compartment of an ice condenser plant, the steam released from the break would be superheated and have potential to make the lower compartment temperature become limiting. The MUR power uprate will increase the operating core power, which, in turn, will increase the mass and energy release from the break. Hence, the impact of the MUR power uprate on the containment integrity due to SLB will be evaluated in this section through the evaluation of its impact on the mass and energy release from the SLB.

UFSAR Section 6.2.1.3.10 discusses the streamline break analysis that was performed using the NRC-approved computer codes TMD and LOTIC-3 (References 38 and 41).

The TMD computer code analysis was performed with modeling of the containment into many nodes so that the non-uniformity of pressure and mass distribution can be properly obtained. However, this short-term period will only involve a few seconds of transient time. The licensee has identified the critical parameters for the short-term SLB mass and energy release as: the no-load initial power that maximizes the secondary system pressure, the flow area of the steam piping at the break, and the total mass of steam in the piping of the unfaulted lines and the header. For the proposed MUR power uprate, the no-load reactor coolant system temperature of 557 degrees Fahrenheit (°F) is unchanged from the current AOR. The short-term mass and energy release is primarily determined by the reactor coolant system water temperature.

Because the short-term SLB mass and energy was calculated using the NRC-approved computer code and methodology initiated from the unchanged reactor coolant system temperature for the 1.4 percent MUR power uprate, the NRC staff finds that the licensee's evaluation meets the GDC 16, GDC 38, and GDC 50 requirements and, therefore, it is acceptable.

The LOTIC-3 computer code analysis was performed with modeling of the containment into fewer nodes but for much longer transient time than the TMD analysis. The licensee has identified the critical parameters for the long-term SLB mass and energy release as: the NSSS power level, reactivity feedback characteristics including the minimum plant shutdown margin, the initial and trip values for the SG water mass, MFW flow, auxiliary feedwater (AFW) flow, main and AFW enthalpies, and the times at which steam line and FW line isolation occur. UFSAR Table 6.2.1-39 presents the calculated mass and energy release data for 30 percent and 100.6 percent power cases. Table 6.2.1-13 listed the 100 percent NSSS power of 3475 MWt for the WBN, Unit 2 analysis. The licensee stated in the LAR that the 1.4 percent MUR power uprate is based on current licensed NSSS power of 3427 MWt.

Because the long-term SLB mass and energy was calculated using the NRC-approved LOTIC-3 computer code and methodology initiated from the NSSS power (3475 MWt) that bounds the NSSS power for the proposed 1.4 percent MUR power uprate, the NRC staff finds that the licensee's evaluation conforms with SRP 6.2.1.4 for SLB mass and energy and will continue to meet the GDC 16, GDC 38, and GDC 50 requirements and, therefore, it is acceptable.

3.2.2.6.10 Conclusion for Containment Performance Analyses

The NRC staff has reviewed the licensee's assessment of the topics addressed in this SE and concludes that they are adequately addressed for the impact of the 1.4 percent MUR power uprate on the containment performance analyses as discussed in the above sections. Therefore, the NRC staff concludes that WBN, Unit 2 will continue to meet the requirements of GDC 16, GDC 38, and GDC 50 following the implementation of the proposed 1.4 percent MUR power uprate.

3.2.2.7 Spent Fuel Pool Accidents (Loss of Pool Cooling)

The licensee addressed the effects of the MUR power uprate on a loss of spent fuel pool cooling in Section II.1.D.ii.35 of Enclosure 2 to the LAR.

3.2.2.7.1 System Description

The spent fuel pool cooling and cleanup system is designed to remove decay heat generated by stored spent fuel assemblies from the spent fuel pool water. The system is described in the WBN UFSAR Section 9.1.3 (Reference 42). Postulated unavailability of the system could lead to a loss of pool cooling and subsequent reduction in coolant inventory.

3.2.2.7.2 Regulatory Evaluation

The NRC staff performed this safety evaluation based on the following regulation:

GDC 61, "Fuel storage and handling and radioactivity control," which requires, in part,[5] the fuel storage and handling systems to be designed to assure adequate safety under normal and postulated accident conditions, including removing residual heat and preventing a significant reduction in fuel storage coolant inventory under accident conditions.

3.2.2.7.3 Summary of Information Provided by the Licensee

The licensee addressed the effects of the MUR power uprate on a loss of spent fuel pool cooling in Section II.1.D.ii.35 of Enclosure 2 to the LAR. The licensee noted the following:

- The spent fuel heat load is evaluated on a cycle-specific basis prior to a full core offload to ensure that the heat load remains less than the maximum allowable heat load.
- In the event of a loss of forced cooling, the large volume of water in the spent fuel pool would take several hours to heat to boiling.
- In addition to the spent fuel pool cooling and cleanup system, several other sources of makeup water are available in the event of a loss of cooling.

[5] The remaining requirements in GDC 61 are addressed by the auxiliary systems analysis provided in Section VI.1.D of the LAR and evaluated in the corresponding section of this SE.

3.2.2.7.4 Technical Evaluation

In terms of a postulated loss of spent fuel pool cooling, the implementation of the MUR would result in a minor increase in the decay heat associated with fuel recently removed from the core, such as in a full core offload. Because the increase in the decay heat load is minor, and the spent fuel heat load is reevaluated prior to a full core offload, the NRC staff determined that the licensee has acceptably addressed the effects of the MUR on spent fuel pool accidents. The NRC staff also determined that the remaining safety considerations—the large volume of water in the pool and the availability of multiple sources of makeup water—would be unaffected by the small increase in decay heat associated with the MUR.

3.2.2.7.5 Conclusion

Based on the considerations discussed above, the NRC staff determined that the licensee has demonstrated that, upon implementation of the MUR power uprate, the fuel storage and handling systems will continue to be able to remove residual heat, and that multiple backup systems continue to assure that a significant reduction in fuel storage coolant inventory under accident conditions would be prevented, consistent with GDC 61 requirements. Therefore, the NRC staff concludes that the proposed MUR power uprate is acceptable with respect to spent fuel pool accidents.

3.2.2.8 Spent Fuel Pool Criticality

The licensee addressed the effects of the MUR power uprate on spent fuel pool criticality in Section II.1.D.ii.36 of Enclosure 2 to the LAR.

3.2.2.8.1 System Description

The spent fuel pool is a reinforced, concrete structure that includes fuel assembly storage racks. It is described in Section 9.1, “Fuel Storage and Handling,” of the WBN UFSAR, and its criticality analysis is described in Section 4.3.2.7, “Criticality of Fuel Assemblies,” of the UFSAR (Reference 43).

3.2.2.8.2 Regulatory Evaluation

The NRC staff performed this safety evaluation based on the following regulations: 10 CFR 50.68, “Criticality accident requirements,” and GDC 62, “Prevention of criticality in fuel storage and handling.” In part, 10 CFR 50.68 requires, for licensees that do not credit soluble boron in the criticality analyses, that the k-effective (k_{eff}) of the spent fuel pool storage racks must not exceed 0.95 at a 95-percent probability, 95-percent confidence level. Additionally, GDC 62 states, “Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.”

3.2.2.8.3 Summary of Information Provided by the Licensee

The licensee addressed the effects of the MUR power uprate on spent fuel pool criticality in Section II.1.D.ii.36 of Enclosure 2 to the LAR. The licensee stated that the existing analysis, which was approved by the NRC staff in a recent license amendment, considers fresh fuel with a bounding, uniform enrichment and high density along the entire length of the fuel pin

(Reference 44). The licensee also stated that the analysis is not power-level dependent and remains bounding of MUR conditions.

3.2.2.8.4 Technical Evaluation

The implementation of an MUR will require a small increase in fissile material in the core to sustain the added energy requirements of fuel cycles operating at the higher power level. This change would, in turn, affect the discharge characteristics, such as reactivity, of the spent fuel. In its safety evaluation approving the implementation of core designs containing tritium producing burnable absorber rods for WBN, Unit 2, the NRC staff reviewed the spent fuel pool criticality analysis, which, as noted by the licensee, assumed a bounding, high enrichment and high fuel density. The NRC staff noted that this bounding approach provided analysis simplicity and margin. In its present review, the NRC staff determined that the analyses remain bounding and applicable, because the high enrichment corresponds to WBN, Unit 2 TS Limiting Condition for Operation 3.7.15, “Spent Fuel Pool Assembly Storage.” This limit, 5.0 weight percent, will not change for the MUR.

3.2.2.8.5 Conclusion

Based on the considerations discussed above, the NRC staff determined that the MUR power uprate will not affect the applicability of the spent fuel pool criticality analyses, which will continue to demonstrate that the k_{eff} of the spent fuel pool storage racks remains below the regulatory limit in 10 CFR 50.68, assuring the requisite prevention of criticality per GDC 62. Therefore, the NRC staff concludes that the proposed MUR power uprate is acceptable with respect to spent fuel pool criticality.

3.2.3 Technical Conclusion

The NRC staff has reviewed the licensee’s assessment of the topics addressed in Section 3.2 of this SE and concludes that they are adequately addressed for the impact of the 1.4 percent MUR power uprate. The NRC staff also concludes that WBN, Unit 2 will continue to meet the requirements of GDCs 10, 15, 16, 38, 50, 61, and 62, 10 CFR 50.46, 10 CFR 50.62, and 10 CFR 50.68 following the implementation of the proposed 1.4 percent MUR power uprate.

3.3 <u>MECHANICAL, STRUCTURAL, AND MATERIAL COMPONENT INTEGRITY AND DESIGN</u>

3.3.1 <u>Reactor Vessel, Reactor Vessel Internals, Reactor Coolant System, and Supports</u>

3.3.1.1 Regulatory Evaluation

The NRC regulations in 10 CFR Part 50, Appendix K, require 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 ECCS analyses for LOCAs. This requirement is included to ensure that instrumentation uncertainties are adequately accounted for in these analyses. The NRC regulations in 10 CFR Part 50, Appendix K, allow 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 proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error.”

The NRC staff's review of the LAR in the areas of mechanical engineering and inservice testing (IST) focused on verifying that the licensee has provided reasonable assurance that the structural and pressure boundary integrity of structures, systems, and components (SSCs) at WBN, Unit 2 will continue to be adequately maintained following the implementation of the proposed MUR power uprate under normal, upset, emergency, and faulted loading conditions, as applicable.

The NRC staff's assessment of the WBN, Unit 2 LAR in the areas of mechanical engineering and IST considered the following NRC regulations:

10 CFR 50.55a, "Codes and standards," and GDC 1, "Quality standards and record," as they relate to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety functions to be performed;

GDC 2, "Design bases for protection against natural phenomena," as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal and accident conditions;

GDC 4, "Environmental and dynamic effects design bases," 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 associated with normal and accident conditions and these structures and components being appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids;

GDC 14, "Reactor coolant pressure boundary," as it relates to the reactor coolant pressure boundary being designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; and

GDC 15, "Reactor coolant system design," as it relates to the RCS being designed with sufficient margin to ensure that the design conditions are not exceeded.

The design and licensing bases establish the principal means by which the facility demonstrates compliance with applicable NRC regulations. As such, the NRC staff's review primarily focused on verifying that the design and licensing basis requirements related to the structural and pressure boundary integrity of SSCs affected by the LAR would continue to be satisfied at MUR power uprate conditions. This, in turn, provides reasonable assurance that compliance with the applicable NRC regulations will be maintained upon implementation of the proposed MUR power uprate.

The primary guidance for MUR power uprate LARs is outlined in RIS 2002-03, which provides a guideline for organizing the LARs. Section IV, "Mechanical/Structural/Material Component Integrity and Design," of RIS 2002-03 provides information on the scope and detail of the information that should be submitted to the NRC regarding the impact that an MUR power uprate has on the structural and pressure boundary integrity of SSCs affected by the implementation of the MUR power uprate.

3.3.1.2 Technical Evaluation

The NRC staff's review covers the structural and pressure boundary integrity of the piping, components, and supports that make up the NSSS and the balance-of-plant (BOP) systems.

The review also includes an evaluation of new or existing SSCs, which are affected by the implementation of the proposed MUR power uprate. Specifically, this review focuses on the impact of the proposed MUR power uprate on the structural integrity of the WBN, Unit 2 pressure-retaining components and their supports and the reactor vessel (RV) internals.

The NRC staff also considered the impact of the proposed MUR power uprate at WBN, Unit 2 on postulated high energy line break (HELB) locations and corresponding dynamic effects resulting from the postulated HELBs, including pipe whip and jet impingement. A review of the impact of the MUR power uprate on moderate energy pipe rupture locations was also performed. The staff's review focused on verifying that the licensee has provided reasonable assurance of the structural and pressure boundary integrity of the piping systems, components, component internals and their supports under normal and transient loadings, including those due to postulated accidents and natural phenomena, such as earthquakes.

The proposed MUR power uprate will increase the reactor core power level from 3411 MWt to 3459 MWt at WBN, Unit 2. In accordance with the 10 CFR Part 50, Appendix K, requirements, the licensee notes in the LAR that the maximum analyzed thermal power of 3479 MWt corresponding to 102 percent of 3411 MWt remains unchanged. As noted in the LAR, the licensee previously performed these analyses assuming a power level of 3479 MWt, and the implementation of the MUR power uprate would revise the RTP to a level lower than that for which the licensee has already analyzed (i.e., 3459 MWt).

3.3.1.2.1 Power Uprate Evaluation Parameters and Design Bases

In Table IV.1-1, "WBN Unit 2 Nuclear Steam Supply System Design Parameters for Current and MUR Power Uprate Conditions," of Enclosure 2 to the LAR, the licensee provided the pertinent temperatures, pressures, and flow rates for the current and proposed uprated conditions. The licensee evaluated the effects of the proposed MUR power uprate at a bounding power level of 102 percent RTP (3479 MWt) for specific analyses. This power level corresponds to the proposed level following the implementation of the MUR power uprate (3459 MWt) plus the revised uncertainty of 0.6 percent.

Section II.1.D.ii, "UFSAR Chapter 15 Analyses," of Enclosure 2 to the LAR indicated that analyses for accidents that are DNB limited are performed at 101.4 percent of 3411 MWt (3459 MWt), plus an RCP net heat input of 16 MWt. For accidents that are not DNB limited, analyses are based upon a core power of 3411 MWt and an NSSS power of 3425 MWt. An uprated core power of 3459 MWt and an NSSS power of 3475 MWt are also supported via evaluation, based upon a redefinition of the 2 percent power uncertainty (i.e., from 2 to 0.6 percent).

As shown in the Table IV.1-1 of Enclosure 2 to the LAR, there is no change in the RCS operating pressure (2250 psia) as a result of the MUR power uprate. The RCS mechanical design flow of 105,000 gallons per minute (gpm) per loop would also remain unchanged after implementation of the MUR power uprate. At full power, the implementation of the MUR power uprate would yield a hot leg temperature (T_{hot}) of 619.1 °F (from the current temperature of 618.6 °F) and a cold leg temperature (T_{cold}) of 557.3 °F (from the current temperature of 557.8 °F), resulting in no change to the average RCS temperature. The main steam (MS) pressure would decrease by 7 psia at the MUR power uprate conditions, and the MS steam flow would increase from 15.08 million pounds mass per hour (Mlbm/hr) to 15.39 Mlbm/hr at the MUR power uprate conditions. The FW temperature would increase by 1.8 °F to 441.8 °F as a result of MUR power uprate.

The WBN UFSAR describes the design criteria applicable to the WBN, Unit 2 SSCs, including loads, load combinations, and acceptance criteria stipulated by the applicable codes of record for these SSCs. In Section IV.1.A of Enclosure 2 to the LAR, the licensee notes that implementation of the LAR would not change current operating transients or introduce additional transients. As such, loads resulting from these transients that are used in the structural evaluations of SSCs would not be affected. Similarly, the proposed MUR power uprate would have no effect on the deadweight and seismic loads of existing SSCs. Therefore, the NRC staff has determined that the loads used in the existing AORs for these SSCs remain valid.

3.3.1.2.2 Pressure-Retaining Components and Component Supports

As stated in Section IV.1 of RIS 2002-03, the LAR should contain a discussion of the effect of the power uprate on the structural integrity of major plant components. For components that are bounded by existing AORs, the discussion should cover accidents and transients for which the existing AOR bounds plant operation at the proposed uprated power level. For components that are not bounded by AORs, a detailed discussion should be provided. The evaluations should focus on determining what impact the MUR power uprate would have on the AOR for a particular SSC in order to determine whether the AOR for the particular SSC needs to be revised as a result of the MUR power uprate. If the AOR for a particular SSC was performed using conditions that bound those that will be present at the proposed MUR power level, no further evaluation is required.

Section IV.1.D of the LAR notes that the applicable Code of record for the WBN, Unit 2 RCS is American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section III. As discussed in the WBN UFSAR, the reactor internals for WBN, Unit 2 were fabricated prior to Subsection NG of the ASME BPV Code becoming an NRC requirement. However, with the exception of the Code Stress Report and Code Stamp, the reactor internals effectively satisfy the design and fabrication requirements of Subsection NG of the ASME BPV Code.

The WBN UFSAR provides a listing of the ASME BPV Code Class 1 Code Cases used for the RCS at WBN, Unit 2. The UFSAR identifies the Code design criteria for interfacing systems. The UFSAR provides a similar listing of Code Cases and provisions of later Code editions and addenda used in analysis of fluid systems. No stress/fatigue analyses were revised and, therefore, no Codes of record changed.

The WBN, Unit 2 licensee confirmed that the MUR power uprate evaluations did not include any changes to the tabulated design Codes of record. The pressure-retaining components and component supports, including piping and pipe supports, which must be evaluated in support of an MUR power uprate include the following: the reactor pressure vessel (RPV), including the RPV shell, RPV nozzles and supports; reactor core support structures and vessel internals; the pressure-retaining portions of the control rod drive mechanisms (CRDMs); NSSS piping, pipe supports, and branch nozzles associated with the RCS; BOP piping and supports; SGs, including their supports, the SG shells, secondary side internal support structures and nozzles; the pressure retaining portions of the RCPs; the pressurizer, including the pressurizer shell, nozzles, and the surge line; and safety-related valves. Further, Section IV.1.B of RIS 2002-03 indicates that the discussion should identify and evaluate any changes related to the power uprate in the following areas:

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

In reviewing the licensee's evaluation of pressure-retaining components and their supports, the NRC staff focused on determining whether those components and supports would be affected by the proposed MUR power uprate. Affected components and supports refer to those for which their AORs are not bounded at MUR power uprate conditions. Pressure-retaining components and their supports generally remain unaffected by the proposed MUR power uprate based on the fact that they have been analyzed at conditions that are more limiting than those that will be present at MUR power uprate conditions (i.e., bounded). The licensee was able to disposition a number of components and their associated supports as unaffected by the proposed MUR power uprate, based on whether the plant parameter changes resulting from the MUR power uprate would affect the loads included in the AORs for the components and their supports. Based on its evaluations of the impact of the proposed MUR power uprate, the licensee stated that the existing AORs related to the structural and mechanical qualifications of the following SSCs are unaffected by the uprate at WBN, Unit 2: the RPV, RPV nozzles, and RPV supports; the pressure-retaining portions of the CRDMs; RCS piping and supports and loop branch nozzles; pressurizer shell, nozzles, and surge line; the SGs, including the shells, nozzles, and secondary side internal support structures; and the pressure-retaining portions of the RCPs.

The NRC staff reviewed the licensee's evaluation of the structural integrity of BOP piping systems for demonstrating that the BOP piping systems will continue to meet their design bases under the proposed MUR power uprate conditions and remain bounded by the current AORs at MUR power uprate conditions. Similarly, the licensee confirmed that the proposed MUR power uprate has no effect on the structural integrity of safety-related valves at WBN, Unit 2, and that these also remain bounded by their current AORs. Based on these considerations, the NRC staff concludes that all pressure-retaining components and supports, including piping and pipe supports, remain bounded at the proposed MUR power uprate conditions.

The NRC staff considered the licensee's assessments of the pressure-retaining components and component supports acceptable based on the following considerations: (1) the licensee's approach to disposition SSCs as unaffected by the proposed MUR power uprate is consistent with RIS 2002-03; (2) the licensee confirmed that the existing AORs for the specified SSCs remain bounding when considering the plant parameter changes at the proposed MUR power uprate level, ensuring that there will be no impact on the structural and pressure boundary integrity of these SSCs at the MUR power uprate level; and (3) the magnitudes of plant parameter changes, as documented in Table IV.1-1 of Enclosure 2 to the LAR, are generally minor and support the licensee's assessment, which concludes that all pressure-retaining components remain bounded. Based on these considerations, the NRC staff concludes that there is reasonable assurance that the structural and pressure boundary integrity of the

WBN, Unit 2 SSCs will be adequately maintained following the implementation of the proposed MUR power uprate.

3.3.1.2.3 Reactor Vessel Internals

In accordance with Section IV.1.A.ii of RIS 2002-03, the licensee evaluated the effects of the proposed MUR power uprate on the WBN, Unit 2 RV internals. Section IV.1.B of RIS 2002-03 indicates that for those SSCs, including RV internals, whose AORs are affected by the implementation of an MUR power uprate, the licensee should address the following, as they relate to the impact of the uprate on the AORs: stresses, cumulative usage factors (i.e., fatigue), flow-induced vibration (FIV), and changes in temperature, pressure, and flow rates resulting from the MUR power uprate. The licensee summarized its evaluation of the effects of the proposed MUR power uprate on the structural integrity of the RV internals in its LAR.

Mechanical and structural evaluations were performed by the licensee to determine any effects on the RV internals due to the conditions that would be present following the proposed MUR power uprate. The mechanical evaluations of FIV performed by the licensee are summarized in Section IV.1.B.iii of Enclosure 2 to the LAR. These evaluations focused on the potential for an increase in the vibratory response of the RV internals resulting from changes in the flow field at the MUR power level. An increase in vibratory response can introduce increased alternating stress intensities and subsequently higher cyclic fatigue of the RV internals. Per the values in Table IV.1-1 of Enclosure 2 to the LAR, the volumetric mechanical design flow remains unchanged for the proposed MUR power uprate. Hence the vortex shedding frequencies remain unchanged. Also, the temperature changes due to the MUR power uprate are less than 0.1 percent, which causes a negligible change in the frequencies of the internals. Thus, the stresses imparted on the RPV internals due to FIV remain unchanged as a result of the proposed MUR power uprate conditions, and the existing AORs remain bounding. Based on these considerations, the licensee confirmed that the FIV characteristics of the RV internals are bounded by the current AORs. The changes in operating temperatures are provided in Table IV.1-1 of Enclosure 2 to the LAR. The average temperature is unchanged, with the cold leg temperature decreasing by 0.5 °F and the hot leg temperature increasing by 0.5 °F. These changes have minimal impact on the mechanical and structural evaluations. Based on this assessment, the licensee noted that the RV internals remain bounded at MUR power uprate conditions, and that no revision to the AORs is required to support the proposed MUR power uprate.

The NRC staff reviewed the licensee's assessment of the RV internals and considers the licensee's evaluation acceptable based on the following rationale. With respect to the effects of the MUR power uprate on the FIV of the RV internals, the NRC staff has determined that the licensee's assessment is acceptable given that it is shown in the licensee's submittal that the RCS operating parameters (flow, temperature, and pressure) that directly affect FIV either do not change or do not change sufficiently to affect the FIV of the RV internals. For the structural evaluations, the NRC staff has determined that the licensee's conclusion that the RV internals are bounded by the current AORs at the proposed MUR power uprate conditions is acceptable based on the fact that the RV internals have previously been evaluated at a power level that is greater than the proposed MUR power uprate level. Additionally, a comparison between the RCS operating parameters before and after the MUR power uprate suggests that there will be a minimal impact on the loads used in the evaluation of the RV internals for structural integrity. Further, no abnormal loads (i.e., transient and seismic) are changing as a result of the MUR power uprate. Therefore, the NRC staff concludes that the design basis analyses of the RV internals will remain unaffected and bounding following the MUR power uprate.

3.3.1.2.4 Nuclear Steam Supply System Piping, Pipe Supports, and Branch Nozzles

The licensee analyzed the operating conditions under the proposed MUR power uprate against the existing design basis analyses for the reactor coolant piping and supports and determined that the MUR power uprate conditions are bounded by the current design analyses. These analyses included the piping stress analyses for reactor coolant piping, which take into account in-line piping and the branch connections welded onto the surface of the in-line piping and associated support system, as well as the piping stress analysis of the pressurizer surge line. Since the changes due to the MUR power uprate remain bounded by the existing design basis analyses, the NRC staff finds that the effects on the nuclear steam supply system piping, pipe supports, and branch nozzles will not result in an increased risk of damage or failure as a result of the uprate and, therefore, are acceptable.

3.3.1.2.5 Stresses

The breadth of the NRC staff's review of the stresses was whether the current analyses of stresses on the different RCS systems remain bounding for the plant as a result of the MUR power uprate. The licensee confirmed that there were no changes in the RCS design or operating transient conditions that would be made and that design transients and design interface loads remained bounded by the current evaluations due to the margins in the primary stresses, primary plus secondary stresses, and fatigue usage factors in the RCS piping loop. The NRC staff finds that the stresses evaluated for the plant will not exceed previously evaluated conditions due to the proposed MUR power uprate and, therefore, are acceptable.

3.3.1.2.6 Cumulative Usage Factors

The NSSS components, piping, and interface systems were reviewed to determine whether the revised design conditions due to the MUR power uprate would impact the existing design basis analyses. The NRC staff determined that the revised conditions remain bounded by the current analyses and that, therefore, the postulated transients previously evaluated for WBN, Unit 2 do not need to be revised. Additionally, this means that the current stress and fatigue values (including cumulative usage factors) remain valid with the proposed MUR power uprate and, therefore, are acceptable.

3.3.1.2.7 Postulated Pipe Ruptures and Associated Dynamic Effects

High-energy fluid systems are defined in Appendix 3.6A of the WBN UFSAR as systems where, during normal plant conditions, the maximum operating temperature and pressure exceed 200°F and 275 psig. These systems can be classified as moderate energy if these temperature and pressure limits are exceeded for less than 1 percent of the normal operating life span of the plant or 2 percent of the time that the system is running to accomplish its design function. Additionally, all other systems that are not classified as high energy are considered moderate energy.

As discussed in its LAR, the licensee evaluated the effects of the proposed MUR power uprate on systems classified as high energy to determine whether any changes to the HELB AOR will result from the MUR power uprate. The licensee stated in a summary to its assessment that the current AORs were reviewed to determine whether the MUR power uprate would have any impact on the current HELB AOR. The licensee determined that because the temperature and pressure changes in high energy systems are considered nominal, no new HELB locations would be required to be postulated as a result of the MUR power uprate.

For postulated moderate energy line breaks (MELBs), the licensee confirmed that the MUR power uprate has no effect on moderate energy piping systems. Therefore, no new moderate energy pipe cracks would be required to be postulated.

The licensee summarized its assessment of the impact of the MUR power uprate on jet impingement and thrust forces (dynamic effects) in Section IV.1.B.viii of Enclosure 2 to the LAR. The NRC staff considers this assessment to be acceptable.

The licensee stated that it had justified the elimination of large primary loop pipe rupture and pressurizer surge line pipe rupture from the design basis for WBN, Unit 2 by using leak-before-break (LBB) concepts. The licensee also confirmed that piping loads used in the LBB evaluation are not affected by the MUR power uprate and concluded that the LBB evaluation remains acceptable and is bounded by existing AORs. The licensee determined that these loads are not affected by the MUR power uprate due to the fact that the changes in the temperatures and pressures of these systems resulting from the MUR power uprate were within the bounds of the temperatures and pressures that have been previously evaluated.

The NRC staff has reviewed the licensee's evaluations related to determinations of pipe rupture locations and their corresponding dynamic effects and considers the licensee's assessments performed for these areas to be acceptable. This acceptance is based on the information presented above, which demonstrates that the AORs related to HELBs, MELBs, and dynamic effects resulting from postulated pipe ruptures will remain bounding under the proposed MUR power uprate. The NRC staff finds this conclusion acceptable, given the small magnitude in temperature and pressure increases that accompany the MUR power uprate. Correspondingly, these small changes generally have no impact on pressure-retaining components, such as piping.

3.3.1.2.8 Safety-Related Valves, Pumps, and Dynamic Restraints (Snubbers)

Safety-Related Valves

The NRC staff reviewed the analyses performed by the licensee for safety-related valves at WBN, Unit 2 in support of the proposed MUR power uprate. The staff examined the overall design change and plant specific evaluations. The staff's acceptance criteria for reviewing the safety-related valve analysis are based on 10 CFR 50.55a.

In the LAR, the licensee described the impact of the proposed MUR power uprate on the existing design basis analysis for the WBN, Unit 2 safety-related valves. No changes in RCS flow, design, or operating pressure are proposed as part of the power uprate. The licensee's evaluations determined that the temperature changes due to the power uprate are bounded by those used in the existing analyses. As a result, none of the safety-related valves required a change to their design or operation as a result of the MUR power uprate. The analyses also confirmed that the existing capacity of the main steam safety valves is adequate for overpressure protection at MUR power uprate conditions, and that the existing lift setpoints are unchanged. The NRC staff reviewed the licensee's analysis and determined that none of the safety-related valves will require a change to their design or operation as a result of the proposed MUR power uprate.

The pressurizer safety valves (PSVs) and the power-operated relief valves (PORVs) and their associated block valves located on top of the pressurizer provide overpressure protection for the RCS. The WBN, Unit 2 MUR power uprate is bounded by the current design basis event transient analyses. Therefore, there is no adverse impact on the pressurizer overpressure protection valves from the MUR power uprate, and the AOR for the pressurizer overpressure protection valves remains bounding at MUR power uprate conditions.

The motor-operated valve (MOV) program for WBN, Unit 2 is not impacted by the MUR power uprate. For example, there are no changes to any regulatory requirements or the scope of the MOV program. The systems that contain MOVs within the scope of the MOV program were evaluated and determined to remain within the existing design parameters after implementation of the MUR power uprate or were determined to not be impacted by the power uprate. The MUR power uprate does not alter the basis, scope, or content of the MOV program. For example, no MOVs will be added to or deleted from the MOV program due to the MUR power uprate. No maintenance or material changes for any MOVs will be required. Therefore, no changes are required to the existing MOV program.

The air-operated valve (AOV) program for WBN, Unit 2 Category 1 and 2 valves is not impacted by the MUR power uprate. The systems that contain these AOVs within the scope of the program were evaluated and determined to remain within existing design parameters after implementation of the MUR power uprate or were determined to not be impacted by the power uprate. The MUR power uprate does not alter the basis, scope, or content of the AOV program. No AOVs will be added to or deleted from the AOV program due to the MUR power uprate. No maintenance or material changes for any AOVs will be required. Therefore, no changes are required to the existing AOV program.

Based on these evaluations, the NRC staff concluded that the performance of existing safety-related valves is acceptable with respect to the proposed MUR power uprate at WBN, Unit 2.

Safety-Related Pumps

The NRC staff reviewed the licensee's analysis of safety-related pumps in support of the WBN, Unit 2 MUR power uprate. The NRC's acceptance criteria for reviewing the safety-related pumps analysis is based on the requirements in 10 CFR 50.55a.

The NRC staff reviewed the impact of the proposed MUR power uprate conditions on the existing design basis analyses for safety-related pumps. The evaluation showed that there are no significant changes to the maximum operating conditions, and no changes to the design basis requirements that would affect pump performance. The current plant design is considered bounding under MUR conditions, and requires no modifications to pump systems. Therefore, the NRC staff concludes that the performance of existing safety-related pumps is acceptable with respect to the MUR power uprate at WBN, Unit 2.

Inservice Testing Program

In the LAR, the licensee described the review of the IST program for safety-related pumps and valves at the proposed MUR power uprate conditions. The IST Code of Record for WBN, Unit 2 is the ASME Operation and Maintenance of Nuclear Power Plants (OM Code), 2004 Edition through the 2006 Addenda, as incorporated by reference in 10 CFR 50.55a. The IST program assesses the operational readiness of pumps and valves within the scope of the ASME OM

Code. There were no significant changes to operating conditions or the design basis requirements that would adversely affect component performance, test acceptance criteria, or reference values. Therefore, the existing IST program at WBN, Unit 2 will not be impacted by the MUR power uprate. Based on this evaluation, the NRC staff concludes that the IST program at WBN, Unit 2 will be acceptable at the proposed MUR power uprate conditions.

Inservice Examination and Testing Program for Dynamic Restraints (Snubbers)

In additional information submitted by letter dated April 29, 2020, the licensee described its review of the inservice examination and testing program for safety-related snubbers at WBN, Unit 2 at the proposed MUR power uprate conditions (Reference 2). The IST Code of Record for WBN 2 is the ASME OM Code, 2004 Edition through the 2006 Addenda, as incorporated by reference in 10 CFR 50.55a. The inservice examination and testing snubber program assesses the operational readiness of snubbers within the scope of the ASME OM Code. There were no significant changes to operating conditions or the design basis requirements that would affect component performance, test acceptance criteria, or reference values for snubbers at WBN, Unit 2. Therefore, the existing snubber program at WBN 2 will not be impacted by the MUR power uprate. Based on the licensee's evaluation, the NRC staff concludes that the snubber program at WBN, Unit 2 will be acceptable at the proposed MUR power uprate conditions.

Potential Loads on Safety-Related Dynamic Restraints (Snubbers)

The NRC staff reviewed the licensee's evaluation of safety-related snubbers impacted by load increases due to the MUR power uprate. The WBN, Unit 2 IST program includes snubbers that are required to ensure the integrity of the reactor coolant pressure boundary or required for systems and components that perform a specific function to bring the reactor to the safe shutdown condition, maintain the safe shutdown condition, or mitigate the consequences of an accident.

The WBN, Unit 2 systems within the scope of the IST program were reviewed and determined to not be impacted by the MUR power uprate. No changes were identified for any of the associated piping analyses. The support loads are not changed and the snubbers are not affected by the MUR power uprate. Based on the evaluation, the NRC staff concludes that the safety-related snubbers are not impacted by the WBN, Unit 2 MUR power uprate.

Inservice Inspection Program

Pursuant to 10 CFR 50.55a(g)(4), the inservice inspection (ISI) program contains the WBN, Unit 2 inspection schedule and requirements in accordance with ASME Code, Section XI for Class 1, 2, and 3 components. The ISI program for Class 1 components is found in WBN UFSAR Section 5.2.8 and the program for Class 2 and 3 components is found in UFSAR Section 6.6 (References 45 and 34). The WBN ISI program was reviewed to determine if any of the changes to the plant due to the MUR power uprate would introduce significant effects to the plant that would require any updates or revisions to the program. The NRC staff's review of this section is based upon whether the increases in pressure and temperature incur any additional stresses that would require additional or more frequent examinations of the same or additional components. The NRC staff concludes that the changes due to the proposed MUR power uprate are not significant enough to require any changes to the ISI program.

3.3.1.3 Technical Conclusion

The NRC staff has reviewed the licensee's assessment of the impact of the proposed MUR power uprate on the structural and pressure boundary integrity of pressure-retaining components and supports and RV internals at WBN, Unit 2. Additionally, the NRC staff reviewed the licensee's assessment of the effects on the HELB and MELB AORs for WBN, Unit 2, including associated dynamic effects. Based on its review, the NRC staff concludes that the LAR is acceptable with respect to the structural integrity of SSCs affected by the MUR power uprate. This acceptance is based on the demonstration that the intent of the applicable NRC regulatory requirements will continue to be satisfied following implementation of the MUR power uprate at WBN, Unit 2. Specifically, the licensee demonstrated that: (1) the structural and pressure boundary integrity pressure retaining components and supports, including piping and pipe supports, at WBN, Unit 2 are not affected by the proposed MUR power uprate, as evidenced by the fact that their AORs are unaffected; (2) the RV internals at WBN, Unit 2 also remain unaffected, when considering the impact of the MUR power uprate on the FIV characteristics and structural integrity of the RV internals; (3) the WBN, Unit 2 AORs related to the postulation of HELB and MELB locations, including dynamic effects associated with these postulated pipe ruptures, will remain bounding; and (4) safety-related valves, pumps, and snubbers remain unaffected by the MUR power uprate. Based on these considerations, the NRC staff concludes that there is reasonable assurance that the structural integrity of SSCs at WBN, Unit 2 will be adequately maintained following the MUR power uprate, such that the MUR power uprate will not adversely impact the ability of these SSCs to perform their intended functions.

3.3.2 <u>Reactor Vessel Integrity</u>

In Section IV.1.C of Enclosure 2 to the LAR, the licensee addressed the impacts of the MUR power uprate on the integrity of the reactor vessel. The licensee's discussion and the NRC staff's evaluation of these impacts are addressed below.

3.3.2.1 Regulatory Evaluation

The NRC staff performed this safety evaluation based on the following regulations and guidance:

10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events," which provides the requirements, methods of evaluation, and safety criteria for pressurized thermal shock (PTS) assessments.

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," which provides fracture toughness requirements for ferritic materials in the RCPB, including requirements for the Charpy upper-shelf energy (USE) for protecting RPV beltline materials against non-brittle failure, and requirements for calculating RCS pressure-temperature (P-T) limits for protection against brittle fracture.

Appendix H to 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements," which contains requirements for the material surveillance program required to monitor changes in the fracture toughness properties of ferritic materials in the RPV beltline region that result from exposure of these materials to neutron irradiation and the thermal environment.

Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," which contains methodologies for determining the increase in transition temperature and the decrease in USE resulting from neutron irradiation (Reference 46).

Regulatory Guide 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," which provides state-of-the-art calculations and measurement procedures that are acceptable to the NRC staff for determining pressure vessel fluence (Reference 47).

3.3.2.2 Licensee's Evaluation

The licensee, in Sections IV.1.C.i through IV.1.C.vi of Enclosure 2 to the LAR, stated that it is using an existing evaluation (approved by the NRC staff) of the effects of neutron fluence on the RPV structural integrity at WBN, Unit 2. The following paragraphs include a summary the licensee's evaluation.

PTS

PTS calculations were performed for the WBN, Unit 2 RPV using the latest procedures specified in 10 CFR 50.61 for end of license (EOL) (i.e., 32 effective full-power years (EFPY)) neutron fluence values. The RPV neutron fluence projections assumed an MUR power uprate of 1.4 percent beginning with Cycle 2 operation and tritium producing core (TPC) designs, which include tritium producing burnable absorber rods (TPBARs) beginning with Cycle 4 operation.

The results of this analysis were summarized in Westinghouse Report WCAP-18191-NP, Revision 0, "Watts Bar Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and Supplemental Reactor Vessel Integrity Evaluations," dated May 2017 (Reference 48). This report was reviewed and accepted by the NRC staff as discussed in an SE contained in NRC letter to TVA, "Watts Bar Nuclear Plant, Units 1 and 2 – Issuance of Amendment Regarding Revision to Watts Bar Nuclear Plant, Unit 2, Technical Specification 4.2.1, 'Fuel Assemblies,' and Watts Bar Nuclear Plant, Units 1 and 2, Technical Specifications Related to Fuel Storage (EPID L-2017-LLA-0427)," dated May 22, 2019 (Reference 44).

Fluence

Fluence projections related to the proposed WBN, Unit 2 MUR power uprate and TPC implementation and the results of those analyses were documented in WCAP-18191-NP, Revision 0. The licensee stated that fluence calculations performed for WBN, Unit 2 adhered to the NRC-approved methodology described in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," dated May 2004 (Reference 49). This methodology follows the guidance and meets the recommendations of RG 1.190 (Reference 47). The NRC staff's review of the licensee's fluence evaluation is provided below in Section 3.3.3.

P-T Limits, Upper Shelf Energy, and Adjusted Reference Temperature

The licensee stated that WCAP-18191-NP, Revision 0, included P-T curves, USE, and adjusted reference temperature (ART) values that were calculated using the MUR power uprate neutron fluence values. The NRC staff previously reviewed and accepted the methodology, which is addressed in the NRC staff's SE dated May 22, 2019.

Cold Overpressure Mitigation System (COMS)

The cold overpressure mitigation system (COMS) was addressed in WCAP-18191-NP, Revision 0, and the licensee stated that the review of the P-T limit and fluence calculations confirmed that these critical inputs to the COMS setpoint calculation bound the proposed WBN, Unit 2 MUR power uprate conditions. The NRC staff previously reviewed and accepted the methodology, which is addressed in the NRC staff's SE dated May 22, 2019.

Surveillance Capsule Withdrawal Program

The licensee stated that any revision to withdrawal schedule requires NRC approval. The NRC staff's SE dated May 22, 2019, includes the evaluation of the surveillance capsule withdrawal program and it was found to be acceptable.

3.3.2.3 Technical Evaluation

Based on its review of its previous SE dated May 22, 2019, the NRC staff determined that the licensee's evaluation of PTS, P-T curves, USE, COMS, and ART values are still valid for WBN, Unit 2 MUR power uprate operations for 32 EFPY, for the following reasons:

- The licensee used conservative fluence values that considered the MUR power uprate including fluence projections related to TPC designs which include TPBARs. The fluence projections also include a small amount of conservatism based on the assumption that the MUR would be implemented at WBN, Unit 2 during Cycle 2, whereas the MUR is presently not planned for implementation until Cycle 4. These fluence values were projected to 32 EFPY using NRC-approved fluence methodology and they are bounding. They were used in the evaluation of PTS, P-T curves, USE, COMS, and ART at WBN, Unit 2.

- The licensee performed its PTS calculations for the WBN, Unit 2 RPV using the latest procedures specified in 10 CFR 50.61 for EOL (i.e., 32 EFPY) neutron fluence values.

- Establishment of Heatup and Cooldown curves complies with the guidance in RG 1.99, Revision 2.

- The development of P-T curves, USE, and ART values were calculated using the methods addressed in RG 1.99, Revision 2.

- The licensee's fluence evaluation complies with the guidance in RG 1.190.

- The NRC staff also noted that the RPV Material Surveillance Program at WBN, Unit 2 is consistent with the Appendix H to 10 CFR Part 50 criterion.

3.3.2.4 Technical Conclusion

Based on its review, the NRC staff concludes that the implementation of the proposed MUR power uprate and the fuel arrangement using TPC with TPBARS will not affect the existing NRC-approved evaluation of the structural integrity of the RPV base metals and welds. The NRC staff's SE dated May 22, 2019, is valid and bounding for WBN, Unit 2 for 32 EFPY, because the SE reflects consideration of analyses and inputs that include the TPC characteristics as well as the MUR power uprate.

3.3.3 Neutron Fluence Evaluation

The neutron fluence evaluation is addressed in Section IV.1.C.ii of Enclosure 2 to the LAR.

3.3.3.1 Background

The fracture toughness of pressure vessel materials is related to a parameter called the material’s reference temperature for nil-ductility transition. The reference temperature is defined by a correlation of the fluence, material chemistry, initial reference temperature, and margin to account for uncertainties in the correlation and input values. Methods for determining the fast neutron fluence are therefore necessary to estimate the fracture toughness of the pressure vessel materials.

3.3.3.2 Applicable Regulatory Requirements

The NRC staff performed this safety evaluation based on the following regulations and guidance:

GDC 14, “Reactor coolant pressure boundary,” GDC 30, “Quality of reactor coolant pressure boundary,” and GDC 31, “Fracture prevention of reactor coolant pressure boundary,” which require the design, fabrication, and maintenance of the reactor coolant pressure boundary with adequate margin to assure that the probability of rapidly propagating failure of the boundary is minimized. In particular, GDC 31 explicitly requires consideration of the effects of irradiation on material properties.

Regulatory Guide 1.190, which provides state-of-the-art calculations and measurement procedures that are acceptable to the NRC staff for determining pressure vessel fluence.

3.3.3.3 Summary of Information Provided by the Licensee

The fluence evaluation is addressed in Section IV.1.C.ii of Enclosure 2 to the LAR. The licensee stated that the applicable fluence calculations were documented in WCAP-18191-NP and were reviewed and accepted by the NRC staff in concert with a recent license amendment request (i.e., the NRC staff’s SE dated May 22, 2019) (Reference 44). These calculations were based on cycle-specific core designs that assumed the following:

- Implementation of the MUR power uprate at the beginning of Cycle 2,
- Early transition from out-in core loading to low-leakage core loading, and
- Implementation of TPBAR cycle designs beginning in Cycle 4.

The licensee also stated that these calculations were performed in accordance with WCAP-14040-A, Revision 4 (Reference 49). These methods have been approved for use by the NRC staff and adhere to RG 1.190.

3.3.3.4 Technical Evaluation

A slight increase in core thermal power associated with an MUR power uprate can lead to an increase in the core peripheral neutron flux, to which the reactor vessel materials are exposed.

However, because the licensee's fluence evaluation assumed that the proposed MUR power uprate was implemented in Cycle 2, the NRC staff notes that the flux projections for Cycles 2 and 3 would be expected to conservatively overestimate the core flux by a small amount, as these two cycles did not operate at the MUR power uprate level. Otherwise, the fluence analysis remains acceptable as noted in the NRC staff's SE dated May 22, 2019. Except for the delay in the MUR power uprate implementation, the calculations reflect cycle-specific core designs and are based on NRC-approved methods that adhere to RG 1.190.

3.3.3.5 Technical Conclusion

Based on the considerations discussed above, the NRC staff determined that the licensee used acceptable methods to estimate the fluence for the reactor vessel materials, with assumptions that represent prior and planned operating conditions, with a small additional measure of conservatism. Thus, the NRC staff determined that the licensee's fluence estimate was acceptable for accounting for the effects of irradiation on the reactor pressure vessel materials, consistent with the requirements of GDCs 14, 30, and 31. The NRC staff therefore concludes that the proposed MUR power uprate is acceptable with respect to reactor vessel neutron fluence.

3.3.4 <u>Chemical Engineering and Steam Generator Integrity</u>

The NRC staff reviewed the LAR in accordance with RIS 2002-03 concerning the following areas: (1) CVCS, (2) SG blowdown system, (3) SG tubes, secondary side internal support structures, shell, and nozzles, (4) flow-accelerated corrosion (FAC), and (5) rapidly propagating fatigue cracks in SG tubes.

3.3.4.1 Chemical and Volume Control System

3.3.4.1.1 Regulatory Evaluation

The CVCS provides a means for: (1) maintaining water inventory and quality in the RCS, (2) supplying seal-water flow to the RCPs and pressurizer auxiliary spray, (3) controlling the boron neutron absorber concentration in the reactor coolant, (4) controlling the primary water chemistry and reducing coolant radioactivity level, and (5) supplying recycled coolant for demineralized water makeup for normal operation and high-pressure injection flow to the ECCS in the event of postulated accidents.

The NRC staff has reviewed the safety-related functional performance characteristics of CVCS components as they relate to the control of primary water chemistry. The NRC staff performed this safety evaluation based on the following regulations and guidance: GDC 14, which states that, "The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture" and SRP Section 9.3.4, "Chemical and Volume Control System (PWR)" (Reference 50).

Additional information regarding the design of the CVCS can be found in WBN UFSAR Section 9.3.4, "Chemical and Volume Control System" (Reference 42).

3.3.4.1.2 Licensee Description

Section 9.3.4 of the WBN UFSAR contains a description of the CVCS and Table 5.2-10, “Reactor Coolant Water Chemistry Specifications,” contains the primary water chemistry limits (Reference 45). Additionally, the licensee stated that the WBN procedure for “System Chemistry Specifications” contains the information used to evaluate plant chemistry. This procedure reflects guidance found in the Electric Power Research Institute (EPRI) PWR Primary Water Chemistry Guidelines (Reference 51) and adheres to Action Levels 1, 2, and 3, with the same required actions as described in the EPRI guidance.

3.3.4.1.3 Staff Evaluation and Conclusion

The NRC staff finds the CVCS design at the proposed MUR power uprate conditions acceptable because the CVCS is designed with a temperature control valve that can divert bypass flow around the demineralizer if the temperature of the water exceeds the demineralizer temperature limit as per Section 9.3.4.2 of the WBN UFSAR. Additionally, the power uprate conditions are not expected to have a significant impact on primary water quality and therefore the staff has reasonable assurance that the ability of the CVCS to purify the primary water will not be impacted. The licensee also stated that the WBN procedures reflect guidance from the EPRI water chemistry guidance including the action levels and required actions discussed in the EPRI guidance. Based on this, the NRC staff has reasonable assurance that GDC 14 will continue to be met at the proposed MUR power uprate conditions and, therefore, the NRC staff concludes that the LAR is acceptable with respect to its review of the water chemistry related functions of the CVCS.

3.3.4.2 Steam Generator Blowdown System

3.3.4.2.1 Regulatory Evaluation

Control of secondary-side water chemistry is important for preventing degradation of SG tubes. The SG blowdown system (SGBS) provides a means for removing SG secondary-side impurities. The design basis of the SGBS includes consideration of expected and design flows for all modes of operation. The NRC staff’s review covered the ability of the SGBS to remove particulate and dissolved impurities from the SG secondary side during normal operation, including condenser in-leakage and primary-to-secondary leakage.

The NRC staff performed this safety evaluation based on the following regulations and guidance: GDC 14, which requires that the RCPB be designed to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture and SRP Section 10.4.8, “Steam Generator Blowdown System,” which provides review guidance for the NRC staff (Reference 52). Additionally, the SGBS is described in Section 10.4.8, “Steam Generator Blowdown System,” of the WBN UFSAR (Reference 53).

3.3.4.2.2 Licensee Description

In its LAR, the licensee stated that the SGBS will remain within its design basis at the proposed MUR power uprate conditions. This includes the blowdown flow rate needed to control secondary chemistry and buildup of solids in the SGs as these are tied to variables not impacted by the MUR power uprate. Additionally, there will be a small decrease in blowdown system inlet pressure, which is not expected to impact blowdown flow control. Therefore, the licensee concluded that the changes to secondary-side design parameters for the proposed MUR power uprate will not affect SG blowdown chemistry control or flow control.

3.3.4.2.3 Staff Evaluation and Conclusion

The NRC staff has reviewed the impacts from the proposed MUR power uprate on the SGBS and the effects on secondary side water chemistry control. Due to the small change in secondary side operating pressure, the SGBS flow capacity is not expected to have a significant effect on blowdown flow control and, therefore, the SGBS will be able to achieve the necessary flow rates. Additionally, because the rate of impurity ingress into the secondary water is independent of the rated power level, secondary water chemistry should not experience a significant change. The licensee also stated that the WBN procedures reflect guidance from the EPRI PWR secondary water chemistry guidance including the action levels and required actions discussed in the EPRI guidance (Reference 54). This provides the NRC staff with reasonable assurance that the SGBS capacity will be acceptable at the MUR power uprate conditions. WBN UFSAR Section 10.4.8.2, “System Description and Operation,” states that there is temperature monitoring at the entrance to the demineralizer bed and the capability to bypass the bed if the temperature exceeds the maximum allowable for the demineralizer resins. This provides the NRC staff with reasonable assurance that the increase in secondary side temperature will not affect the ability of the SGBS to remove impurities via the demineralizer resins.

The NRC staff has reviewed the licensee’s evaluation of the proposed MUR power uprate on the SGBS and has determined that the licensee has adequately addressed changes in system flow and impurity levels and their effects on the SGBS. The NRC staff has further determined that the licensee has demonstrated that the SGBS will continue to meet the requirements of GDC 14 following the implementation of the MUR power uprate. Therefore, the NRC staff concludes that the LAR is acceptable with respect to the SGBS.

3.3.4.3 Steam Generator Tubes, Secondary Side Internal Support Structures, Shell, and Nozzles

3.3.4.3.1 Regulatory Evaluation

SG tubes constitute a large part of the RCPB. As a result, their integrity is important to the safe operation of a reactor. The NRC staff’s review in this area covered the effects of changes in operating conditions resulting from the proposed MUR power uprate on SG materials and the SG program. The NRC staff’s review criteria for the SG Program are based on the WBN, Unit 2 TSs. TS 3.4.17, “Steam Generator (SG) Tube Integrity,” requires that SG tube integrity be maintained and TS 5.7.2.12, “Steam Generator (SG) Program,” governs the SG inspections for WBN, Unit 2. Details of the WBN, Unit 2 SGs can be found in WBN UFSAR Section 5.5.2, “Steam Generators” (Reference 45). Specific review criteria for this topic are contained in SRP Section 5.4.2.1, “Steam Generator Materials and Design,” for the SG materials (Reference 55) and Section 5.4.2.2, “Steam Generator Program,” for the SG program (Reference 56). Additionally, RIS 2002-03 recommends that the licensee provide a discussion regarding the impacts of the MUR power uprate on the structural integrity of the SG tubes, secondary side internal support structures, shell, and nozzles.

The review guidance in SRP Section 5.4.2.1 states, in part:

> The purpose of this review is to ensure that (1) the materials used to fabricate the steam generator are selected, processed, tested, and inspected to appropriate specifications, (2) the design of the steam generator limits the susceptibility of the materials to degradation and corrosion, (3) the fracture toughness of the ferritic materials is adequate, (4) the materials used in the steam generator are compatible with the environment to which they will be exposed, (5) the design of the secondary side of the steam generator permits the chemical or mechanical removal of chemical impurities, and (6) any degradation to which the materials are susceptible (including fracture) is avoided, can be managed through the inservice inspection program, or can be controlled through limits placed on operating parameters. Performing periodic steam generator inspections will ensure that the integrity of the steam generator is maintained at a level comparable to that in the original design requirements.

The review guidance in the SRP, Section 5.4.2.2, states, in part:

> The purpose of this review is to (1) ensure that the design of the steam generator is adequate for implementing a steam generator program and (2) verify that the steam generator program will result in maintaining tube integrity during operation and postulated accident conditions. The steam generator program is intended to ensure that the structural and leakage integrity of the tubes is maintained at a level comparable to that of the original design requirements.

3.3.4.3.2 Licensee Description

The WBN 2 SGs are Westinghouse model D3. Each SG utilizes mill-annealed SB163 Ni-Cr-Fe (nickel-chromium-iron) (Alloy 600) tubes. In its LAR, the licensee stated that the structural evaluation for the SGs applies to the MUR power uprate conditions, including the hardware changes made to the SGs after installation. The licensee described the operating parameters that will change at the MUR power uprate conditions and stated that the design conditions of the SGs bound the MUR power uprate conditions. Additionally, the licensee stated that the MUR power uprate will not change or add any new operating transients and, therefore, the existing stress reports for the SGs remain applicable. The licensee also stated that the SG tube repair criteria in TS 5.7.2.12 are not changed due to the MUR power uprate.

In Section IV.1.F of Enclosure 2 to the LAR, the licensee stated that it performed an evaluation in response to Bulletin 88-02, “Rapidly Propagating Fatigue Cracks in Steam Generator Tubes” (Reference 57), to address SG tube high cycle fatigue concerns. The licensee also stated that this analysis considered the MUR power uprate operating parameters.

3.3.4.3.3 Staff Evaluation and Conclusion

The NRC staff evaluated the material provided by the licensee and determined that the changes in operating conditions at MUR power uprate conditions would be relatively small and similar to previously approved MUR power uprates. Additionally, the licensee has analyzed the MUR power uprate conditions up to 10 percent tube plugging and stated that its structural analysis of record bounds the MUR power uprate conditions. Therefore, the NRC staff finds the impacts of the operating conditions at MUR power uprate conditions acceptable with respect to the SGs.

With respect to impact on the SG materials due to the MUR power uprate, the NRC staff has determined that the materials used in the SGs remain acceptable, the fracture toughness of the ferritic materials is adequate, the design limits the susceptibility of the materials to degradation and corrosion, the materials used in the SG remain compatible with the environment, the design permits the removal of impurities, and that any degradation that could occur is either avoided or can be managed.

In addition, as evaluated in Section 3.3.4.5 of this SE, the NRC staff has reviewed the licensee's evaluation of the impact of the power uprate on SG tube vibration and fatigue and determined that it remains within acceptable limits for safe operation.

With respect to the SG program, the NRC staff has determined that the changes in operating conditions have no effect on the ability to implement the SG program. As a result, the NRC staff has determined that the design of the SGs remains adequate for implementing the SG program. The changes in operating conditions may result in increased susceptibility to degradation and may result in increased degradation growth rates. Although this may occur, the NRC staff has determined that the SG program is still acceptable since it requires the licensee to continue to ensure tube integrity for the operating interval between inspections.

With respect to the tube repair criteria included in the TSs for the SG program, the small changes in operating conditions are expected to have a relatively small effect on the structural limits for the tubes. Since the tube repair criteria are determined from the structural limits, they may also be slightly affected by the MUR power uprate conditions. The NRC staff has determined that the tube repair criteria remain valid under the MUR power uprate conditions. This determination is based on the NRC staff's approval of repair criteria at other similarly designed and operated units and the performance-based requirement to ensure tube integrity for the operating interval between inspections. As a result of the above, the NRC staff has determined that the SG program remains acceptable for MUR power uprate conditions.

The NRC staff reviewed the licensee's evaluation of the effects of the MUR power uprate on SG tube integrity and has determined that the licensee has adequately assessed the continued acceptability of the plant's TSs in terms of the changes in temperature, differential pressure, and flow rates. The NRC staff has also confirmed that the licensee has a program that ensures SG tube integrity and that the applicability of the SG program has not changed as a result of the MUR power uprate. Therefore, the NRC staff concludes that the LAR is acceptable with respect to the SG tube material and program.

3.3.4.4 Flow-Accelerated Corrosion

3.3.4.4.1 Regulatory Evaluation

FAC is a corrosion mechanism that occurs in carbon steel components exposed to either single-phase or two-phase water flow. Components made from stainless steel are not affected by FAC and FAC is significantly reduced in components containing a small amount of chromium or molybdenum. The rates of material loss due to FAC depend on the system flow velocity, component geometry, fluid temperature, steam quality, oxygen content, and pH. During plant operation, it is not normally possible to maintain all of these parameters in a regime that minimizes FAC, therefore, loss of material by FAC can occur. The licensee stated that its FAC program is based on the "latest revision of the Electric Power Research Institute (EPRI) NSAC-202L, 'Recommendations for an Effective Flow-Accelerated Corrosion Program.'" At the time of the submittal of the LAR, the latest revision of NSAC-202L was Revision 4

(Reference 58). The NRC's acceptance criteria are based on the structural evaluation of the minimum acceptable wall thickness for the components undergoing degradation by FAC.

3.3.4.4.2 Licensee Description

In Section IV.1.E.iii, "Flow Accelerated Corrosion Program," of Enclosure 2 to the LAR, the licensee stated that there is an established FAC program at WBN 2 based on the latest revision of NSAC-202L. The licensee also stated that the EPRI CHECWORKS™ predictive software is used to provide a calculated estimated of component wear due to FAC. Changes in operating conditions due to the proposed MUR power uprate have been incorporated into the plant's CHECWORKS™ model. Additionally, Table IV.1.E-1 of Enclosure 2 to the LAR provides a wear rate analysis that assesses the impacts of the MUR power uprate on certain components at WBN, Unit 2. The licensee stated that the increase in wear rates due to the MUR power uprate is minor and that the existing FAC program is adequate to handle the increased wear rates.

3.3.4.4.3 Staff Evaluation and Conclusion

The NRC staff has 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 material loss so that repair or replacement of affected components can be made before reaching a critical thickness. The NRC staff finds the basis for the FAC program acceptable because it is based on Revision 4 of the EPRI FAC guidance. NSAC-202L has been endorsed in NRC guidance (e.g., the Generic Aging Lessons Learned Report for Subsequent License Renewal) as an acceptable basis to develop a FAC program. The NRC staff finds the increases in predicted FAC wear rates acceptable because they are relatively small (on the order of a few percent), which provides reasonable assurance that the licensee's FAC program will be able to account for these increases while planning inspections.

The NRC staff concludes that the licensee has adequately addressed the impact of changes in plant operating conditions on the FAC analysis. Additionally, the NRC staff has reasonable assurance that the licensee has demonstrated that the updated analyses will predict the loss of material by FAC and will ensure timely repair or replacement of affected components following implementation of the proposed MUR power uprate. The NRC staff has found that the FAC program will provide reasonable assurance that components susceptible to FAC will be managed appropriately after the MUR power uprate. Therefore, the NRC staff concludes that the proposed MUR power uprate is acceptable with respect to the impacts of FAC.

3.3.4.5 Rapidly Propagating Fatigue Cracks in Steam Generator Tubes

3.3.4.5.1 Regulatory Evaluation

Section IV.1.F of Attachment 1 to RIS 2002-03 states that licensees should address whether the effect of the power uprate on SG tube high cycle fatigue is consistent with NRC Bulletin 88-02.

Bulletin 88-02 was addressed to all licensees of Westinghouse designed reactors with SGs that utilize carbon steel support plates. The bulletin described an SGTR event at North Anna Unit 1 that was caused by rapidly propagating fatigue cracks due to high cycle fatigue. The bulletin described actions for the addressees to take to minimize the potential for a SGTR, such as the one that occurred at North Anna Power Station, Unit No. 1.

3.3.4.5.2 Licensee Description

In its LAR, the licensee stated that its response (Reference 59) and its supplement responses (References 60 and 61) to Bulletin 88-02 addressed the SG tube high cycle fatigue concerns. The licensee's response to Bulletin 88-02 described the actions that the licensee had implemented. These actions included evaluation of eddy current inspection data to determine penetration depth of antivibration bars, which would then be used in a thermal-hydraulic analysis to identify susceptible tubes and to address any susceptible tubes by permanent long-term corrective actions (e.g., stabilization, hardware modification, operational changes, etc.). The licensee stated in its supplemental responses that the corrective actions taken would permanently preclude rapidly propagating fatigue cracks and, therefore, certain monitoring actions were not necessary.

In its letter dated July 31, 2012, the licensee provided the results of the thermal-hydraulic analyses and details of permanent long-term corrective actions in response to Bulletin 88-02.

3.3.4.5.3 Staff Evaluation and Conclusion

The NRC staff evaluated the licensee responses to Bulletin 88-02, as well as the associated evaluation for tube vibration-induced fatigue, with respect to impacts from the proposed MUR power uprate. Because the previously submitted analyses bound the proposed MUR power uprate conditions, the NRC staff finds the impacts of the MUR power uprate on SG fatigue cracking acceptable. Additionally, because this information is consistent with the actions required in Bulletin 88-02, the staff finds that the licensee is consistent with the guidance in Section IV.1.F of Attachment 1 to RIS 2002-03.

The NRC staff reviewed the licensee's evaluation of the effect of the MUR power uprate on SG tube integrity with respect to SG tube high cycle fatigue concerns. The staff found that the licensee had addressed these concerns in its responses to Bulletin 88-02. Therefore, the NRC staff has reasonable assurance that the proposed MUR power uprate is acceptable with respect to SG tube high cycle fatigue.

3.3.4.6 Containment Coatings Program

3.3.4.6.1 Regulatory Evaluation

Protective coating systems (paints) provide a means for protecting the surfaces of facilities and equipment from corrosion and contamination from radionuclides and also provide wear protection during plant operation and maintenance activities. The NRC staff reviewed the protective coating systems used inside containment for their suitability for and stability under design basis LOCA conditions, considering temperature, pressure, radiation, and chemical effects on the ECCS.

The NRC staff performed this safety evaluation based on the following regulations and guidance:

Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," which provides quality assurance requirements for the design, fabrication, and construction of safety-related SSCs.

RG 1.54, Revision 0, “Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants” (Reference 62), which is discussed in WBN UFSAR Section 6.1.4, “Degree of Compliance with Regulatory Guide 1.54 for Paints and Coatings Inside Containment” (Reference 34).

Guidance for the NRC staff on specific review criteria is found in SRP Section 6.1.2, Revision 3, “Protective Coating Systems (Paints) – Organic Materials” (Reference 63).

3.3.4.6.2 Licensee Description

Section VII.6.B, “Containment Coatings Program,” of Enclosure 2 to the LAR states that the licensee evaluated the impacts of the proposed MUR power uprate on Service Level I, II, and III coatings as well as the Generic Safety Issue (GSI)-191 treatment of coatings in containment. The licensee concluded that because the DBA pressure, temperature, and dose analyses are bounding for the MUR power uprate conditions, Service Level I coatings that are currently DBA-qualified remain qualified. Additionally, the licensee stated that the GSI-191 treatment of coatings (i.e., quantity of failed coatings within the zone of influence for a given break) is not power dependent and, therefore, will not change with the MUR power uprates conditions. The licensee therefore concludes that no changes are needed to its coatings program.

3.3.4.6.3 Staff Evaluation and Conclusion

The NRC staff has reviewed the information provided by the licensee as well as the WBN UFSAR with regards to the containment coatings program. The staff has determined that the program is acceptable at the proposed MUR power uprate conditions as the proposed conditions in containment after a DBA LOCA due to the MUR power uprate are bounded by the current analyses. Because the proposed post-DBA LOCA conditions in containment are bounded by current analyses, the coating qualifications continue to bound the predicted conditions in containment after a DBA LOCA at the proposed MUR power uprate conditions. Therefore, the NRC staff has reasonable assurance that the coatings in containment will not be adversely impacted by the power uprate conditions and finds the MUR power uprate acceptable with respect to protective coatings. The NRC staff has also determined that the protective coatings continue to meet the requirements of 10 CFR Part 50, Appendix B, as well as Revision 0 of RG 1.54, with an acceptable alternative to ANSI N101.4-1972, “Quality Assurance for Protective Coatings Applied to Nuclear Facilities” (Reference 64), and the testing requirements of ANSI N101.2-1972, “Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities” (Reference 65).

3.4 ELECTRICAL EQUIPMENT DESIGN

3.4.1 Regulatory Evaluation

The licensee developed the LAR consistent with the guidelines in RIS 2002-03. The NRC staff performed this safety evaluation based on the following regulations:

10 CFR 50.49, “Environmental qualification of electric equipment important to safety for nuclear power plants,” which requires that licensees establish programs to qualify electric equipment important to safety.

10 CFR 50.63, “Loss of all alternating current power,” which requires that all nuclear plants have the capability to withstand a loss of all AC power (i.e., station blackout (SBO)) for a specified duration, and for recovery.

GDC 17, “Electric power systems,” which requires, in part, that an onsite electric power system and an offsite electric power system be provided with sufficient capacity and capability to permit functioning of SSCs important to safety. Conformance to GDC 17 is discussed in Section 3.1 of the WBN UFSAR.

3.4.2 Technical Evaluation

The proposed MUR power uprate would allow WBN, Unit 2 to operate at a 1.4 percent higher reactor thermal power level, increasing core power output from 3411 MWt to 3459 MWt. The electrical equipment design information is provided in Section V of Enclosure 2 to the LAR. The NRC staff notes that Sections V.1.D.1 – V.1.D.4 of Enclosure 2 to the LAR appear to be mislabeled as Sections VI.D.1 – VI.D.4 and their review is included here with the rest of LAR Enclosure 2, Section V. The NRC staff reviewed the licensee’s evaluation of the impact of the MUR power uprate on the following electrical systems/components:

- Electrical Power Systems
  - unit station service transformers (USSTs)
  - common station service transformers (CSSTs)
  - 6.9 kilovolt (kV) AC system switchgear and distribution equipment
  - 480 V AC system switchgear and distribution equipment
  - plant 120 V AC and 125 V direct current (dc) vital systems
  - emergency diesel generators (EDGs)
  - station blackout (SBO)
  - grid stability
  - environmental qualification (EQ) program

3.4.2.1 Electrical Power Systems

The Watts Bar Nuclear site is connected into a 500 kV transmission grid. The 500 kV switchyard is a double breaker - double bus configuration. Both units and five 500 kV transmission lines can be connected to either or both buses through a 500 kV breaker. Preferred power is supplied from the existing Watts Bar Hydro 161 kV switchyard over two radial lines located entirely on TVA property. The Watts Bar Hydro 161 kV switchyard is interconnected with the TVA power system through six 161 kV transmission lines and five hydroelectric generators. In its LAR, the licensee stated that a grid stability impact study was performed which determined that no additional interconnection facilities or system protection upgrades would be needed for the WBN, Unit 2 MUR power uprate. Further details of the grid stability impact study are reviewed in Section 3.4.2.4 below.

During normal operation, non-safety related station auxiliary power system loads are fed through the USSTs from the main generators, and from the 161 kV system through CSSTs A and B. Class 1E safety-related loads are fed from the 161 kV system via CSSTs C and D. During startup and normal shutdown, all auxiliary power is supplied from the 161 kV system through CSSTs A, B, C, and D. The licensee stated in Section V of Enclosure 2 to the LAR that the USSTs and CSSTs continue to have adequate capacity and capability for plant operation at the proposed MUR power uprate conditions and are bounded by the existing analysis and calculations of record for the plant.

The offsite 161 kV system, WBN, Unit 2's preferred power system, supplies Class 1E circuits and is normally fed via CSSTs C and D. CSSTs C and D are connected to 6.9 kV common switchgear C and D and then to the 6.9 kV shutdown boards. The licensee stated in Section V of Enclosure 2 to the LAR that the CSSTs continue to have adequate capacity and capability for plant operation at the proposed MUR power uprate conditions and are bounded by the existing analysis and calculations of record for the plant.

As described in WBN UFSAR Chapter 8, the onsite AC power system is a Class 1E system that consists of: (1) the standby AC power system and (2) the 120 V vital AC system (Reference 66). The safety function of the standby AC power system is to supply power to permit functioning of components and systems required to ensure that (1) fuel design limits and reactor coolant pressure boundary design conditions are not exceeded due to anticipated operational occurrences and (2) the core is cooled and vital functions are maintained in the event of a postulated accident in one unit and to safely shutdown the other unit, subject to loss of the preferred power system and subject to any single failure in the standby power system. The safety function of the 120 V vital AC system is to supply power continuously to reactor protection, instrumentation, and control systems; engineered safety features instrumentation and control systems; and other safety-related components and systems, subject to loss of offsite and standby AC power sources and any single failure within the vital AC system.

The licensee stated in its LAR that the analyses of record for the auxiliary power system (i.e., the onsite electrical power system) bound the resultant AC electrical load requirements due to the MUR power uprate. The existing load flow analysis for the plant already accounts for plant operation at MUR power uprate conditions. The licensee performed the analysis using the industry standard Electrical Transient Analyzer Program. The post-MUR power uprate pump brake horsepower are below the rated motor nameplate horsepower and acceptable margin remains available in the current AC electrical power analyses of record. The licensee also stated that the capability of the transmission system to maintain the post-trip voltage levels at the safety buses above the reset value of the degraded voltage relay on a steady-state basis has been verified.

In its LAR, the licensee stated that plant 120 V AC and 125 V dc vital systems are not impacted and continue to have adequate capacity and capability for plant operation with the MUR power uprate. The existing LEFM system loads do not change and are already reflected in existing plant calculations and are bounded by the existing analysis and calculations of record for the plant.

In its LAR, the licensee stated that the plant electrical power system and offsite power components required for electrical power conversion were evaluated at the proposed MUR power uprate conditions and determined to be acceptable including the main turbine-generator, isolated-phase bus, main power transformer, switchyard power circuit breakers and switches, system protection relaying, and grid interconnections. The licensee stated that the WBN, Unit 2 generator protection calculation was reviewed to evaluate the settings of the protective devices that could potentially be impacted by the MUR power uprate and that changes made by the power uprate have no impact to the existing settings on the protective relays. In its LAR, the licensee noted that one of the key inputs to the settings calculations is the machine ratings, which the licensee stated are not changing. The NRC staff notes that the additional anticipated

18 megawatt electric (MWe) main generator output increase does not challenge the main generator ratings and, therefore, does not challenge the protection equipment settings.

Based on the information provided in the LAR, the NRC staff determined that the system protective relaying can accommodate the MUR power uprate conditions. Based on its review, the NRC staff determined that the electrical power systems (including electrical power conversion systems) continue to have adequate capacity and capability for plant operation with the MUR power uprate and are bounded by the existing analyses and calculations of record for the plant.

The NRC staff reviewed the LAR and has determined that the AC power system load changes are minor and will not adversely impact the loadings and voltages of the normal and essential auxiliary electrical distribution systems. Therefore, the NRC staff concludes that the AC power system has adequate capacity to operate the plant equipment within its design to support implementation of the proposed MUR power uprate.

3.4.2.2 Emergency Diesel Generators

The WBN onsite standby AC power system is a safety-related system that supplies power to Units 1 and 2 for all essential-to-safety AC loads when normal AC power is interrupted. The onsite standby AC power system components include four 4.4 MW EDGs, 6.9 kV shutdown boards, 6.9 kV shutdown relay logic panels, 6.9 kV/480 V transformers, 480V shutdown boards, and motor control centers supplied by the 480 V shutdown boards.

In its LAR, the licensee stated that there are no load changes to the emergency bus loads supported by the EDGs due to the proposed MUR power uprate and that the existing accident analyses remain bounding. Therefore, the EDG system has adequate capacity and capability to power the essential-to-safety loads at MUR power uprate conditions.

Based on its review of the licensee's LAR, the NRC staff concludes that the analyses for the EDG system bound MUR power uprate conditions and that, therefore, the onsite power system will continue to meet the requirements of GDC 17.

3.4.2.3 Station Blackout

10 CFR 50.63, "Loss of all alternating current power," identifies the factors that must be considered in specifying the SBO duration and requires that the plant be capable of maintaining core cooling and appropriate containment integrity. As described in Section 8.3 of the WBN UFSAR, the licensee is required to support a four-hour coping duration (Reference 66). In the WBN UFSAR, the licensee stated that the systems credited for operation during the SBO coping period were evaluated and determined to be acceptable for the proposed MUR power uprate conditions.

The SBO coping strategy relies on 125 V and 250 V dc system's batteries, turbine-driven auxiliary feedwater (AFW) pumps, the condensate storage tank, and nitrogen gas cylinders. The nitrogen gas cylinders operate steam generator power-operated relief valves and auxiliary feedwater level control valves. The turbine-driven AFW pumps maintain steam generator levels by supplying water from the condensate storage tank. The RCS relies on natural circulation for the 4-hour coping time. The 125 V dc battery system provides control and instrumentation power for the turbine-driven AFW pump, RCS indication, emergency lighting, instrument power for AFW system level control, and power to attempt three starts of the EDG. The 250 V dc batteries provide power for various oil pump motors, the Technical Support Center inverter, battery boards, and switchyard power circuit breakers.

In the LAR, the licensee stated that the number of full valve strokes available, given the existing nitrogen supply, bounds the anticipated operation of the valves during an SBO under MUR power uprate conditions for the four-hour coping period. The licensee also stated that none of the electrical loads are reactor power dependent and, therefore, they are not impacted by the MUR power uprate. Regarding RCS flow and inventory, the licensee stated that the RCS was evaluated to ensure natural recirculation without makeup for the 4-hour coping time and RCS inventory was evaluated as part of compliance with NRC Order EA-12-049. The EA-12-049 compliance evaluation was performed at a power level that is equivalent to the proposed MUR power uprate. Finally, the licensee stated that the CST and AFW pump have adequate capacity to support the 4-hour coping time under MUR power uprate conditions.

Based on its review of the licensee's LAR, the NRC staff finds that the MUR power uprate will have no impact on the WBN, Unit 2 SBO coping duration. Therefore, the NRC staff concludes that WBN, Unit 2 will continue to meet the requirements of 10 CFR 50.63 upon implementation of the MUR power uprate.

3.4.2.4 Grid Stability

In Section V.1.D of Enclosure 2 to the LAR, the licensee stated that the proposed MUR power uprate is expected to produce approximately 17-18 MWe uplift in generator output. The licensee also stated that, at 100 percent MUR power uprate conditions, generator output is expected to be approximately 1240 MWe at nominal backpressure with a theoretical maximum of 1283 MWe with low winter backpressure based on PEPSE®[6] (Performance Evaluation of Power System Efficiencies) heat balances. The NRC staff found that the rated capability of the generator bounds the output at MUR power uprate conditions.

The licensee performed a grid stability impact study. The study addressed load flow analysis, short circuit analysis, transient stability, reactive power capability, and voltage control. The results were summarized in Sections VI.1.D.1 - VI.1.D.4 (mislabeled) of Enclosure 2 to the LAR. The licensee also provided a summary of the models used for the study and stated that the study determined that no additional interconnection facilities or system protection upgrades would be needed for the proposed MUR power uprate.

Based on its review of the licensee's LAR, the NRC staff concludes that the analyses for the grid stability study at WBN, Unit 2 demonstrate that existing interconnection facilities and system protections will not require any changes for the MUR power uprate conditions.

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[6] PEPSE is a registered trademark of Scientech, a business unit of Curtiss-Wright Flow Control Service Corporation.

3.4.2.5 EQ Parameters

3.4.2.5.1 Regulatory Evaluation

The NRC staff performed this safety evaluation based on the following regulations:

10 CFR 50.49(e)(1), which requires that the time-dependent temperature and pressure at the location of the electric equipment important to safety must be established for the most severe design basis accident during or following which this equipment is required to remain functional.

10 CFR 50.49(e)(4), which requires that the radiation environment must be based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional.

10 CFR 50.49(b)(2), which requires qualification of nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified in subparagraphs (b)(1)(i)(A) through (C) of 10 CFR 50.49 by the safety-related equipment.

3.4.2.5.2 Technical Evaluation

The NRC staff reviewed the LAR to ensure that the EQ of electrical equipment important to safety remained bounded as a result of the proposed MUR power uprate.

The licensee evaluated the impact of the proposed MUR power uprate on the areas/rooms/zones with regard to temperature, pressure, and radiation. The licensee noted that potential changes in ambient temperatures, system temperatures, system pressures, and potential accident external pressures (e.g., HELB) and accident temperatures were considered during its review. In Section II.1.D.iii of Enclosure 2 to the LAR, the licensee noted that its review included a review of both TVA EQ Program-level documentation and EQ calculations and environmental data drawings for specific components installed at WBN, Unit 2. The focus of the review was on the EQ parameters of temperature, pressure, and radiation, with respect to any potential parameter changes due to the MUR power uprate. The licensee's evaluation determined that no programmatic changes to the EQ Program are required due to the WBN, Unit 2 MUR power uprate.

The WBN, Unit 2 LEFM CheckPlus System is not within the scope of the WBN EQ Program because the LEFM CheckPlus System components and associated cabling are located in areas with a mild environment in WBN, Unit 2 and electrical equipment that is located in a mild environment is beyond the scope of 10 CFR 50.49.

The licensee's response to Section II.1 of RIS 2002-03 is provided in Table II.1-1, "WBN Unit 2 Analyses," of Enclosure 2 to the LAR. The licensee noted that the value of 3479 MWt corresponds to 102 percent of rated thermal power, which remains the bounding power level for the MUR power uprate conditions when uncertainty is applied. The licensee indicated that some of the analyses use higher power levels (e.g., 3480 MWt or 3565 MWt) because of rounding differences or to provide additional conservatism.

Regarding temperature, pressure, radiation, and humidity, the NRC staff reviewed the licensee's LAR to determine whether the licensee adequately addressed the impact of the proposed MUR power uprate on the EQ of electrical equipment inside and outside of containment (e.g., auxiliary building, main steam valve vault (MSVV), etc.). For the containment and the MSVV, the NRC staff confirmed that the current design basis analyses were performed at 102 percent of 3411 MWt (i.e., 3479 MWt), which bounds the MUR power uprate. Therefore, there is no impact on the EQ of electrical equipment with respect to temperature or pressure due to the MUR power uprate in these areas. However, the licensee did not provide adequate information for the NRC staff to confirm whether the existing accident analyses for all areas of the plant were performed at 102 percent RTP versus being limited to inside containment and the MSVV. Based on this, the staff requested the licensee to provide additional information to help determine whether electric equipment in other areas of WBN, Unit 2 will remain qualified due to the proposed MUR power uprate during normal operation and accident conditions (Reference 21). In its letter dated April 29, 2020, the licensee clarified that the conditions used in the WBN, Unit 2 EQ Program are based on accident analyses that bound the MUR power uprate thermal power level and that these conditions are applied to all areas of the plant and are not limited to inside containment and the MSVV (Reference 2). The NRC staff reviewed the additional information provided by the licensee and determined that there is no impact on the EQ of electrical equipment with respect to temperature, pressure, radiation, and humidity due to the MUR power uprate in all other areas of WBN, Unit 2.

In Section V.1.C, “Environmental qualification of electrical equipment,” of Enclosure 2 to the LAR, the licensee noted that the TVA EQ Program addresses safety-related electrical equipment within the scope of 10 CFR 50.49 for WBN, Unit 2. According to 10 CFR 50.49(b)(2), certain nonsafety-related electric equipment also needs to be considered for EQ. Based on its review of the LAR, it was not clear to the NRC staff that the licensee addressed nonsafety-related electric equipment whose failure in postulated environmental conditions could prevent satisfactory accomplishments of safety functions by the safety-related equipment. Based on this, the NRC staff requested the licensee to provide additional information to help determine that electric equipment in other areas will remain qualified due to the proposed MUR power uprate (Reference 21). In its letter dated April 29, 2020, the licensee noted that the analyses of record, which document compliance with 10 CFR 50.49(b)(2) for nonsafety-related electrical equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions performed by safety-related equipment, were reviewed and determined to not be impacted by the MUR power uprate conditions (Reference 2). The NRC staff reviewed the additional information provided by the licensee and determined that there is no impact on the licensee’s EQ Program with respect to nonsafety-related electric equipment due to the MUR power uprate. Based on its review of the information provided by the licensee, the NRC staff also confirmed that no areas within WBN, Unit 2 transition from mild to harsh due to the proposed MUR power uprate.

3.4.2.5.3 Technical Conclusion for EQ Parameters

Based on its review of the LAR and the supplemental information provided by the licensee, the NRC staff concludes that the MUR power uprate will not adversely impact the EQ of electrical equipment at WBN, Unit 2 since the existing qualification for the normal and accident conditions for electrical equipment inside and outside containment remain adequate and bound the MUR power uprate conditions. Therefore, the NRC staff finds that the proposed MUR power uprate will have no adverse impact on the WBN, Unit 2 EQ Program because it continues to comply with the requirements of 10 CFR 50.49.

3.4.3 Technical Conclusion Regarding Electrical Equipment Design

The NRC staff reviewed the licensee's technical evaluations described above and, based on that information, the NRC staff has determined that WBN, Unit 2 will continue to meet the requirements of 10 CFR 50.49, 10 CFR 50.63, and GDC 17. Therefore, the NRC staff concludes that the LAR is acceptable with respect to electrical equipment design evaluations.

3.5 SYSTEM DESIGN

The licensee evaluated the effect of the proposed MUR power uprate on the plant systems and discussed this evaluation in the LAR. The NRC staff review below covers the impact of the MUR power uprate on the following major plant systems and events:

- NSSS interface systems,
- containment systems,
- safety-related cooling water systems,
- radioactive waste systems,
- engineered safety features heating, ventilation, and air conditioning systems,
- flooding analyses, and
- high energy line breaks.

The NRC staff conducted its review to verify that the licensee's analyses bound the proposed plant operation at the proposed MUR power level of 3459 MWt and that the results of the licensee's analyses related to the areas under review continue to meet the applicable acceptance criteria following implementation of the proposed MUR power uprate.

3.5.1 NSSS Interface Systems

The NSSS interface systems include the:

- main steam system
- main turbine-generator
- condensate and feedwater systems
- auxiliary feedwater
- condenser circulating water
- extraction steam system
- heater and moisture separator drains
- process sampling system
- safety injection system
- SG blowdown system

The licensee indicated that there will be an expected increase in main steam flow at the uprated power (i.e., 3459 MWt) compared to the current licensed thermal power (CLTP). This increase in steam and mass flow is seen in the remainder of the BOP systems (e.g., condensate, feedwater, extraction steam). The licensee performed an evaluation of the structural integrity of these systems and determined that the BOP piping systems' design bases (pressure and temperature) remain bounded by the analysis and design of record after the MUR power uprate. The BOP piping design pressure and temperature limits do not change as a result of the MUR power uprate; therefore, the NRC staff finds that the BOP piping is acceptable for MUR power uprate conditions.

The licensee performed an evaluation of the conditions of the NSSS interface systems and indicated that these systems are not expected to see any significant change in operating conditions and that the systems will continue to meet their design basis under MUR power uprate conditions.

The NRC staff reviewed the information and evaluations performed by the licensee showing that the design of the NSSS interface systems at the increased power level is bounded by existing plant analyses and, based on this information, determined that they are acceptable. The NRC staff finds that the MUR power uprate will not challenge the NSSS interface systems. Therefore, the NRC staff concludes that the LAR is acceptable regarding NSSS interface systems.

3.5.2 Containment Systems

The containment systems are described in WBN UFSAR Section 6.2 (Reference 34). These systems are designed to limit offsite releases following a design basis accident. These systems include the containment spray, containment isolation system, containment combustible gas control system, emergency gas treatment system, and ice condenser refrigeration system. The licensee performed an evaluation of the impact of the MUR power uprate on the containment systems and determined that the existing analyses remain valid. The licensee also evaluated the Containment Leakage Rate Testing Program discussed in WBN, Unit 2 TS 5.7.2.19 and determined that the MUR power uprate does not have any impact on the programmatic aspects of that program.

The NRC staff reviewed the following areas of containment design and analysis for the proposed MUR power uprate: long-term LOCA containment response, short-term LOCA containment response, containment response to main-steam line break inside containment, and the impact of MUR on combustible gas control. The WBN, Unit 2 short and long-term LOCA peak containment pressure analysis is documented in UFSAR Sections 6.2.1.3.1 through 6.2.1.3.9. The maximum reverse pressure differential analysis is documented in UFSAR Section 6.1.2.3.11. These analyses are performed to demonstrate that peak containment pressures and temperatures are acceptable and to ensure that the pressure and temperature profiles assumed in the environmental qualification analyses are acceptable. The licensee reviewed the containment analyses for the expected conditions following the MUR power uprate and determined that the existing analyses bound the expected conditions following the MUR power uprate.

The NRC staff also evaluated the impact of the MUR power uprate on the Containment Leakage Rate Testing Program. The staff identified that the associated analyses assumed an initial power level that bounds MUR power uprate conditions; therefore, the MUR power uprate does not impact the Containment Leakage Rate Testing Program.

The licensee's existing analysis was performed using a thermal core power that bounds the MUR power uprate level. The long-term LOCA containment performance analyses assumed a power level of 3479 MWt, which is 102 percent of the current rated thermal power, thus the analyses remain bounding for the MUR power uprate since the uprate results in a less than 2 percent increase in power. The short-term LOCA mass and energy release used in the analyses of the containment subcompartment response will not be significantly impacted by the MUR power uprate and thus remain acceptable with respect to the short-term LOCA analyses. The steam line break analysis was based on a power level of 3475 MWt, which is also greater than the MUR power uprate power level of 3459 MWt and, therefore, the containment response remains conservative and bounded for plant operation at the MUR power uprate power level. The NRC staff reviewed the information and evaluations performed by the licensee showing that the containment systems existing analyses remain valid and, based on this, the NRC staff concludes that the LAR is acceptable regarding containment systems.

3.5.3 Safety-Related Cooling Water Systems

The safety-related cooling water systems include the component cooling system, the nuclear service water system, the ultimate heat sink (UHS), residual heat removal (RHR), and the spent fuel pool cooling and cleanup system (SFPCCS).

3.5.3.1 Component Cooling System

The component cooling system (CCS) is described in Section 9.2.2 of the WBN UFSAR (Reference 42). The CCS provides sufficient cooling capacity to fulfill all system requirements under normal and accident conditions. The licensee evaluated the CCS to confirm that the heat removal capabilities are sufficient to satisfy the MUR power uprate heat removal requirements during normal plant operations, refueling, shutdown, and accident cooldown conditions. The licensee stated that the existing design analysis bounds operation under MUR power uprate conditions.

The NRC staff has reviewed the information provided in the LAR regarding the CCS and determined that the component cooling system will perform acceptably upon implementation of the MUR power uprate.

3.5.3.2 Nuclear Service Water System

The essential raw cooling water system (ERCW) is described in WBN UFSAR Section 9.2.1. The licensee stated that the ERCW provides cooling water for various heat exchangers during all phases of station operation. The licensee reviewed the system applications and indicated no change in requirements following the MUR power uprate. The design analyses which credit ERCW bound operation under MUR power uprate conditions.

The NRC staff has reviewed the licensee’s analysis of the impact of the MUR power uprate on the ERCW system and has determined that the ERCW system will continue to perform its intended function upon implementation of the MUR power uprate.

3.5.3.3 Ultimate Heat Sink

The UHS is described in WBN UFSAR Section 9.2.5. The licensee indicated that the UHS for WBN is the Tennessee River and the associated dams near the station and the plant intake channel. The UHS provides at least 30 days of cooling water to dissipate the waste heat rejected during a unit LOCA plus a unit cooldown. The licensee evaluated the expected conditions following the MUR power uprate and determined that the MUR does not impact the design basis heat loads to the UHS or change the UHS flow rates or pumping requirements.

The NRC staff has reviewed the information provided in the LAR regarding the impact of the MUR power uprate on the UHS and determined that the UHS will continue to perform its intended safety function upon implementation of the MUR power uprate.

3.5.3.4 Residual Heat Removal

The RHR heat exchanger and RHR pump seal water heat exchanger are cooled by the CCS. The licensee indicated that the current RHR cooldown analysis of record assumed a bounding power level of 3459 MWt, which supports operation at the MUR power level and that no changes to the system are required. Therefore, there is no impact to this system due to the MUR power uprate.

The NRC staff has reviewed the information provided in the LAR regarding the impact of the MUR power uprate on RHR heat exchanger and RHR pump seal water heat exchanger and determined that because the analysis of record was performed at a higher power level than the MUR power level, the components will continue to perform their safety function upon implementation of the MUR power uprate.

3.5.3.5 Spent Fuel Pool Cooling and Cleanup System

The SFPCCS is described in WBN UFSAR Section 9.1.3. The principal function of the SFPCCS is to provide storage and cooling of the spent fuel. The licensee indicated that the analysis for the SFP is performed to a bounding design basis heat load, which is used to determine acceptable offload times for each cycle to ensure that the heat load limits are not exceeded. The LAR indicated that the licensee re-evaluates the analysis for fuel pool operation for each offload. The analyzed design basis heat load will bound the post-MUR power uprate heat loads based on the selected offload delay time, in accordance with plant procedures. Therefore, the system will continue to perform its design functions of spent fuel decay heat removal after the MUR power uprate.

The NRC staff has reviewed the information provided in the LAR and determined that the implementation of the proposed MUR power uprate will not result in a change to the operation of the SFP storage and cooling system. Therefore, the NRC staff has determined that the SFPCCS will continue to perform its safety function upon implementation of the MUR power uprate.

3.5.3.6 NRC Staff Conclusion Regarding Safety-Related Cooling Water Systems

The NRC staff has reviewed the licensee's evaluation of safety-related cooling water systems. Based upon the analyses provided that show that these systems were evaluated for 102 percent RTP, the NRC staff concludes that there is reasonable assurance that the systems will perform acceptably after implementation of the MUR power uprate.

3.5.4 Radioactive waste systems

The radioactive waste management systems are described in WBN UFSAR Chapter 11 (Reference 67). The licensee stated that these systems provide the means to sample, collect, process, temporarily hold, and discharge, as necessary, gaseous and liquid low-level effluents generated during normal operation. Additionally, Section 11.1 "Source Terms," of the WBN UFSAR indicates that the radioactivity in waste management systems and components is determined using ANSI/ANS-18.1-1984 (Reference 68). The core thermal power is used to calculate the site source term and expected gaseous and liquid releases including tank ruptures. The licensee indicated that the core thermal power used to perform these evaluations bounds the core thermal power following the MUR power uprate. The systems will continue to maintain normal offsite doses within the requirements of 10 CFR Part 50, Appendix I.

The solid waste system is described in Section 11.5 of the WBN UFSAR. This system is designed to contain solid radioactive waste and support preparation for shipment to an offsite disposal facility. The licensee indicated that the solid waste system is a shared system between both units and was previously evaluated for MUR power uprate conditions as part of the WBN, Unit 1 MUR power uprate. Therefore, the WBN solid waste system remains acceptable for MUR power uprate conditions.

The NRC staff has reviewed the information provided in the LAR regarding the impact of the MUR power uprate on the radioactive waste management systems and determined that the systems will continue to perform their intended function upon implementation of the MUR power uprate.

3.5.5 Engineered Safety Features Heating, Ventilation, and Air Conditioning Systems

The control building heating, ventilation, and air conditioning (HVAC) and air cleanup system is described in WBN UFSAR Section 9.4.1. The Control Building HVAC and air cleanup system is designed to maintain the temperature and humidity in the building for personnel comfort, protection, and operation of plant controls, and to provide safe, uninterrupted occupancy of the main control room habitability zone during normal, accident, and post-accident recovery conditions. The auxiliary building ventilating system is described in UFSAR Section 9.4.3. The auxiliary building essential safety features equipment coolers are described in UFSAR Section 9.4.5.3 and the fuel handling area ventilation system, a subsystem of the auxiliary building HVAC system, is described in UFSAR Section 9.4.2. The diesel generator building ventilation system is described in UFSAR Section 9.4.5.2. The licensee indicated that the diesel generator building HVAC is designed to provide ventilation to the diesel generator building to maintain the required environmental conditions for safety-related equipment and to prevent hydrogen buildup in the battery area during normal operation and design basis event conditions. The licensee evaluated the expected conditions following the MUR power uprate and stated that these HVAC systems are not impacted by the MUR power uprate.

The reactor building purge ventilating system and containment air cooling system are described in WBN UFSAR Sections 9.4.6 and 9.4.7, respectively. The licensee indicated that the reactor building purge ventilating system is designed to maintain the environment in the primary containment and Shield Building annulus within acceptable limits and to provide a filtration path for any through-duct outleakage from the primary containment. The licensee indicated that the safety-related reactor building purge ventilating system isolation functions are not affected by the MUR power uprate.

The licensee evaluated the reactor building HVAC systems and determined that the systems will not experience any increase in heat loads or temperature with MUR power uprate conditions, except the areas that house the main steam and feedwater piping. These areas will see an increase in temperatures but will remain bounded by existing design.

The NRC staff reviewed the information presented by the licensee showing that the HVAC systems' existing analyses remain valid and, based on this, the NRC staff finds that the HVAC systems will be adequate for the MUR uprated power level.

3.5.6 Flooding Analyses

In Section II.1.D.iii of Enclosure 2 to the LAR, the licensee stated that the plant grade elevation at WBN can be exceeded by large rainfall and seismically induced dam failure floods. Sections 2.4.14 (Reference 69), 3.4.1 (Reference 70), 3.8.1, and 3.8.4 (Reference 71) of the WBN UFSAR describe the plant design features and procedures used to provide assurance that WBN can be safely shut down and maintained in these extreme flood conditions. The licensee also stated that the MUR power uprate will not affect the external flooding sources or the protective structural design features. The NRC staff confirmed that the MUR power uprate will have no impact on plant design features and procedures used to safely shut down and maintain the plant during extreme flood conditions. Therefore, the NRC staff finds that the power uprate will not affect the flooding design basis of the plant.

The licensee also addressed internal flooding of safety-related structure HELBs and MELBs. The concern of flooding through HELBs is based on the rate of the high-energy system's blowdown. Aside from the MFW system, none of these rates increase as a result of the MUR power uprate and, therefore, the NRC staff finds that the effects of the MUR power uprate are acceptable for these systems. Due to the MUR power uprate's effect on the feedwater conditions, it was determined that a postulated break in the feedwater line would result in an increase in flow from the break. The licensee stated that the calculated flood levels have approximately 50 percent margin to the maximum acceptable level which will not be exceeded due to the increased flow rates. Additionally, the plant has implemented multiple defense-in-depth measures to identify and track flaws prior to fully breaking as well as measures to mitigate leaks as they are identified. Therefore, the NRC staff finds that the HELB analysis remains valid under the proposed MUR power uprate conditions and that the increased flow rates for a feedwater line break will not result in an unacceptable decrease in level of quality and safety.

The concern for flooding through MELBs and flooding of the turbine building through the condenser circulating water (CCW) system are based on the release rates due to the maximum operating pressure that the systems experience. Internal flooding caused by pipe failures is discussed in Section 3.6A.2.1.4 of the WBN UFSAR (Reference 70). Internal flooding of the turbine building caused by failure of the CCW system is addressed in Section 10.4.5 of WBN UFSAR (Reference 53). None of the moderate energy systems or the CCW system will experience an increase in operating pressure due to the implementation of the MUR power uprate. Therefore, the NRC staff finds that the release rates from MELBs or a CCW break will not be impacted and will remain bounded by the current analyses.

3.5.7 High Energy Line Breaks

The licensee evaluated the consequences of an HELB inside the containment building and the turbine building with respect to impact on safety-related equipment. HELBs are analyzed for piping for which the maximum operating pressure exceeds 275 psig and the maximum operating temperature equals or exceeds 200 °F. The licensee's evaluation concluded that no new lines are added, no break locations are changed, and no results change to the mass and energy blowdown from any postulated break. Therefore, the licensee finds that the MUR power uprate has no impact on the HELB analyses that were originally performed and that, therefore, the MUR power uprate is bounded by the existing HELB analyses for WBN, Unit 2.

The NRC staff has reviewed the information provided in the LAR regarding the impact of the MUR power uprate on the HELB and determined that the HELB analyses remain acceptable.

3.5.8 Technical Conclusion Regarding Plant Systems

The NRC staff has reviewed the licensee's safety analyses of the impact of implementation of the proposed MUR power uprate on plant systems. The NRC staff has determined that the results of the licensee's analyses related to these areas would continue to meet the applicable acceptance criteria following implementation of the MUR power uprate. Therefore, the NRC staff concludes that the LAR is acceptable regarding the impact of changes to plant systems.

3.6 <u>OTHER CONSIDERATIONS</u>

3.6.1 Human Factors

3.6.1.1 Regulatory Evaluation

The NRC staff performed this safety evaluation based on the following regulations and guidance:

GDC 19, "Control room," which states that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Equipment at appropriate locations outside the control room shall be provided: (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

10 CFR 50.120, "Training and qualification of nuclear power plant personnel," which requires that the licensee establish, implement, and maintain a training program.

SRP Section 13.5.2.1, Revision 2, "Operating and Emergency Operating Procedures," which provides the methodology for the NRC staff's review of operating procedures that will be used by the operating organization to ensure that routine operating, off-normal, and emergency activities are conducted in a safe manner (Reference 72).

3.6.1.2 Technical Evaluation

3.6.1.2.1 Operator Actions

In Section VII.1 of Enclosure 2 to the LAR, the licensee stated that there are no new time critical operator actions associated with the proposed MUR power uprate because there are no changes to the associated analyses of record, shown in Section II, "Accidents and Transients for which the Existing Analyses of Record Bound Plant Operation at the Proposed Uprated Power Level," of Enclosure 2.

The NRC staff finds that the statements provided by the licensee conform to Section VII.1 of Attachment 1 to RIS-2002-03 and that the proposed MUR power uprate will not adversely impact the licensee-identified operator actions, including the time available for operator actions.

3.6.1.2.2 Emergency and Abnormal Operating Procedures

In Section VII.2 of Enclosure 2 to the LAR, the licensee stated that the proposed MUR power uprate will be implemented under the TVA design change process. The TVA procedures provide the controls relevant to identifying impacts to procedures, controls, displays, alarms, the ICS and other operator interfaces, the simulator, and training. The design change process ensures that any impacted emergency and abnormal operating procedures are revised prior to the implementation of the MUR power uprate. However, the proposed request does not involve any changes to the design or functional requirements of the safety and support system.

The NRC staff finds that the statements provided by the licensee conform to Section VII.2.A of Attachment 1 to RIS-2002-03 and concludes that the proposed changes to the emergency and abnormal operating procedures will not adversely affect defense-in-depth or safety margins.

3.6.1.2.3 Changes to Control Room Controls, Displays, and Alarms

In Section VII.2.B of Enclosure 2 to the LAR, the licensee stated that only minor modifications to control room controls, displays, and alarms are necessary (e.g., software modification that redefines the new 100 percent rated thermal power). As part of the TVA design change process, instrument loop scaling updates for the MUR service conditions will be implemented for the following functions: SG MFW flow, temperature, and pressure; MFW header pressure and flow; pressurizer pressure; and turbine impulse pressure.

The licensee has described the LEFM CheckPlus System stating that the system status indication already exists that alerts the operators both visually and audibly when there is a problem. Validated LEFM data including calculated results, status, and signal process information is sent to the ICS screen in the control room. Any status other than NORMAL will generate the alarm; the other statuses being ALERT (indicating that the system is in MAINTENANCE mode) or FAIL. The licensee states that this is consistent with alarms for various other ICS inputs. As part of the TVA design change process, the alarm will be updated, as necessary, to support the use of the LEFM for continuous calorimetric power determination to support the MUR power uprate.

The NRC staff finds that the statements provided by the licensee conform to Section VII.2.B of Attachment 1 to RIS-2002-03 and that the proposed changes to the control room controls, displays, and alarms will not adversely affect defense-in-depth or safety margins.

3.6.1.2.4 Control Room Plant Reference Simulator

In Section VII.2.C of Enclosure 2 to the LAR, the licensee stated that a review of the plant simulator will be conducted, and necessary changes made, under the TVA design change process. However, as stated in previous sections of this evaluation, there are no time critical operator actions and only minor modifications to the software.

The NRC staff finds that the statements provided by the licensee conform to Section VII.2.C of Attachment 1 to RIS-2002-03 and that the proposed changes to the control room plant reference simulator will not adversely affect defense-in-depth or safety margins.

3.6.1.2.5 Operator Training

In Section VII.2.C of Enclosure 2 to the LAR, the licensee stated that operator training on the plant changes required to support the MUR power uprate will be completed prior to MUR power uprate implementation. This is part of the licensee's normal process for implementing license amendment requests. Furthermore, training on operating and maintenance of the Caldon LEFM CheckPlus System will be updated, as necessary, prior to implementation.

The NRC staff finds that the statements provided by the licensee conform to Section VII.2.C of Attachment 1 to RIS-2002-03 and that the changes to the operator training program will not adversely affect defense-in-depth or safety margins.

3.6.1.2.6 Modifications

In Section VII.3 of Enclosure 2 to the LAR, the licensee stated that the changes/modifications to the simulator and the associated manuals and instructional materials will be implemented in accordance with the TVA design process to capture the plant changes resulting from the MUR power uprate.

The NRC staff finds that the statements provided by the licensee conform to Section VII.3 of Attachment 1 to RIS-2002-03 and that the applicable modifications will be completed prior to implementation of the MUR power uprate.

3.6.1.2.7 Temporary Operation Above Licensed Full Power Level

In Section VII.4 of Enclosure 2 to the LAR, the licensee provided the following statement:

> Operating procedures have been reviewed and required changes will be documented and implemented as part of the TVA design change process including the procedure related to temporary operation above full steady-state licensed power levels.

The NRC staff finds that the statement provided by the licensee conforms to Section VII.4 of Attachment 1 to RIS-2002-03 and that the licensee will revise existing plant operating procedures related to temporary operation above full steady-state licensed power levels to reduce the magnitude of the allowed deviation from the licensed power level.

3.6.1.3 Conclusion

The NRC staff has completed its human factors review of the LAR and has determined that the licensee has adequately considered, or will consider, the impact of the MUR power uprate on operator actions, emergency and abnormal operating procedures, control room components, the plant reference simulator, and operator training programs. The NRC staff has determined that the results of the licensee's review of these areas would continue to meet the applicable requirements in GDC 19 and 10 CFR 50.120 and the acceptance criteria in SRP Section 13.5.2.1, and are in conformance with Section VII, Items 1 through 4, of Attachment 1 to RIS 2002-03. Therefore, the NRC staff concludes that the MUR power uprate is acceptable with respect to human factors.

3.6.2 Radiological Dose Assessment

The NRC staff reviewed the impact of the proposed MUR power uprate on analyzed DBA radiological consequences.

3.6.2.1 Regulatory Evaluation

3.6.2.1.1 Radiation Protection

The NRC's acceptance criteria for normal occupational and public doses are based on: (1) 10 CFR Part 20, insofar as it establishes requirements for radioactivity in liquid and gaseous effluents released to unrestricted areas; (2) 10 CFR Part 50, Appendix I, insofar as it establishes numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" criterion; (3) GDC 60, insofar as it requires that the plant design include means to control suitably the release of radioactive effluents; and (4) Item II.B.2 of NUREG-0737, insofar as it relates to plant shielding for spaces/systems that may be used in post-accident operations (Reference 73). Specific review criteria are contained in SRP Section 11.1 (Reference 67).

3.6.2.1.2 Design-Basis Accident

The NRC staff's review of the licensee's analysis of radiological dose consequences follows the guidance of RIS 2002-03, which recommends that, for efficiency of review, licensees requesting an MUR power uprate identify existing DBA analyses of record, which bound plant operation at the proposed uprated power level. For any existing DBA analyses of records that do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the reanalysis.

The NRC staff performed this safety evaluation based on the following regulations and guidance:

10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distances," which states, in part:

> (a) As an aid in evaluating a proposed site, an applicant should assume a fission produce release[7] from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following:
>
> (1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in

[7] The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

> excess of 25 rem [Roentgen equivalent man][8] or a total radiation dose in excess of 300 rem[8] to the thyroid from iodine exposure.
>
> (2) A low population zone of such size that an individual located at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

10 CFR 50.67, “Accident source term,” which states, in part:

> (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv [sievert] (25 rem)[9] total effective dose equivalent (TEDE).
>
> (ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).
>
> (iii) Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for the duration of the accident.

GDC 19, “Control room,” which states:

> A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident. Equipment at appropriate locations

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[8] The whole body dose of 25 rem referred to above corresponds numerically to the once in a lifetime accidental or emergency dose for radiation workers which, according to NCRP [National Council on Radiation Protection & Measurements] recommendations may be disregarded in the determination of their radiation exposure status (see NBS [National Bureau of Standards] Handbook 69 dated June 5, 1959). However, neither its use nor that of the 300 rem value for thyroid exposure as set forth in these site criteria guides are intended to imply that these numbers constitute acceptable limits for emergency doses to the public under accident conditions. Rather, this 25 rem whole body value and the 300 rem thyroid value have been set forth in these guides as reference values, which can be used in the evaluation of reactor sites with respect to potential reactor accidents of exceedingly low probability of occurrence, and low risk of public exposure to radiation.

[9] The use of 0.25 Sv (25 rem) TEDE is not intended to imply that this value constitutes an acceptable limit for emergency doses to the public under accident conditions. Rather, this 0.25 Sv (25 rem) TEDE value has been stated in this section as a reference value, which can be used in the evaluation of proposed design basis changes with respect to potential reactor accidents of exceedingly low probability of occurrence and low risk of public exposure to radiation.

> outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.
>
> Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under [10 CFR] 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in [10 CFR] 50.2 for the duration of the accident.

3.6.2.1.3 Other Regulatory Guidance:

Regulatory Guide 1.24, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Gas Storage Tank Failure,” March 1972, which provides the methodology for analyzing the radiological consequences of a gas storage tank failure (Reference 74).

Regulatory Guide 1.4, “Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors,” June 1974, which provides the methodology for analyzing the radiological consequences of the design basis loss of coolant accident to demonstrate compliance with the offsite dose consequence guidelines of 10 CFR Part 100 (Reference 75).

Regulatory Guide 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000, which provides the methodology for analyzing the radiological consequences of several DBAs to show compliance with 10 CFR 50.67 (Reference 76). It also provides guidance to licensees on acceptable application of alternate source term (AST) submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

RIS 2002-03, which recommends that to improve the efficiency of the NRC staff’s review, licensees requesting an MUR power uprate should identify existing DBA analyses of record that bound plant operation at the proposed uprated power level. For any existing DBA analyses of record that do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the re-analysis.

3.6.2.2 Technical Evaluation

3.6.2.2.1 Radiation Protection

3.6.2.2.1.1 Radiological Effluents

The NRC staff reviewed the radioactive source term associated with the proposed MUR power uprate to ensure the adequacy of the sources of radioactivity used by the licensee as input to calculations to verify that the radioactive waste management systems have adequate capacity for the treatment of radioactive liquid and gaseous wastes. The NRC staff's review included the parameters used to determine: (1) the concentration of each radionuclide in the reactor coolant; (2) the fraction of fission product activity released to the reactor coolant; (3) concentrations of all radionuclides other than fission products in the reactor coolant; (4) leakage rates and associated fluid activity of all potentially radioactive water and steam systems; and (5) potential sources of radioactive materials in effluents that are not considered in the plant's UFSAR related to liquid waste management systems and gaseous waste management systems.

The core isotopic inventory is a function of the core power level. The reactor coolant isotopic activity concentration is a function of the core power level, the migration of radionuclides from the fuel, radioactive decay, and the removal of radioactive material by coolant purification systems. Radiation sources in the reactor coolant include activation products, activated corrosion products, and fission products. During reactor operation, some stable isotopes in the coolant passing through the core become radioactive (activated) as a result of nuclear reactions. For example, the non-radioactive isotope oxygen-16 (O-16) is activated to become radioactive nitrogen-16 (N-16) by a neutron-proton reaction as it passes through the neutron-rich core at power. Under MUR conditions there is a necessary increase in the steam flow. This increase in steam flow tends to balance the increase in the activation products in the reactor coolant resulting in no significant increase in the coolant concentrations. Another source of activity in the reactor coolant is from the activation of metallic corrosion products contained in the coolant as it passes through the reactor core. Under the MUR conditions, an increase in steam flow will lead to an increase in the activation rate of metallic corrosion products resulting in an increase of corrosion products present in the reactor coolant.

The licensee reviewed the radiological effects for the proposed MUR power uprate using current licensing basis methodologies to verify that expected coolant concentrations at the MUR power levels will be bounded by the current licensing basis values. Under the MUR power uprate conditions, the activation rate in the reactor region increases with power and the filter efficiency of the condensate demineralizers may decrease. The net result may be an increase in the activated corrosion product production. However, total activated corrosion product activity levels in the reactor water remain less than the design basis activated corrosion product activity. Therefore, no change is required in the design basis activated corrosion product concentrations for the MUR power uprate.

As discussed by the licensee, during normal operations, the controls for the release rates of radwaste systems do not change with operating power. Thus, no impact on routine releases is anticipated due to the MUR power uprate. Actual, measured doses due to normal effluent associated with the reactor operating at CLTP are documented in the annual radioactive effluent release reports. A review of historical liquid and gaseous release data indicates that resultant doses are a small fraction of annual limits. The effluent doses are determined in accordance with the offsite dose calculation manual which is a licensee-controlled document required under the Administrative Controls section of the Technical Specifications. The offsite dose calculation manual methodologies ensure that doses to the public remain within regulatory dose limits and are as low as is reasonably achievable (ALARA). In addition, the MUR power uprate will not result in changes to the offsite dose calculation manual. Therefore, the NRC staff concludes that the proposed license amendment is acceptable with respect to the effect of the power uprate on radiological effluents from radwaste systems.

3.6.2.2.1.2 Individual or Cumulative Occupational Radiation Exposure

WBN, Unit 2 was designed with sufficient margin for higher-than-expected radiation sources. During normal and post-accident conditions, radiation levels in most areas of the plant increase by no more than the percentage increase in power level. The licensee reviewed the radiological effects for the proposed MUR power uprate using current licensing basis methodologies to verify that expected coolant concentrations at the MUR power levels will be bounded by the existing analyses of record at 102 percent of the CLTP. Due to the design of the shielding and containment surrounding the reactor vessel, and since the reactor vessel is inaccessible to plant personnel during operation, an increase in the radiation sources in the reactor core over the original licensed thermal power level will have no effect on occupational worker personnel doses during power operations. Similarly, the radiation shielding provided in the balance of plant is conservatively sized such that the increased source terms discussed above are not expected to significantly increase the dose rates in the normally occupied areas of the plant. In addition, occupational exposure is controlled by the plant radiation protection program and is maintained within limits required by regulations. Thus, the increase in radiation levels does not affect radiation zoning or shielding in the various areas of the plant because it is offset by conservatism in the design, source terms, and analytical techniques. Therefore, no change is required in the design basis radiation protection design features for the MUR power uprate.

The current ALARA program practices at WBN, Unit 2 (e.g., work planning, source term minimization, etc.), coupled with existing radiation exposure procedural controls, will be able to compensate for the anticipated increases in dose rates associated with the proposed MUR power uprate. Thus, the increased radiation sources resulting from the MUR power uprate, as discussed above, will not adversely impact the licensee's ability to maintain doses resulting from plant operation within the applicable limits in 10 CFR Part 20 and ALARA. Therefore, the NRC staff concludes that the proposed license amendment is acceptable with respect to the effect of the power uprate on individual or cumulative occupational radiation exposure.

3.6.2.2.2 Design Basis Accidents

The NRC staff reviewed the regulatory and technical analyses performed by the licensee in support of its proposed MUR power uprate license amendment, as they relate to the radiological consequences of DBA analyses. RIS 2002-03 recommends that to improve the efficiency of the NRC staff's review, licensees requesting an MUR power uprate should first identify existing DBA analyses of record that bound plant operation at the proposed uprated power level. Secondly, for any existing DBA analyses of record that do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the reanalysis. Information regarding these analyses was provided by the licensee in Enclosure 2 to the LAR, which is consistent with RIS 2002-03. The findings of this safety evaluation are based on the descriptions and results of the licensee's analyses and other supporting information docketed by the licensee.

In Enclosure 2 to the LAR, the licensee stated that the WBN accident and safety analyses, including LOCA events, that are addressed in Chapter 15 of the WBN UFSAR were performed to support the issuance of Facility OL No. NPF-96 for WBN, Unit 2 on October 22, 2015

(Reference 77). The licensee stated that the results and conclusions reported in the WBN UFSAR, which were approved by the NRC in License Condition 2.A of the OL, remain applicable at the proposed MUR power uprate level. The licensee also stated that the NRC approval is documented in NUREG-0847, “Safety Evaluation Report Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2,” as supplemented through Supplement 29 (available at https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0847/).

In its LAR, the licensee discussed each analysis in support of the proposed MUR power uprate, including the assumed core power level in each analysis and whether the analysis remains bounding for the MUR power uprate. As previously discussed, the MUR power uprate would increase the WBN, Unit 2 authorized core power level from 3411 MWt to 3459 MWt, which is an increase of approximately 1.4 percent rated thermal power, based on the use of the LEFM CheckPlus System. The licensee verified that evaluations of systems inside containment and in the main steam valve vault are based on a value of 3479 MWt, which corresponds to 102 percent of the authorized core power level of 3411 MWt and continues to bound the power level for the MUR power uprate when uncertainty is applied. The licensee stated that some analyses apply higher power levels (e.g., 3480 MWt or 3565 MWt) due to rounding differences or to provide additional conservatism.

The NRC staff reviewed the impact of the proposed 1.4 percent MUR power uprate on DBA radiological consequence analyses, as documented in Chapter 15 of the WBN UFSAR. The NRC staff confirmed that the current licensing basis dose consequence analyses performed at 3479 MWt remain bounding for the proposed MUR uprated power level of 3459 MWt with a margin that is within the assumed uncertainty associated with advanced flow measurement techniques, including use of the LEFM CheckPlus System credited by the licensee. Specific DBA analyses identified by the licensee include:

- UFSAR Section 15.3.5, “Waste Gas Decay Tank Rupture” (LAR Section II.1.D.iii.20);
- UFSAR Section 15.4.2.1, “Major Rupture of a Main Steam Line” (LAR Section II.1.D.iii.23);
- UFSAR Section 15.4.3, “Steam Generator Tube Rupture” (LAR Section II.1.D.iii.26);
- UFSAR Section 15.4.5, “Fuel Handling Accident,” and 15.5.6, “Environmental Consequences of a Postulated Fuel Handling Accident” (LAR Section II.1.D.iii.28);
- EQ Parameters, Radiation (LAR Section II.1.D.iii.32); and
- Tritium Production Accident Releases (LAR Section II.1.D.iii.37).

The NRC staff also confirmed that the licensee has accounted for the potential for an increase in measurement uncertainty should the LEFM CheckPlus System experience operational limitations.

The LEFM CheckPlus System and its operating modes are described in SE Section 3.1.2.

An LEFM CheckPlus System fail status indicates a loss of function. Loss of the LEFM CheckPlus System results in reverting to the calibrated venturi-based monitoring system. If the LEFM CheckPlus System becomes unavailable, the secondary side calorimetric is performed with inputs from the flow venturis, which requires a core power adjustment toward a lower core power based on the 2-percent uncertainty associated with the venture nozzle inaccuracies. Reactor power will be reduced from 3459 MWt (i.e., the MUR power uprate level) to 3411 MWt (i.e., the pre-MUR power uprate level) within 24 hours if LEFM functionality cannot be restored to the normal mode. These required actions will ensure that the current licensing basis dose consequence analyses remain bounding.

The NRC staff confirmed that the applicable current licensing basis dose consequence analyses remain bounding at the proposed MUR uprated power level of 3459 MWt with a margin that is within the assumed uncertainty associated with advanced flow measurement techniques, including the use of the LEFM CheckPlus System credited by the licensee. The NRC staff also confirmed that the licensee has accounted for the potential for an increase in measurement uncertainty should the LEFM system experience operational limitations. Using the licensing basis documentation as contained in the current WBN UFSAR, in addition to information in the LAR, the NRC staff verified that the existing radiological analyses and release assumptions bound the conditions for the proposed MUR power uprate, considering the higher accuracy of the proposed FW flow measurement instrumentation.

3.6.2.3 Technical Conclusion

As described above, the NRC staff reviewed the assumptions, inputs, and methods used by the licensee to assess the source terms for radwaste systems radiological consequence analyses, individual or cumulative occupational radiation exposure, and the postulated DBA radiological dose consequence analyses at the proposed MUR uprated power level. The NRC staff finds that operating WBN, Unit 2 at the proposed MUR uprated power level will continue to meet the applicable dose limits. The NRC staff has reasonable assurance that with the approval of the LAR, WBN, Unit 2 will continue to provide sufficient safety margins, with adequate defense-in-depth, to address unanticipated events and to compensate for uncertainties in accident progression, analysis assumptions, and input parameters. Therefore, the NRC staff concludes that the proposed MUR power uprate is acceptable with respect to radiological dose.

3.6.3 Fire Protection

3.6.3.1 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 or significantly increase the risk of radioactive releases to the environment. The NRC staff's review focused on the effects of the increased decay heat due to the proposed MUR power uprate 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 review criteria for the fire protection program are based on (1) 10 CFR 50.48, "Fire protection," (2) GDC 3, "Fire protection," and (3) GDC 5, "Sharing of structures, systems, and components." The regulation at 10 CFR 50.48 requires the development of a fire protection program to ensure, among other things, the capability to safely shutdown the plant.

GDC 3 requires that "structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components."

GDC 5 requires that "structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units."

Appendix R to 10 CFR Part 50 establishes fire protection features required to satisfy GDC 3 with respect to certain generic issues for nuclear power plants licensed to operate before January 1, 1979. Requirements III.G, III.J, and III.O are the only requirements that apply to plants licensed after January 1, 1979. Specifically, Requirement III.G specifies the fire protection features that must be provided for SSCs important to safe shutdown. Requirement III.J specifies that emergency lighting units with at least an 8-hour battery power supply shall be provided in all areas needed for operation of safe shutdown equipment and in access and egress routes thereto. Requirement III.O specifies that the reactor coolant pump shall be equipped with an oil collection system if the containment is not inerted during normal operation.

RIS 2002-03, Attachment 1, Sections II and III, recommend improving the efficiency of the NRC staff's review by having prospective LARs identify current accident and transient AORs that bound plant operation at the proposed uprated power level. For any design basis accident for which the existing AORs do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the reanalysis.

3.6.3.2 Technical Evaluation

The licensee developed the LAR consistent with the guidelines in RIS 2002-03. In the LAR, the licensee reevaluated the applicable SSCs and safety analyses at the proposed MUR uprated power level of 3459 MWt against the previously analyzed power level of 3411 MWt.

The NRC staff reviewed Enclosure 2 to the LAR, Section VII.6.A, "Fire Protection Program," which includes specific evaluations of each item outlined in RIS 2002-03. The NRC staff also reviewed the licensee's continued compliance with 10 CFR 50.48, i.e., the approved fire protection program, at the proposed MUR uprated power. The review covered the impact of the LAR on the results of the post-fire safe-shutdown as noted in RIS 2002-03, Attachment 1, Sections II and III. The review focused on the effects of the LAR on the post-fire safe-shutdown capability and increase in the decay heat generation following plant trips.

In Enclosure 2 to the LAR, Section VII.6.A, the licensee stated that the proposed MUR power uprate does not alter the components or component functions used in achieving and maintaining post-fire safe-shutdown and that the plant safe-shutdown analysis continues to be valid. The results of the 10 CFR Part 50, Appendix R, reactor coolant system cooldown analysis remain acceptable and the plant will still be able to achieve cold shutdown within 72 hours.

Further, the licensee stated that the proposed MUR power uprate does not cause environmental conditions in the plant to interfere with the performance of post-fire safe-shutdown operator actions. No new operator actions were identified. The licensee also conducted a review of the fire protection features of the plant and administrative controls credited in the approved fire protection program as outlined in the WBN fire protection report under MUR power uprate conditions. In addition, the licensee has credited the high-pressure fire water system to provide a backup source for non-fire protection purposes. The licensee stated that use of the high-pressure fire water system for non-fire protection purposes is not adversely impacted by the plant operating under MUR power uprate conditions. The licensee concluded that the MUR power uprate will have no effect on fire protection administrative controls, fire barriers, fire protection responsibilities of plant personnel, or resources necessary for systems required to achieve and maintain safe-shutdown.

The NRC staff has reviewed the information provided in the LAR regarding the fire protection program, post-fire safe-shutdown, and operator manual actions for compliance with 10 CFR Part 50, Appendix R. As stated above, the LAR involves no changes to the fire protection program that may adversely impact the post-fire safe-shutdown capability in accordance with 10 CFR Part 50, Appendix R. No new operator actions have been identified in areas where environmental conditions, such as heat, would challenge the operator or would become a challenge with MUR power uprate conditions. Because the MUR power uprate is being performed at a constant pressure and temperature, the normal temperature environments are not affected by the uprate. Therefore, the operator manual actions required to mitigate the consequences of a fire are not affected.

The licensee would make no significant changes to the plant configuration as a result of modifications necessary to implement the MUR power uprate. The licensee would make no changes and there would be no adverse effects created by the MUR power uprate on the fire detection or suppression systems, fire protection administrative controls, fire barriers, fire protection responsibilities of plant personnel, resources necessary for systems required to achieve and maintain safe-shutdown.

The NRC staff reviewed the licensee's statements in the LAR related to the impact of the proposed MUR power uprate on the plant safe-shutdown and impacts due to increase in the decay heat. For the MUR power uprate, the licensee reviewed its systems to achieve and maintain the plant in the cold shutdown condition. The WBN, Unit 2, Appendix R analysis demonstrates that the plant can reach cold shutdown with significant margin to the 72-hour requirements in 10 CFR Part 50, Appendix R, Sections III.G.1.b and III.L. The MUR power uprate and the additional decay heat removal needed would not affect the ability to reach and maintain cold shutdown within 72 hours. The NRC staff concludes that WBN, Unit 2 meets the 72-hour requirements in 10 CFR Part 50, Appendix R, Sections III.G.1.b and III.L, with the increased decay heat at MUR power uprate conditions. Further, the NRC staff concludes that the changes proposed in the LAR will not adversely impact post-fire safe-shutdown capability.

The NRC staff reviewed the LAR information concerning the use of fire protection water for non-fire suppression activities. The licensee stated that the proposed MUR power uprate operation will not impact the utilization of the high-pressure fire water system as a credited backup source for non-fire protection use. The NRC staff finds the licensee information acceptable because the licensee's analysis concluded that non-fire protection aspects of the fire protection system for other than fire protection activities would not be impacted, nor would they impact fire protection system design demands of the proposed MUR power uprate.

The information provided in the LAR demonstrates 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 does not change the credited equipment necessary for post-fire safe-shutdown or require rerouting of essential cables, relocation of essential components/equipment, or introduction of new or changes to existing operator manual actions credited for post-fire safe-shutdown. Further, the licensee would make no changes to the plant configuration or combustible loading to implement the MUR power uprate. The NRC staff finds that the proposed 1.4 percent power uprate will not have adverse effects on the post-fire safe-shutdown capability of the plant.

3.6.3.3 Technical Conclusion

Based on its review, the NRC staff concludes that the proposed MUR power uprate will not have a significant impact on the fire protection program or post-fire safe-shutdown capability and, therefore, finds the proposed amendment acceptable.

3.6.4 Containment Leakage Rate Testing Program

The NRC staff evaluated the impact of the proposed MUR power uprate on the Containment Leakage Rate Testing Program. The staff identified that the associated analyses assumed an initial power level that bounds MUR power uprate conditions; therefore, the MUR power uprate does not impact the Containment Leakage Rate Testing Program.

3.6.5 10 CFR 51.22 Criteria for Categorical Exclusion from Environmental Review

3.6.5.1 Regulatory Evaluation

10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," contains requirements for excluding licensing, regulatory, and administrative actions from an environmental assessment or an environmental impact statement. One of the categories of actions that is a categorical exclusion is the issuance of an amendment that changes a requirement with respect to the installation or use of facility components located within the restricted area, as defined in 10 CFR Part 20, or changes inspection or surveillance requirements, provided that:

> (i) The amendment … involves no significant hazards consideration;
>
> (ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite; and
>
> (iii) There is no significant increase in individual or cumulative occupational radiation exposure.

3.6.5.2 Non-Radiological Effluents

The NRC staff reviewed the effects of the proposed MUR power uprate on the non-radiological effluents. The staff also reviewed the non-radiological discharges (i.e., chemical and thermal discharge limits) from WBN to the environment as defined in National Pollutant Discharge Elimination System (NPDES) Permit No. TN0020168 for the years 2017 through 2019 (References 78, 79, 80). Based on its review, the NRC staff concludes that the MUR power uprate will not change the chemical discharges controlled by the NPDES permit. The types or amounts of effluents released into the environment will not change due to the power uprate. In addition, per a review of the information provided by the licensee, the NRC staff finds that the MUR power uprate will not change the thermal discharges. The cooling tower blowdown has a maximum thermal discharge limit of 95°F. The increased heat load on the cooling towers due to the MUR power uprate will have a negligible impact on cooling tower basin temperature (approximately 0.1°F). Supplemental CCW taken from the Tennessee River and gravity fed to the cooling tower basins provides for further blowdown dilution, as necessary. Therefore, the discharge and mixing zone thermal limits will continue to be met under MUR power uprate conditions and the NRC staff concludes that the thermal discharge will continue to comply with the NPDES requirements.

3.6.5.3 Radiological Effluents

The NRC staff reviewed the effects of the proposed MUR power uprate on the radiological effluents. During normal operations, the controls for the release rates of radwaste systems do not change with operating power. Thus, the NRC staff anticipates that there would be no impact on routine radiological releases due to the MUR power uprate. The licensee summarizes radiological environmental monitoring data and radioactive effluent release data at the WBN site in two annual reports: the Annual Radiological Environmental Operating Report and the Annual Radioactive Effluent Release Report. The WBN Offsite Dose Calculation Manual (ODCM) specifies the limits for all radiological releases. The radiological environmental monitoring program (REMP) is a sitewide program that monitors the radiological impacts from all radiation sources on the site. The NRC staff reviewed the annual radioactive effluents release reports for 2017 – 2019 (References 81, 82, 83), the 2018 Annual Radioactive Environmental Operating Report (Reference 84), which contains the measured doses due to normal effluents associated with the reactor operating at the CLTP, and the WBN ODCM and determined that the liquid and gaseous release data indicates that resultant doses are a small fraction of annual limits. Radioactive materials in liquid and gaseous effluents are reduced prior to being released into the environment so that the resultant dose to members of the public from these effluents will be below the NRC and federal U.S. Environmental Protection Agency dose standards. The effluent doses are determined in accordance with the ODCM. The licensee stated that the MUR power uprate will not result in changes to the ODCM. In addition, plant operations after implementation of the MUR power uprate would not result in significant doses compared to regulatory dose limits. Therefore, based on its review of the annual effluent release reports and the 2018 Annual Radioactive Environmental Operating Report, the NRC staff finds that the MUR power uprate for WBN, Unit 2 will not cause doses from liquid and gaseous effluent releases to exceed allowable limits.

3.6.5.4 Conclusion

Based on the above, the NRC staff concludes that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and no significant increase in individual or cumulative occupational radiation exposure.

3.7 CHANGES TO THE OPERATING LICENSE AND TECHNICAL SPECIFICATIONS ASSOCIATED WITH THE MUR POWER UPRATE

3.7.1 Regulatory Evaluation

The NRC staff reviews changes to the TS in accordance with 10 CFR 50.36, “Technical specifications,” in which the NRC established its regulatory requirements related to the content of TS. Specifically, 10 CFR 50.36(c) requires that TS include items in the following categories, in part: safety limits, limiting safety system settings, and limiting control settings; limiting conditions for operation; surveillance requirements; design features; and administrative controls.

3.7.2 Technical Evaluation

As described in Section 2.3 of this SE, the licensee proposed changes to the OL and TSs as discussed below.

WBN, Unit 2 OL Item 2.C.(1) would be revised to increase the maximum core power level from 3411 MWt to 3459 MWt.

The definition of RTP in TS 1.1, "Definitions," would be revised to account for the increase in reactor core thermal power level as follows:

> RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3459 MWt.

TS 5.9.5, "CORE OPERATING LIMITS REPORT (COLR)," paragraph b, would be revised to state:

> The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102% RTP is specified in a previously approved method, 100.6% RTP may be used only when feedwater flow measurement (used as input for reactor thermal power measurement) is provided by the leading edge flowmeter (LEFM) as described in document number 11 listed below. When feedwater flow measurements from the LEFM are unavailable, the originally approved initial power level of 102% RTP (3411 MWt) shall be used. The approved analytical methods are specifically those described in the following documents:

Additionally, the NRC-approved Caldon Topical Reports for LEFMs would be added as document number 11 in the list of documents in TS 5.9.5b.

As discussed throughout this SE, the NRC staff has determined that the licensee's proposal to increase the RTP from 3411 MWt to 3459 MWt as part of an MUR power uprate is acceptable. NRC Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," provides the framework for determining operating limits on a cycle-specific basis under administrative control (Reference 85). GL 88-16 recommends that such limits should be determined in accordance with NRC-approved methods and that those methods should be included in the TS. The methods that the licensee proposes to include have been approved for use by the NRC staff and, as described elsewhere in this SE, the NRC staff has determined that they are applicable to and acceptable for use at WBN, Unit 2. Therefore, the NRC staff determined that their proposed inclusion in WBN, Unit 2 TS 5.9.5.b is acceptable. Similarly, the NRC staff has determined that changing the RTP from 3411 MWt to 3459 MWt as stated in the OL and in TS 1.1 and making the proposed changes to TS 5.9.5b are acceptable.

As stated in the LAR, the TS Bases pages were provided for information only and will be updated by the TS Bases Implementation Program.

3.7.3 Conclusion

The NRC staff has reviewed the licensee's requested OL and TS changes associated with the implementation of the proposed MUR power uprate and determined that these changes are acceptable and that the TS, as revised, will continue to meet the regulatory requirements of 10 CFR 50.36. Therefore, the NRC staff concludes that the OL and TS changes associated with the LAR are acceptable.

3.8 TECHNICAL CONCLUSION

The NRC staff confirmed that the licensee provided all of the information discussed in RIS 2002-03 necessary to justify a smaller margin for power measurement uncertainty. Because the methodology used to quantify the uncertainty in the reactor thermal power uncertainty calculation is consistent with the limitations and conditions in the NRC-approved topical reports, the NRC staff has determined that the licensee may apply a reduced margin for power measurement uncertainty consistent with 10 CFR Part 50, Appendix K. Therefore, the NRC staff concludes that the licensee's request to correspondingly uprate the current licensed power for WBN, Unit 2 from 3411 MWt to 3459 MWt and make associated changes to the TSs is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Tennessee State official was notified of the proposed issuance of the amendment on May 19, 2020. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined, as described above in SE Section 3.6.5, that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on March 3, 2020 (85 FR 12581). 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 Commission 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) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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Principal Contributors:

H. Vu, NRR
S. Peng, NRR
G. Bedi, NRR
A. Chereskin, NRR
G. Cheruvenki, NRR
S. Wyman, NRR
M. Sodallah, NRR
D. Ki, NRR
Y. Cheng, NRR
P. Clark, NMSS
S. Sun, NRR
C. Basavaraju , NRR
R. Hernandez, NRR
B. Parks, NRR
A. Young, NRR
M. McConnell, NRR
E. Dickson, NRR
N. Iqbal, NRR
S. Smith, NRR

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LIST OF ACRONYMS

ACRONYM	DEFINITION
°F	degrees Fahrenheit
10 CFR	Title 10 of the *Code of Federal Regulations*
AC	alternating current
AFW	auxiliary feedwater
ALARA	as low as reasonably achievable
AMSAC	ATWS mitigation system circuitry
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
AOR	analysis of record
AOV	air-operated valve
ART	adjusted reference temperature
ASME	American Society of Mechanical Engineers
ASME OM Code	ASME Operation and Maintenance of Nuclear Power Plants Code
AST	alternative source term
ATWS	anticipated transient(s) without scram
BOP	balance-of-plant
BPV	ASME Boiler and Pressure Vessel Code
CCS	component cooling system
CLTP	current licensed thermal power
COLR	core operating limits report
COMS	cold overpressure mitigation system
CRDM	control rod drive mechanism
CSST	common station service transformer
CT	completion time
CVCS	chemical and volume control system
DBA	design basis accident
dc	direct current
DNB	departure from nucleate boiling
ECCS	emergency core cooling system
EDG	emergency diesel generator
EFPY	effective full-power year
EOL	end of license
EPRI	Electric Power Research Institute
EQ	environmental qualification
ERCW	essential raw cooling water
FIV	flow-induced vibration
ft^2	square feet (feet squared)
FW	feedwater
GDC	General Design Criterion
gpm	gallons per minute
GSI	Generic Safety Issue
HELB	high energy line break
HVAC	heating, ventilation, and air conditioning
HZP	hot zero power
ICS	integrated computer system
in^2	square inches (inches squared)

ACRONYM	DEFINITION
ISA	International Society of Automation
ISI	inservice inspection
IST	inservice testing
kV	kilovolt
LAR	license amendment request
LBB	leak-before-break
LEFM®	leading edge flow meter
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LOOP	loss of offsite power
MELB	moderate energy line break
MFW	main feedwater
Mlbm/hr	million pounds mass per hour
MOV	motor-operated valve
MS	main steam
MSVV	main steam valve vault
MUR	measurement uncertainty recapture
MWe	megawatt electric
MWt	megawatt thermal
NRC	U.S. Nuclear Regulatory Commission
NSSS	nuclear steam supply system
ODCM	Offsite Dose Calculation Manual
OL	Operating License
PORV	power-operated relief valve
psi	pounds per square inch
psia	pounds per square inch absolute
psig	pounds per square inch gauge
PSV	pressurizer safety valve
P-T	pressure-temperature
PTS	pressurized thermal shock
PWR	pressurized-water reactor
RAI	request for additional information
RCCA	rod cluster control assembly
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
rem	Roentgen equivalent man
RG	regulatory guide
RHR	residual heat removal
RIS	Regulatory Issue Summary
RPV	reactor pressure vessel
RTDP	Revised Thermal Design Procedure
RTP	rated thermal power
RV	reactor vessel
SAFDL	specified acceptable fuel design limit
SBO	station blackout
SE	safety evaluation
SFP	spent fuel pool
SG	steam generator
SGBS	steam generator blowdown system

ACRONYM	DEFINITION
SGTR	steam generator tube rupture
SLB	steam line break
SRP	Standard Review Plan
SRSS	square-root-sum-of-squares
SSCs	structures, systems, and components
Sv	sievert
TCD	thermal conductivity degradation
T_{cold}	cold leg temperature
T_{hot}	hot leg temperature
TEDE	total effective dose equivalent
TMD	transient mass distribution computer code
TPBAR	tritium producing burnable absorber rod
TPC	tritium producing core
TS	technical specification
TVA	Tennessee Valley Authority
UFM	ultrasonic flow meter
UFSAR	updated final safety analysis report
UHS	ultimate heat sink
USE	Charpy upper-shelf energy
USST	unit station service transformer
V	volt
WBN	Watts Bar Nuclear Plant

SUBJECT: WATTS BAR NUCLEAR PLANT, UNIT 2 - ISSUANCE OF AMENDMENT NO. 42 REGARDING MEASURMENT UNCERTAINTY RECAPTURE POWER UPRATE (EPID L-2019-LLS-0000) DATED OCTOBER 21, 2020

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