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

November 9, 2021

Mr. Daniel G. Stoddard
Senior Vice President and
Chief Nuclear Officer
Dominion Nuclear
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: MILLSTONE POWER STATION UNIT NO. 3 - ISSUANCE OF AMENDMENT NO. 280 REGARDING MEASURMENT UNCERTAINTY RECAPTURE POWER UPRATE (EPID L-2020-LLS-0002)

Dear Mr. Stoddard:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 280 to Renewed Facility Operating License No. NPF-49 for the Millstone Power Station Unit No. 3 (Millstone 3), in response to your application dated November 19, 2020, as supplemented by letter dated June 2, 2021.

This amendment increases the authorized reactor core power level by approximately 1.6 percent rated thermal power from 3,650 megawatts thermal (MWt) to 3,709 MWt, based on the use of the existing Cameron Technology US LLC (currently known as Sensia, formerly known as Caldon) Leading Edge Flow Meter CheckPlus system as an ultrasonic flow meter in each of the four main feedwater lines supplying the steam generators. The amendment also revises operating license paragraph 2.C.(1) and Technical Specification (TS) 1.27, to reflect the increase in rated thermal power. Additionally, TS 3.7.1.1, Action Statement “a” and TS Table 3.7-1, “Operable MSSVs Versus Maximum Allowable Power” are updated to revise the maximum allowable power levels corresponding to the number of operable main steam safety valves (MSSVs) per steam generator, and TS 2.1.1.1 is revised to make an editorial correction.

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

Sincerely,

/RA/

Richard V. Guzman, Senior Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-423

Enclosures:
1. Amendment No. 280 to NPF-49
2. Safety Evaluation

cc: Listserv

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

DOMINION ENERGY NUCLEAR CONNECTICUT, INC., ET AL

DOCKET NO. 50-423

MILLSTONE POWER STATION UNIT NO. 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 280
Renewed License No. NPF-49

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

   A. The application for amendment by Dominion Energy Nuclear Connecticut, Inc. (DENC, the licensee), dated November 19, 2020, as supplemented by letter dated June 2, 2021, 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 Renewed Facility Operating License No. NPF-49 are hereby amended to read as follows:

   (1) Maximum Power Level

   DENC is authorized to operate the facility at reactor core power levels not in excess of 3709 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, revised through Amendment No. 280 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated into the license. DENC 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 Millstone 3 spring 2022 refueling outage no later than December 31, 2022.

FOR THE NUCLEAR REGULATORY COMMISSION

Bo M. Pham

Digitally signed by Bo M. Pham
Date: 2021.11.09 13:50:28 -05'00'

Bo M. Pham, Director
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility Operating License and Technical Specifications

Date of Issuance: November 9, 2021

ATTACHMENT TO LICENSE AMENDMENT NO. 280

MILLSTONE POWER STATION UNIT NO. 3

RENEWED FACILITY OPERATING LICENSE NO. NPF-49

DOCKET NO. 50-423

Replace the following pages of the Renewed Facility Operating License with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

Remove	Insert
3	3
4	4

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	Insert
1-5	1-5
2-1	2-1
3/4 7-1	3/4 7-1
3/4 7-2	3/4 7-2

B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses;

(1) DENC, pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, use and operate the facility at the designated location in New London County, Connecticut in accordance with the procedures and limitations set forth in this license; Green Mountain Power Corporation and Massachusetts Municipal Wholesale Electric Company, pursuant to the Act and 10 CFR Part 50, to possess the facility at the designated location in New London County, Connecticut in accordance with the procedures and limitations set forth in this renewed operating license;

(2) DENC, pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) DENC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use at any time any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) DENC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material without restriction to chemical or physical form for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) DENC, pursuant to the Act and 10 CFR Parts 30, 40, and 70 to possess, but not separate, such byproducts and special nuclear materials as may be produced by the operations of the facility.

C. This renewed operating license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter 1 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) <u>Maximum Power Level</u>

DENC is authorized to operate the facility at reactor core power levels not in excess of 3709 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(1) Technical Specifications

The Technical Specifications contained in Appendix A, revised through Amendment No. 280 and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto are hereby incorporated into the license. DENC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(2) DENC shall not take any action that would cause Dominion Energy, Inc. or its parent companies to void, cancel, or diminish DENC's Commitment to have sufficient funds available to fund an extended plant shutdown as represented in the application for approval of the transfer of the licenses for MPS Unit No. 3.

(3) Immediately after the transfer of interests in MPS Unit No. 3 to DNC*, the amount in the decommissioning trust fund for MPS Unit No. 3 must, with respect to the interest in MPS Unit No. 3, that DNC* would then hold, be at a level no less than the formula amount under 10 CFR 50.75.

(4) The decommissioning trust agreement for MPS Unit No. 3 at the time the transfer of the unit to DNC* is effected and thereafter is subject to the following:

(a) The decommissioning trust agreement must be in a form acceptable to the NRC.

(b) With respect to the decommissioning trust fund, investments in the securities or other obligations of Dominion Energy, Inc. or its affiliates or subsidiaries, successors, or assigns are prohibited. Except for investments tied to market indexes or other non-nuclear-sector mutual funds, investments in any entity owning one or more nuclear power plants are prohibited.

(c) The decommissioning trust agreement for MPS Unit No. 3 must provide that no disbursements or payments from the trust, other than for ordinary administrative expenses, shall be made by the trustee until the trustee has first given the Director of the Office of Nuclear Reactor Regulation 30 days prior written notice of payment. The decommissioning trust agreement shall further contain a provision that no disbursements or payments from the trust shall be made if the trustee receives prior written notice of objection from the NRC.

(d) The decommissioning trust agreement must provide that the agreement cannot be amended in any material respect without 30 days prior written notification to the Director of the Office of Nuclear Reactor Regulation.

* On May 12, 2017, the name “Dominion Nuclear Connecticut, Inc.” changed to “Dominion Energy Nuclear Connecticut, Inc.”

PHYSICS TESTS

1.21 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

1.22 DELETED

PURGE - PURGING

1.23 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.24 QUADRANT POWER TILT RATIO 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. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.27 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3709 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.28 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its 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.

REPORTABLE EVENT

1.29 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

SHUTDOWN MARGIN

1.30 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, Reactor Coolant System highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT; and the following Safety Limits shall not be exceeded:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained greater than or equal to 1.14 for the WRB-2M DNB correlation.

2.1.1.2 The peak fuel centerline temperature shall be maintained less than 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the Reactor Core Safety Limit is violated, restore compliance and be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

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

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2750 psia be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves (MSSVs) shall be OPERABLE with lift settings as specified in Table 3.7-3.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

- NOTE -

Separate Condition entry is allowed for each MSSV.

- -

a. With one or more steam generators (SGs) with one MSSV inoperable, and the Moderator Temperature Coefficient (MTC) zero or negative at all power levels, within 4 hours reduce THERMAL POWER to less than or equal to 59% RATED THERMAL POWER (RTP); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one or more SGs with two or more MSSVs inoperable, within 4 hours reduce THERMAL POWER to less than or equal to the maximum allowable % RTP specified in Table 3.7-1 for the number of OPERABLE MSSVs, and reduce the Power Range Neutron Flux High setpoint to less than or equal to the maximum allowable % RTP specified in Table 3.7-1 for number of OPERABLE MSSVs within the next 32 hours*; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With one or more SGs with one MSSV inoperable and the MTC positive at any power level, within 4 hours reduce THERMAL POWER to less than or equal to the maximum allowable % RTP specified in Table 3.7-1 for the number of OPERABLE MSSVs and reduce the Power Range Neutron Flux High setpoint to less than or equal to the maximum allowable % RTP specified in Table 3.7-1 for number of OPERABLE MSSVs within the next 32 hours*; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

* Applicable only in MODE 1.

TABLE 3.7-1
OPERABLE MSSVS VERSUS MAXIMUM ALLOWABLE POWER

NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM ALLOWABLE POWER (PERCENT OF RATED THERMAL POWER)
4	59
3	41
2	24

TABLE 3.7-2

DELETED

ENCLOSURE 2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 280

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-49

DOMINION ENERGY NUCLEAR CONNECTICUT, INC., ET AL

MILLSTONE POWER STATION UNIT NO. 3

DOCKET NO. 50-423

Contents

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1.0 INTRODUCTION

By letter dated November 19, 2020 (Reference 1), as supplemented by letter dated June 2, 2021 (Reference 2), Dominion Energy Nuclear Connecticut, Inc. (DENC, the licensee), submitted a license amendment request (LAR) proposing changes to the Renewed Facility Operating License (RFOL) and technical specifications (TSs) for Millstone Power Station Unit No. 3 (Millstone 3 or MPS3).

The licensee proposed to revise the RFOL and TS to implement a measurement uncertainty recapture (MUR) power uprate that would increase the authorized reactor core power level by approximately 1.6 percent rated thermal power from 3,650 megawatts thermal (MWt) to 3,709 MWt, based on the use of the existing Cameron Technology US LLC (currently known as Sensia, formerly known as Caldon) Leading Edge Flow Meter (LEFM) CheckPlus System. The LAR would also revise RFOL Item 2.C.(1) and TS 1.27, "Definitions – Rated Thermal Power," to reflect the increase in rated thermal power and update TS 3.7.1.1, Action Statement "a" and TS Table 3.7-1, "Operable MSSVs Versus Maximum Allowable Power" to revise the maximum allowable power levels corresponding to the number of operable main steam safety valves (MSSVs) per steam generator. Additionally, the proposed changes would revise TS 2.1.1.1 to make an editorial correction.

The supplemental letter dated June 2, 2021, 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 April 1, 2021 (86 FR 17211).

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] Evaluation Models," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50 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. This change provided licensees the option of maintaining the 2 percent power margin between licensed power level and the ECCS evaluation assumed power level or applying a reduced ECCS evaluation margin based on an accounting of uncertainties due to instrumentation error.

In order to provide guidance to licensees seeking an MUR power uprate on the basis of improved feedwater (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 3). 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.

The neutron flux instrumentation continuously indicates the RTP. This instrumentation must be calibrated periodically 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.[1] 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

1 "Venturi" will generally be used in the remainder of this document to reference both venturi and flow nozzles.

computer-controlled electronic transducers that do not have differential pressure elements that are susceptible to fouling.

2.2 Implementation of an MUR Power Uprate at Millstone 3

The licensee intends to use a power measurement uncertainty of 0.325 percent based on utilizing an installed Cameron Technology US LLC (formally, Caldon)[2] Leading Edge Flow Meter (LEFM) CheckPlus system as an ultrasonic flow meter (UFM) located in each of the four main feedwater lines supplying the steam generators. This system provides a more accurate measurement of feedwater (FW) flow (and correspondingly, RTP) as compared to the accuracy of the venturi flow meter-based instrumentation, including those available when 10 CFR Part 50, Appendix K, was issued. The proposed MUR power uprate is based on a redistribution of analytical margin originally required of ECCS evaluation models performed per the requirements of 10 CFR 50, Appendix K. Based on the use of the Caldon instrumentation to determine core power level with a power measurement uncertainty of approximately 0.325 percent, the licensee proposes to reduce the licensed power uncertainty within the requirements of 10 CFR Part 50, Appendix K, resulting in an approximately 1.6 percent increase in RTP.

Millstone 3 was initially licensed to operate at a maximum of 3,411 MWt. By NRC License Amendment No. 242, the NRC approved the licensee's LAR for a stretch power uprate (SPU) at the current power level of 3,650 MWt (Reference 4).

UFMs were installed, calibrated and commissioned for Millstone 3 in 2004, and provide on-line main feedwater flow and temperature measurement to determine reactor thermal power. The UFMs use acoustic energy pulses to measure main feedwater temperature and flow rate. The UFMs consist of a measuring section containing 16 ultrasonic multi-path transit time transducers, one dual resistance temperature detector (RTD), and one pressure transmitter installed in each of the four feedwater lines, and an electronic signal processing cabinet.

The LEFM CheckPlus system will be used in lieu of the current venturi-based feedwater flow indication and RTD temperature indication to perform the plant calorimetric measurement calculation. Although the LEFM CheckPlus system is part of the implementation of the MUR power uprate, the currently installed venturi-based feedwater flow instruments will remain in place and continue to provide inputs to other indication and control systems and will be used as a backup method, if the UFM is not functional.

The LEFM system was developed over a number of years. Caldon submitted Topical Report (TR), Engineering Report (ER)-80P, Revision 0 (ER-80P), "Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM System," in March 1997 (Reference 5), that describes the LEFM and includes calculations of power measurement uncertainty obtained using a Check system in a typical two-loop pressurized-water reactor or a two-FW-line boiling-water reactor. The topical report also provided guidance for determining plant-specific power calorimetric uncertainties. The NRC staff approved the use of ER-80P 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 (ER-80P SE) (Reference 6), which allowed a 1 percent power uprate using the LEFM. The NRC staff's SE included four criteria that should be addressed by licensees incorporating ER-80P in their plant licensing basis.

[2] This safety evaluation refers to Caldon and Cameron interchangeably.

Following the publication of the changes to 10 CFR Part 50, Appendix K, which allowed for an uncertainty less than 2 percent, Caldon submitted TR ER-160P, Revision 0, “Supplement to Engineering Report ER-80P: Basis for a Power Uprate with the LEFM System” (Reference 7). The NRC staff approved a plant-specific precedent using ER-160P by letter dated January 19, 2001 (Reference 8), for a power uprate of up to 1.4 percent at Watts Bar Nuclear Plant, Unit 1.

Subsequently, in an SE dated December 20, 2001 (Reference 9), the NRC staff approved a plant-specific precedent using ER-157P, Revision 5, “Supplement to Engineering Report ER- 80P: Basis for a Power Uprate with the LEFM or LEFM CheckPlus System,” (Reference 10) for a power uprate of up to 1.7 percent using the CheckPlus system.

Caldon submitted ER-157P, Revision 8, “Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or an LEFM CheckPlus System” (ER-157P, Rev. 8), in May 2008 (Reference 11) and an associated errata on October 15, 2010 (Reference 12). The ER-157P, Rev. 8 corrected minor errors in Revision 5, provided clarifying text, and incorporated revised analyses of coherent noise, non-fluid delays, and transducer replacement. It also added two new appendices, Appendix C and Appendix D, which describe the assumptions and data that support the coherent noise and transducer replacement calculations, respectively.

In its SE for ER-157P, Rev. 8 (Reference 13), dated August 16, 2010 (ER-157P SE), the NRC staff approved ER-157P, Rev. 8 and associated errata, subject to five criteria that should be addressed by licensees incorporating ER-157P, Rev. 8 in their plant licensing basis.

2.3 Licensee’s Proposed Changes

The licensee requested the following changes:

- The Millstone 3 RFOL Paragraph 2.C(1), “Maximum Power Level,” would be revised to increase in maximum reactor core power level from 3,650 MWt to 3,709 MWt.

- The definition of RTP in TS 1.27 would be revised to account for the increase in reactor core power level as follows:

  RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3709 MWt.

- TS Section 2.1.1.1 would be revised to replace “correlations” with “correlation” since the departure from nuclear boiling ratio (DNBR) acceptance criteria for the WRB-2M is a singular correlation as follows:

  The departure from nucleate boiling ration (DNBR) shall be maintained greater than or equal to 1.14 for the WRB-2M DNB correlation.

- TS Section 3.7.1.1, Action Statement “a” would be revised to require reduction of thermal power to 59 percent RTP as follows:

  a. With one or more steam generators (SGs) with one MSSV inoperable, and the Moderator Temperature Coefficient (MTC) zero or negative at all power levels, within 4 hours reduce THERMAL POWER to less than or equal to 59% RATED THERMAL POWER (RTP); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

- TS Table 3.7-1, “Operable MSSVs Versus Maximum Allowable Power” would be updated to revise the maximum allowable power levels corresponding to the number of operable MSSVs per steam generator, from 60.1 to 59, from 42.8 to 41, and from 25.5 to 24 which corresponds to 4, 3, and 2 operable MSSVs per steam generator, respectively.

2.4 Regulatory Requirements and Guidance

Due to the numerous technical review sections of an MUR power uprate, a regulatory evaluation section is included for each technical section rather than consolidated here. Below are the regulatory requirements and guidance documents that are universally applicable to Millstone 3 and the LAR.

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.

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 (Reference 3).

The Millstone 3 Final Safety Analysis Report (FSAR) specifies the design bases of Millstone 3 as measured against the NRC General Design Criteria for Nuclear Power Plants, Appendix A, to 10 CFR 50, as amended through October 27, 1978. The General Design Criteria (GDC) include the single failure definition and 55 individual criteria, and are intended to establish the basic requirements for the principal design criteria of nuclear power plants.

The GDC that constitute the licensing bases for Millstone are those described in the FSAR, Chapter 3.1, “Conformance with NRC General Design Criteria,” and in applicable FSAR sections. The staff identified the criteria in Chapter 3.1 as being applicable to the proposed amendment and a basis for review acceptance criteria. The NRC staff considered the following GDC as part of its review of the Millstone 3 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.

Regulatory Guide (RG) 1.105, Revision 4, “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 14). RG 1.105 endorses Part I of Instrument Society of America[3] (ISA)-S67.04.01-2018, "Setpoints for Nuclear Safety-Related Instrumentation" (Reference 15). The NRC staff used this guide to establish the adequacy of the licensee's setpoint calculation methodologies and the related plant surveillance procedures.

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 Millstone 3 LEFM CheckPlus ultrasonic flow meter system to revise the licensed power measurement uncertainty to achieve an increase of approximately 1.6 percent in the licensed power level.

Neutron flux instrumentation is calibrated to the core thermal power, which is determined by an automatic or manual calculation of the energy balance around the plant nuclear steam supply system. The accuracy of this calculation depends primarily on the accuracy of feedwater flow and feedwater net enthalpy measurements. Feedwater flow is the most significant contributor to the core thermal power uncertainty. A more accurate measurement of this parameter will result in a more accurate determination of core thermal power.

Nuclear power plants are licensed to operate at a specified core thermal power. Appendix K to 10 CFR Part 50, "ECCS Evaluation Models," requires LOCA and ECCS analyses to assume "that the reactor has been operating continuously at a power level at least 1.02 times the licensed thermal power level to allow for instrumentation uncertainties." Alternatively, Appendix K allows such analyses to assume a value lower than the specified 102 percent, but not less than the licensed thermal power level, "provided the proposed alternative value has been demonstrated to account for uncertainties due to power level instrumentation error." This allowance gives licensees the option of maintaining the two percent (2 percent) power uprate with reduced margin between the licensed power level and the power level assumed in the ECCS analysis by using more accurate instrumentation to calculate the RTP.

Feedwater flow rate is typically measured using a venturi. This device generates a differential pressure proportional to the square of the feedwater velocity in the pipe. Due to the need to improve flow instrumentation measurement uncertainty, the industry evaluated other flow measurement techniques and found the Cameron Check and LEFM CheckPlus ultrasonic flow meters to be a viable alternative.

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

3.1.2 Leading Edge Flow Meter Technology and Measurement

The 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, which are used in the LEFM, 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 fluid density may be used to determine the feedwater 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.

The Millstone 3 LEFM CheckPlus system has two operational modes: NORMAL operation and MAINTENANCE mode.

- NORMAL operation is defined as CheckPlus operation: In this mode, both planes of transducers are in-service and system operations are processed by both central processing units (CPUs).

- MAINTENANCE mode: If the system is subjected to a failure involving a transducer, failure of one plane of operation or if a CPU-related malfunction occurs that affects one plane of operation, the system reverts to the Maintenance mode. An annunciator on the main control board will alert the control room operators whenever a failure occurs that impacts the accuracy of the UFM system, as well as a failure that results in the system being non-functional. In parallel, the alarm will annunciate on the plant process computer.

In the Millstone 3 Control Building Instrument Rack Room, the proposed Cameron LEFM CheckPlus system includes an electronic cabinet and four measurement spool pieces to be installed in each of the four main feedwater flow headers upstream of the existing feedwater venturi flow meters.

The licensee stated in its LAR,

> The feedwater mass flow rate, pressures and temperatures are transmitted to the plant process computer (PPC) where they are utilized in the calculation of the calorimetric measurement (secondary plant energy balance) of reactor thermal output. Additionally, the PPC will normalize the existing feedwater venturi flow, temperature, and pressure values used in the calorimetric measurement to the UFM [ultrasonic flow meter] readings. The normalized values will be used for feedwater venturi calorimetric power calculations should the UFM be unavailable for short periods of time. The UFM system will utilize continuous calorimetric power determination via redundant serial links with the plant computer and will

incorporate self-verification features. These features ensure that the UFM system performance is consistent with the design basis.”

In the LAR, the licensee stated that when the reactor power is greater than 3,650 MWt and a Millstone 3 LEFM system is non-functional (when ramping up in power between 3,650 MWt and 3,709 MWt), power ascension cannot continue until the UFM system is restored to functional status. The normalized signals from the existing functional feedwater flow venturis will be used as input to the secondary calorimetric portion of the RTP calculation in place of the LEFM system.

3.1.3 Reason for Proposed Changes

Based on the Cameron LEFM instrumentation, the licensee is able to determine the core power level with a power measurement uncertainty value (rounded up) of 0.4 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 1.6 percent in the licensed power level. This would increase the licensed power output from 3,650 MWt to 3,709 MWt for Millstone 3.

Because Millstone 3 is requesting a 1.6 percent power uprate, Millstone 3 would be permitted to operate at 3,709 MWt with LEFM CheckPlus System functional. However, a 48-hour allowed outage time (AOT) is currently proposed for when the LEFM CheckPlus System is non-functional. AOT in the LAR is also referred to as completion time (CT).

The NRC staff reviewed the proposed plant-specific implementation of the feedwater 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 (Reference 3) which relates to 10 CFR 50, Appendix K. The NRC staff confirmed that the licensee’s implementation of the proposed feedwater flow measurement device is consistent with staff-approved Topical Reports ER-80P (Reference 5) and ER 157P (Reference 11) 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 0.4 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 Evaluation of Proposed Total Thermal Power Uncertainty Value up to 0.4 Percent

3.1.4.1 Licensee’s Methodology

The licensee requested a 1.6 percent power uprate for Millstone 3 based on utilizing an installed Cameron Technology US, LLC, LEFM CheckPlus system as a UFM located in each of the four main feedwater lines supplying the SGs.

In the LAR, the licensee stated that:

> The methodology used for the feedwater venturis to determine the channel statistical allowance is the Square Root of the Sum of the Squares (SRSS). The SRSS method combines the uncertainty components for a channel in an appropriate combination of groups that are statistically and functionally independent. Those uncertainties that are not independent are conservatively treated by arithmetic summation, and then combined by SRSS with the independent terms. Appropriate bias terms are arithmetically added as necessary for the function. This methodology is described in WCAP-16617-P, Revision 0," (Reference 16).

The vendor's determination of the uncertainty of the Cameron LEFM CheckPlus System is consistent with this methodology, as described in the referenced approved topical reports. In addition, the NRC staff reviewed ER-369 (Reference 17), ER-394 (Reference 18), and WCAP-16617-P (Reference 16), to verify that the setpoint methodology in these documents is SRSS and consistent with the RG 1.105 and Section 4.4, "Combination of uncertainties," of Part I ISA S67.04-1994 (Reference 15).

3.1.4.2 Instrumentation Uncertainties Evaluation

In the LAR, the licensee stated that:

> The total thermal power uncertainty using the UFM at MPS3 [Millstone 3] is 0.325 % of RTP when operating in the UFM System Normal Mode. The thermal power uncertainty is 0.466% when operating in the UFM System Maintenance Mode. The uncertainty analysis bounds the CheckPlus system installed at Millstone 3.
>
> The uncertainty calculation for MPS3 is documented in Cameron report ER-369 (Reference 17)[4], and the non-proprietary and proprietary versions of this report are provided in Attachments 6 and 7 [of the LAR], respectively. The key parameters and their uncertainties from Reference 17 are summarized in LAR Table I.1.E-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 will be able to operate at up to 3,709 MWt (uprated RTP). The normalized calorimetric instrumentation retains the accuracy of the UFM above 90 percent RTP. However, if the LEFM CheckPlus System becomes non-functional above 3,650 MWt, the maximum permitted power level must then be reduced to pre-MUR power level of 3,650 MWt (98.4 percent of the uprated RTP) within 48-hour period. The 48-hour completion time is discussed in Section 3.1.5.2 in this SE. The term completion time in the LAR is also referred to as allowed outage time (AOT).

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[4] Cameron Report ER-369 is Reference I-12 of the LAR, but it is Reference 17 of this SE.

The licensee provided ER-369 Revision 6 (Reference 17)[5] in Attachment 6 of the LAR, (LAR MUR power uprate Cameron non-proprietary documents). In this report, all random errors and bias terms are calculated and combined by the means of the SRSS. This report provided:

- The thermal power uncertainty using a fully functional LEFM CheckPlus system is ± 0.325 percent.
- Thermal power uncertainty using a maintenance mode LEFM CheckPlus system is ±0.466 percent.

This indicates the proposed total thermal power uncertainty 0.4 percent bounds the calculated uncertainty value (0.325 percent). In addition, the total power uncertainty of 0.4 percent is still less than the thermal power uncertainty when using a maintenance mode LEFM CheckPlus system (± 0.466 percent). In maintenance mode, with ± 0.466 percent thermal power uncertainty, the power would be 101.534 percent and bound the calculated uncertainty value (0.325 percent).

In Attachment 6 of LAR, the licensee also provided ER-394 (Cameron non-proprietary version) (Reference 18). This report documents the profile factors (e.g., calibration coefficients) and their uncertainty for the Millstone 3 LEFM Check flow meters constructed. This report includes a description of the calibration facility, a description of the tests conducted, a discussion of the profile factor calculation, and an evaluation of the uncertainty in the meter profile factor/discharge coefficient. The methodology for the calculating uncertainty or accuracy of the LEFM Check meter profile factor and for the universal venturi tubes (UVT) discharge coefficient is the SRSS.

The NRC staff performed an audit review (Reference 19) of the WCAP-16617-P, Revision 0 (Reference 16), to verify the licensee's use of the methodology for the feedwater venturis to determine the channel statistical allowance of the Millstone 3 is the SRSS. The parameter uncertainties that are used in the Millstone 3 uncertainty analysis are consistent with the requirements of the RG 1.105, Revision 3 (Reference 14).

The NRC staff independently verified that the proposed uncertainty of 0.4 percent is more conservative than the calculated uncertainty of 0.325 percent. The proposed power increase of 101.6 percent RTP with uncertainty 0.4 percent is consistent with the guidance and satisfies the regulations requirements in Section 2.4 of this SE and therefore, is acceptable. The Millstone 3 power level relationships are illustrated in Figure 1 below.

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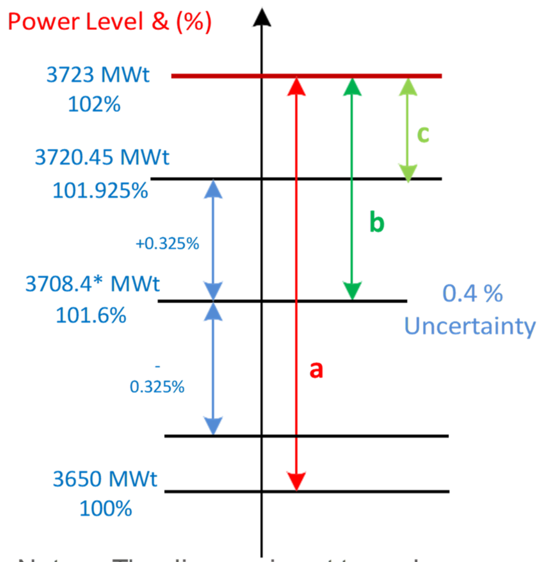
[5] In Cameron Report ER-369: (1) the LEFM CheckPlus is "LEFM✓+", (2) LEFM Check is "LEFM ✓"

Figure 1: Millstone 3 Power Level Relationships

Millstone Unit 3 Power Level Relationships



Note: The diagram is not to scale

* 3650 MWt x 101.6% = 3708.4

Margin Calculation

Row #		% Uncertainty	Margin (MWt)
1	a	2.000	73.00
2	b	0.400	14.60
3	**c = 0.4% - 0.325%**	0.075	2.55
			%
4	% Margin = (b/a)*100		20.00
5	% Margin = (c/a)*100		3.49

a: Margin between 102% power and original (100%) power
b : Margin between 102% power and power increase with 0.4% uncertainty
c: Margin between 102% power and power increase with 0.325% uncertainty

The NRC staff, therefore, made the following determinations with respect to the proposed power increase to 101.6 percent RTP (with proposed total uncertainty 0.4 percent) for Millstone 3:

- The value of 3,723 MWt which corresponds to 102 percent of RTP is greater than 3,708.4 MWt (which corresponds to 101.6 percent of RTP with the uncertainty 0.4 percent). This indicates that the 102 percent of RTP remains the bounding power level for MUR power uprate conditions when total uncertainty 0.4 percent is applied.

- The margin percentage between margins “b” and “a” (20.00 percent in Row 4) is adequate.

- In addition, this percentage margin is greater than the percentage margin between margins “c” and “a” (3.49 percent in Row 5). That showed the total thermal power uncertainty 0.4 percent is more conservative than 0.325 percent overall uncertainty since the reactor will be operating below the calculated power level. Therefore, the licensee chose the 0.4 percent instead of 0.325 percent for uncertainty.

The NRC staff verified that the proposed total thermal power uncertainly, that has been chosen (0.4 percent), is achieved by the LEFM system operating with the bounds of its uncertainty analysis (0.325 percent). This 0.4 percent for 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 to 101.6 percent RTP with total uncertainty of 0.4 percent is acceptable.

3.1.5 Evaluation of LAR 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 feedwater 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 6 and 13, respectively).

In its response to item I.1.C, the licensee stated that the LEFM CheckPlus System is installed in Millstone 3 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 feedwater 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 50, Appendix K.

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 6 and 13, respectively). The licensee's LAR addresses each of the criteria as follows:

<u>Criterion 1 from the SE for ER-80P</u>

> 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 the LEFM CheckPlus system has been operating at MPS3 since its commissioning in 2004, and as such, the necessary maintenance and calibration procedures of the LEFM CheckPlus System have been implemented to ensure that the system is properly maintained and calibrated. These procedures will be reviewed and revised as required with respect to Millstone 3 operation at the uprated power level. LEFM CheckPlus system maintenance is based upon applicable requirements of Cameron's maintenance and troubleshooting manual. Preventive maintenance scope and frequency is based on vendor recommendations and performance data reviews.

The preventive maintenance activities include: general terminal and cleanliness inspection, power supply inspection, CPU inspection, acoustic processor unit checks, analog input/output checks, alarm relay checks, watchdog timer checks that ensure the software is running, wall thickness measurement of LEFM spool pieces, communication checks, transducer checks, and calibration checks on each feedwater pressure transmitter.

Contingency plans for plant operation with an inoperable LEFM are discussed in Section 3.1.5.5 of this SE below.

The NRC staff reviewed the licensee's evaluation of maintenance and calibration procedures including the processes and contingencies for an inoperable LEFM CheckPlus system and the NRC staff concludes that the licensee adequately addressed Criterion 1 from the SE for ER-80P.

<u>Criterion 2 from the SE for ER-80P</u>

> 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 Millstone 3 LEFM system installed instrumentation is bounded by the analysis and assumptions set forth in Topical Report ER-80P.

Since the installation and calibration of LEFMs in Millstone 3 according to analysis and assumptions set forth in the TR ER-80P, there have been minor maintenance activities without compromising reliability of the system. Components such as transducers, power supplies and analog power units have been replaced due to unexpected failures as well as through discovery during the performance of proactive site maintenance practices using the guidance and instructions from the vendor. Corrective actions were taken to prevent recurrences of failures according to site corrective action program. Both feedwater flow venturis and the steam flow venturis can be normalized to the LEFM flow to monitor for consistency and accuracy of the measurements. Currently, at full power operation, the steam flow venturis are normalized based on the calculated LEFM feedwater flowrate values in real time. The Millstone 3 LEFMs are operating as designed.

The NRC staff reviewed the licensee's evaluation of operational and maintenance history for the plant operation with an inoperable LEFM CheckPlus system and the NRC staff concludes that the licensee has adequately addressed Criterion 2 from the SE for ER-80P.

<u>Criterion 3 from the SE for ER-80P</u>

> 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 Millstone 3 LEFM system uncertainty calculation methodology is described in Cameron reports ER-80P and ER-157P and based on: (1) the American Society of Mechanical Engineers

(ASME) Performance Test Code (PTC) 19.1-1985, "Measurement Uncertainty" (Reference 165); (2) Millstone 3 LEFM CheckPlus system was calibrated at Alden Research Labs with traceability to National Standards. A copy of the Alden Research Labs certified calibration report is contained in the Cameron ER-394 (the non-proprietary version in Attachment 6 of LAR (Reference 18), and the proprietary version of this report is provided in Attachment 7 of LAR; (3) Millstone 3 LEFM uncertainty analysis is provided in Cameron report ER-369 (in Attachment 6 of LAR). This methodology is described in WCAP-16617-P, Revision 0 (Reference 16); and (4) the ISA Standard 67.04, Part I (Reference 15) methodology has been used for instrument uncertainty calculations for multiple MUR power uprates that were accepted by the NRC. An alternative methodology for calculating LEFM uncertainty was not used.

The methodology consists of statistically combines inputs into groups which are then combined using the square root of the sum of squares (SRSS) approach to determine the overall uncertainty. This approach is used to determine the Millstone 3 LEFM based power calorimetric uncertainty. The SRSS method combines the uncertainty components for a channel in an appropriate combination of groups that are statistically and functionally independent. Those uncertainties that are not independent are conservatively treated by arithmetic summation, and then combined by SRSS with the independent terms.

Based on the discussion above and the staff's review of the licensee's LAR, the NRC staff concludes that the licensee adequately addressed Criterion 3 from the SE for ER-80P (Reference 6).

<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 licensee stated that this criterion does not apply to Millstone 3. The LEFM system was calibrated in a site-specific model test at the Alden Research Laboratory. The UFM was installed and commissioned in accordance with Cameron procedures. These procedures included verification of ultrasonic signal quality and hydraulic velocity profiles as compared to those during site-specific model testing. An Alden Research Laboratory data report for these tests is contained in the Cameron Engineering Report ER-394, Revision 0 (Reference 18) and was provided in LAR Attachments 6 and 7 to the LAR. The piping configuration at Millstone 3 remains bounded by the original LEFM flow meter installation and calibration assumptions as analyzed in Cameron engineering reports ER-80P and ER-157P (References 5 and 13).

Based on the information given above and the staff's review of the licensee's submitted calibration data in Cameron Engineering Reports ER-394 and ER-369, the NRC staff finds that Criterion 4 from the SE for ER-80P is not applicable to the Millstone 3 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.

In its LAR, the licensee stated, in part:

> With reactor power >3,650 MWt and a LEFM system non-functional, normalized signals from the existing functional feedwater flow venturis will be used as input to the secondary calorimetric portion of the RTP calculation in place of the LEFM system. During normal LEFM operations, the signals from the flow venturi are normalized to the LEFM signals, and upon LEFM failure, the feedwater venturi normalization factor will be based on the most recent accurate LEFM data.

The licensee proposed a 48-hour AOT for operating above the current licensed thermal power limit (3,650 MWt) if the normalized feed water parameters are adjusted to match the UFM values. The licensee specifically noted that during 48-hour AOT in the event of a UFM failure, the flow venturi signals to the secondary calorimetric portion of the RTP calculation demonstrates that instrumentation and RTP drift should be insignificant (NRC staff reviewed the drift data in WCAP-16617-P, Revision 0 (Reference 16). This indicates that Millstone 3 would use alternate plant instruments (i.e., the existing feedwater venturis and RTDs) for the feedwater flow rate calculation. If the UFM System is not restored within 48 hours, the reactor thermal power must be lowered to less than or equal to 3,650 MWt.

The licensee also performed a statistical analysis and review of normalization factor data for plant instrumentation providing the flow venturi signals to the secondary calorimetric portion of the RTP calculation. This demonstrates that instrumentation and RTP drift should be insignificant over a 48-hour period. The licensee stated:

> A review of flow venturi fouling history demonstrates that fouling/de-fouling should not introduce significant error/drift over a 48-hour period. This indicates that Millstone 3 can be operated for 48 hours without exceeding the licensed RTP limit when the normalized feedwater flow venturi signals are used as an input to the secondary calorimetric portion of the RTP calculation in place of the LEFM system.

Moreover, the NRC has approved a 48-hour AOT for previous MUR power uprate applications, such as North Anna Power Station Units 1 and 2 and for Surry Power Station, Units 1 and 2 (References 20 and 21, respectively).

Thus, over a 48-hour period, feedwater flow nozzle instrument drift would have an insignificant effect on the feedwater flow measurement. Based on the discussion above and the staff's review of the licensee's LAR, the NRC staff finds that the licensee adequately addressed Criterion 1 of the SE for ER-157P.

<u>Criterion 2 from the SE for ER-157P</u>

> 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.

In its LAR, the licensee stated that:

> MPS3 will consider a CheckPlus system with a single failure as a non-functional LEFM and the same actions identified in response to Criterion 1 from ER-157P, Revision 8 above would be followed.

The NRC staff reviewed the licensee's consideration of the CheckPlus operating with a single failure as a non-functional LEFM and the applicable actions as discussed in response to Criterion 1 from ER-157P above, and concludes the licensee adequately addressed Criterion 2 of the SE for ER-157P.

<u>Criterion 3 from the SE for ER-157P</u>

> An applicant with a comparable geometry can reference the above Section 3.2.1 [of Reference 13] 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 its LAR, the licensee stated that:

> As stated in response to Criterion 2 from ER-157P, Revision 8 above, MPS3 will consider a CheckPlus system with disabled components as a non-functional LEFM and the same actions identified in response to Criterion 1 above would be followed.

As determined in the NRC staff's review of the Criterion 2 from ER-157P, Revision 8, above, Millstone 3 considers a CheckPlus system with disabled components as a non-functional LEFM and the same actions identified in the licensee's response to Criterion 2 in the LAR would be followed. Therefore, the NRC staff concludes the licensee adequately addressed Criterion 3 of the SE for ER-157P.

<u>Criterion 4 from the SE for ER-157P</u>

> An applicant that requests a MUR with the upstream flow straightener configuration discussed in Section 3.2.2 [of Reference 13] should provide justification for claimed CheckPlus uncertainty that extends the justification provided in [Reference 22]. Since the [Reference 22] 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 feedwater configurations at Millstone 3 LEFM do not use upstream flow straighteners. This criterion is therefore not applicable to the Millstone 3 LAR.

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 23].

The licensee designed the internal moisture separation equipment to ensure that the moisture carryover does not exceed 0.25 percent by weight (wt percent) for the Millstone 3 Westinghouse Model ‘F’ steam generators. The Millstone 3 uncertainty associated with steam enthalpy due to moisture is 0.066 percent. This value was used in the calculation of the total power measurement uncertainty. As described in ER-764 (Reference 23), since the uncertainty in the measurement of the moisture content is known, the combination of the moisture uncertainty with the other thermal power uncertainties as the root sum squares is justified.

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 Millstone 3.

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 (formerly Caldon) Document ER-369, (Reference 17). The licensee identified the calculated core thermal power uncertainty at Millstone 3 due to the LEFM system is 0.325 percent which is bounded by the proposed power uncertainty of 0.4 percent.

As described in Section 3.1.4 of this SE, the NRC staff evaluated the methodology used to calculate the total core 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 approved topical reports. In Section 3.1.4 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. The planning and execution of these activities fall under site work control processes and procedures.

Controlling Software and Hardware Configuration

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

Performing Corrective Actions

The licensee stated that problems with Millstone 3 plant instrumentation are documented in the Millstone 3 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 DENC corrective action program.

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

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

After reviewing the above statements, the NRC staff finds 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 staff concludes that the licensee has provided the information identified 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 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.

Criterion 1 from the SE for ER-157P is addressed in Section 3.1.5.2 in this SE. The NRC staff evaluated the proposed allowable outage time (48 hours) for the licensee's required actions when the LEFM flow meters is in Maintenance mode to restoring to Normal mode and found that this proposed AOT is acceptable.

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

3.1.6 Conclusion

The NRC staff's evaluation of the identified instrumentation for new power level for Millstone 3 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 1.6 percent RTP is acceptable.

3.2 <u>Accident Analyses</u>

3.2.1 Regulatory Evaluation

The regulations in 10 CFR Part 50, Appendix A, establish minimum requirements (General Design Criteria or GDC) for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The NRC staff identified the following GDCs as being applicable to the review of the transients and accidents in the application:

- GDC-4, Environmental and dynamic effects design bases, insofar as it requires that systems, structures, and components important to safety be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, and that such systems, structures, and components be protected against dynamic effects;

- GDC-10, Reactor design, insofar as it requires that the reactor core 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-15, Reactor coolant system design, insofar as it requires reactor coolant system and certain associated systems must be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operating conditions, including anticipated operational occurrences;

- GDC-16, Containment design, insofar as it requires that the reactor containment be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment;

- GDC-27, Combined reactivity control systems capability, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained;

- GDC-33, Reactor coolant makeup, insofar as it requires to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system is required to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function to maintain coolant inventory during normal reactor operation is accomplished;

- GDC-34, Residual heat removal, insofar as it requires to remove decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded;

- GDC-35, Emergency Core Cooling, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA;

- GDC-38, Containment heat removal, insofar as it requires that a containment heat removal system be provided, and that its function shall be to reduce rapidly the containment pressure and temperature following a loss-of-coolant accident and maintain them at acceptably low levels;

- GDC-50, Containment design basis, insofar as it requires that the containment and its associated heat removal systems be designed so that the containment structure can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure conditions resulting from any loss-of-coolant accident;

The NRC staff identified the following regulatory requirements as being applicable to the review of the Millstone 3 MUR power uprate LAR:

- 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," which, in part, establishes standards for the calculation of emergency core cooling systems accident performance and acceptance criteria for that calculated performance.

- 10 CFR 50.62, "Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled nuclear power plants," which requires, in part, that each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS.

- 10 CFR 50.63, "Loss of all alternating current power," which requires, in part, that the plant withstand and recover from a station black out (SBO) event of a specified duration.

- 10 CFR Part 50, Appendix K, "ECCS Evaluation Models," which, in part, establishes required and acceptable features of evaluation models for heat removal by the Emergency Core Cooling Systems (ECCS) after the blowdown phase of a loss-of-coolant accident.

As discussed above, 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.

3.2.2 Technical Evaluation

As stated by the licensee in the LAR, its approach to modeling the core and/or nuclear steam supply system (NSSS) thermal power in the safety analyses generally fall into one of three categories.

- The analyses in the first category apply a 2 percent increase to the initial power level to account for the power measurement uncertainty. These analyses have not been re-performed for the MUR uprate conditions because the sum of the proposed core power level and the decreased power measurement uncertainty is within the previously analyzed conditions.

- The analyses in the second category employ a nominal core power level. These analyses have either been previously evaluated at, or re-performed for, the proposed power level. The statistical DNBR events were analyzed previously at a core power of 3,712 MWt which bounds the proposed RTP of 3,709 MWt.

- The analyses in the third category are performed at zero percent power conditions or do not model core power level. These analyses have not been re-performed because they are unaffected by the core power level.

A summary of the licensing basis transients and accidents is contained in the table below.

RIS 2002-03 states the following:

> When licensees submit measurement uncertainty recapture power uprate applications, the [NRC] staff intends to use the following general approach for their review:
>
> - In areas (e.g., accident/transient analyses, components, systems) for which the existing analyses of record *do not bound* the plant operation at the proposed uprated power level, the [NRC] staff will conduct a detailed review.
>
> - In areas (e.g., accident/transient analyses, components, systems) for which the existing analyses of record *do bound* plant operation at the proposed uprated power level, the [NRC] staff will not conduct a detailed review.
>
> - In areas that are amenable to generic disposition, the [NRC] staff will utilize such dispositions.

The NRC staff utilized the approach discussed above in its review of the LAR. The NRC staff did not conduct a detailed review of the licensee's analyses that were determined by the licensee to be bounded by the current analysis of record (AOR). For these analyses, the NRC staff determined that existing analyses will continue to bound plant operation after implementation of the proposed MUR power uprate as summarized in Table 1 below.

Section 3.2.2.1 of this SE discusses the licensee's planned implementation of NRC-approved WCAP-17642-P-A, "Westinghouse Performance Analysis and Design Model (PAD5)," (Reference 24) and a new Millstone 3 Large Break Loss of Coolant Accident (LBLOCA) analysis using the Full Spectrum LOCA (FSLOCA) evaluation model. Implementation of WCAP-17642-P-A and the Millstone 3 FSLOCA analysis are planned following the Millstone 3 spring 2022 refueling outage for fuel cycle 22 operation, coincident with the MUR power uprate.

Table 1: Evaluation of Accident and Transient Analyses

Accident / Transient and FSAR Section	Licensee's Bounding Event Determination and/or Reference for NRC-Approved Method	NRC Staff's Review Determination
Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature (FSAR Section 15.1.1)	No specific transient or DNB analysis is performed. Bounded by event in FSAR 15.1.3	The NRC staff finds the licensee's evaluation acceptable because the licensee confirmed that the bounding analysis remains valid at uprated conditions, consistent with the guidance of RIS 2002-03.
Feedwater System Malfunctions that Result in an Increase in Feedwater Flow (FSAR Section 15.1.2)	NRC approval in SPU Reference 4. Subsequent analysis performed with NRC-approved methods (Reference 25) and implemented under the provisions of 10 CFR 50.59.	The NRC staff finds the licensee's evaluation of the transient at hot full power (HFP) conditions acceptable because it is based on an NRC-approved methodology and the overall methodology results in a conservative calculation of DNBR. The NRC staff finds the licensee's justification for the HZP transient to be acceptable because the current licensing basis for this transient is not affected by the MUR uprate.
Excessive Increase in Secondary Steam Flow (FSAR Section 15.1.3)	NRC approval in SPU Reference 4. Subsequent analysis performed with NRC-approved methods (Reference 25) and implemented under the provisions of 10 CFR 50.59.	The NRC staff finds the licensee's evaluation of the transient at uprated conditions acceptable because it is based on NRC-approved methodologies and is evaluated at 101.7% of the RTP. This power level is appropriate because the Virginia Power Statistical DNBR Evaluation Methodology (Reference 27), incorporated via Dominion Energy Core Design and Safety Analysis Methodologies (Reference 25), takes nominal (rather than bounding) power level as an input, and the licensee conservatively uses a power level that is slightly greater than the nominal uprated power level of 3709 MWt.
Inadvertent Opening of a SG Relief or Safety Valve (FSAR Section 15.1.4)	No specific transient or DNB analysis is performed. Bounded by event in FSAR 15.1.5.	Because the MUR uprate does not affect break size, the NRC staff agrees that the steam line break event described in FSAR 15.1.5 will continue to bound inadvertent opening of a SG relief or safety valve at uprated conditions and finds this evaluation acceptable consistent with the guidance of RIS 2002-03.
Steam System	Analysis	The HZP analysis is not affected by the MUR uprate because

Accident / Transient and FSAR Section	Licensee's Bounding Event Determination and/or Reference for NRC-Approved Method	NRC Staff's Review Determination
Piping Failure at Hot Zero Power (HZP) (FSAR Section 15.1.5)	performed with NRC- approved methods (Reference 25) and implemented under the provisions of 10 CFR 50.59.	it is performed at zero power. NRC staff finds this evaluation acceptable because it is based on NRC-approved methodologies and is unaffected by the MUR uprate.
Steam System Piping Failure at Hot Full Power (HFP)	NRC approval in SPU Reference 4. Subsequent analysis performed with NRC-approved methods (Reference 25) and implemented under the provisions of 10 CFR 50.59.	The NRC staff finds the licensee's evaluation of the transient at HFP conditions acceptable because it is based on an NRC-approved methodology and the overall methodology results in a conservative calculation of DNBR. The analysis is evaluated at 101.7 percent of the RTP (3,712 MWt), which does not bound the MUR power of 3,709 MWt plus the calorimetric uncertainty listed in Section 2.3 of Attachment 4 to the LAR. NRC staff reviewed the most recently reviewed and approved analysis of this transient and noted that the calculated peak linear heat rate was 21.0 kW/ft, compared to the limit of 22.6 kW/ft (SPU License Amendment (Reference 4). Based on the margin to the peak linear heat generation rate limit present in this analysis, the NRC staff finds it unlikely that an increase in power of 2 percent, bounding the 1.6 percent uprate plus the calorimetric uncertainty, will lead to the peak linear generation rate exceeding its limit. NRC staff finds the licensee's evaluation acceptable because the available information provides reasonable assurance that the fuel centerline temperature limit will be met.
Steam System Piping Failure (Radiological Consequences) (FSAR Section 15.1.5.4)	NRC approval in SPU Reference 4.	
Loss of External Electrical Load (FSAR Section 15.2.2)	Bounded by event in FSAR 15.2.3.	This transient is bounded by the turbine trip transient described in FSAR section 15.2.3 which results in a more rapid reduction in steam flow. The licensee's determination is based on valve closure times, which are not affected by the MUR uprate. The NRC staff finds this acceptable because the bounding analysis remains valid at uprated conditions, consistent with the guidance of RIS 2002-03.
Turbine Trip (FSAR Section 15.2.3)	NRC approval in SPU Reference 4. Subsequent analysis performed with NRC-approved methods (Reference 25 and Reference 26) and implemented	The NRC staff finds the licensee's evaluation for meeting the RCS and MSS pressure criteria acceptable because it is based on an NRC-approved methodology and is evaluated at 102% of the RTP, which bounds the MUR power of 3,709 MWt plus the calorimetric uncertainty listed in Section 2.3 of Attachment 4 to the LAR. The NRC staff finds the licensee's evaluation for meeting the DNBR criterion acceptable because it is based on an NRC-approved methodology and is evaluated at 101.7% of the RTP. This power level is appropriate because the Virginia Power Statistical DNBR Evaluation Methodology (Reference 27) takes nominal (rather than bounding) power level as an input,

Accident / Transient and FSAR Section	Licensee's Bounding Event Determination and/or Reference for NRC-Approved Method	NRC Staff's Review Determination
	under the provisions of 10 CFR 50.59.	and the licensee conservatively uses a power level that is slightly greater than the nominal uprated power level of 3709 MWt.
Inadvertent Closure of Main Steam Isolation Valves (FSAR Section 15.2.4)	No specific transient or DNB analysis is performed. Bounded by event in FSAR 15.2.3.	This transient is bounded by the turbine trip event described in FSAR section 15.2.3. The bounding event determination remains valid at uprated conditions. The NRC staff finds this evaluation acceptable because the bounding event determination remains valid at uprated conditions consistent with the guidance of RIS 2002-03.
Loss of Condenser Vacuum and Other Events Resulting in TT (FSAR Section 15.2.5)	No specific transient or DNB analysis is performed. Bounded by event in FSAR 15.2.3.	The transient is bounded by the turbine trip transient described in FSAR section 15.2.3. The bounding event determination remains valid at uprated conditions. Also as indicated in FSAR section 15.2.5, the adverse effects resulting from loss of condenser vacuum are already assumed to have occurred in the analysis in FSAR section 15.2.3. The licensee also considered other events that could initiate a turbine trip in FSAR section 15.2.3. Because this is not affected by the MUR uprate, NRC staff agrees that the bounding event determination remains valid at uprated conditions and finds this evaluation acceptable consistent with the guidance of RIS 2002-03.
Loss of Non-Emergency Alternating Current (AC) Power to the Station Auxiliaries (FSAR Section 15.2.6)	Bounded by events in FSAR 15.2.3, 15.2.7, and 15.3.2.	As stated in the LAR, overpressure of both the primary and secondary side are bounded by the turbine trip event in FSAR section 15.2.3; pressurizer fill is bounded by the loss of normal feedwater flow event described in FSAR section 15.2.7, and the DNB aspects are bounded by the complete loss of flow event described in FSAR section 15.3.2. The NRC staff finds this acceptable because the licensee confirmed that the bounding analysis remains valid at uprated conditions, consistent with the guidance of RIS 2002-03.
Loss of Normal Feedwater Flow (FSAR Section 15.2.7)	NRC approval in SPU Reference 4.	The NRC staff finds the licensee's evaluation acceptable because it is based on an NRC-approved methodology and is evaluated at 102% of the RTP, which bounds the MUR power of 3,709 MWt plus the calorimetric uncertainty listed in Section 2.3 of Attachment 4 to the LAR. This analysis was implemented in support of the Millstone 3 SPU approved by NRC (Reference 4).
Feedwater System Pipe Break (FSAR Section 15.2.8)	NRC approval in SPU Reference 4.	The licensee demonstrated core integrity by showing that bulk boiling does not occur and the core remains covered throughout the transient. The NRC staff finds the licensee's evaluation acceptable as it is based on an NRC-approved methodology and evaluated at 102% of RTP, which bounds the MUR power of 3,709 MWt plus the calorimetric uncertainty listed in Section 2.3 of the LAR. This analysis was implemented in support of the SPU License Amendment Millstone 3 SPU approved by NRC (Reference 4).
Partial Loss of	NRC approval in	

Accident / Transient and FSAR Section	Licensee's Bounding Event Determination and/or Reference for NRC-Approved Method	NRC Staff's Review Determination
Forced Reactor Coolant Flow (FSAR Section 15.3.1)	SPU Reference 4. Subsequent analysis performed with NRC-approved methods (Reference 25) and implemented under the provisions of 10 CFR 50.59.	In a letter dated June 2, 2021 (Reference 2), the licensee stated that since the uprate has a negligible effect on the local core conditions (temperature, pressure, and flow), the DNB calculations were performed using these conditions at a conservative power level of 3712 MWt (101.7% RTP) for comparison with the DNB acceptance criteria. This power level is appropriate because the Virginia Power Statistical DNBR Evaluation Methodology (Reference 27) takes nominal (rather than bounding) power level as an input, and the licensee conservatively uses a power level that is slightly greater than the nominal uprated power level of 3709 MWt. The licensee's approach is summarized in the FSAR Section 15.0.3.1. The NRC staff finds the licensee's evaluation of the transient acceptable because it is based on an NRC-approved methodology and the overall methodology results in a conservative calculation of DNBR. The licensee's analysis of this transient and core thermal limits were implemented under the provisions of 10 CFR 50.59 in support of the Millstone 3 methods transition approved by NRC (Reference 25).
Complete Loss of Forced Reactor Coolant Flow (FSAR Section 15.3.2)	NRC approval in SPU Reference 4. Subsequent analysis performed with NRC-approved methods (Reference 25) and implemented under the provisions of 10 CFR 50.59.	
RCP Shaft Seizure (Locked Rotor) (FSAR Section 15.3.3)	NRC approval in SPU Reference 4. Subsequent analysis performed with NRC-approved methods (Reference 25 and Reference 26) and implemented under the provisions of 10 CFR 50.59.	In a letter dated June 2, 2021 (Reference 2), the licensee stated that since the uprate has a negligible effect on the local core conditions (temperature, pressure, and flow), the DNB calculations were performed using these conditions at a conservative power level of 3712 MWt (101.7% RTP) for comparison with the DNB acceptance criteria. This power level is appropriate because the Virginia Power Statistical DNBR Evaluation Methodology (Reference 27) takes nominal (rather than bounding) power level as an input, and the licensee conservatively uses a power level that is slightly greater than the nominal uprated power level of 3709 MWt. The licensee's approach is summarized in the FSAR Section 15.0.3.1. The NRC staff finds the licensee's evaluation of the transient with respect to DNB acceptable because it is based on an NRC-approved methodology and the overall methodology results in a conservative calculation of DNBR. The NRC staff finds the licensee's evaluation of the transient with respect to RCS pressure, PCT, and the zirconium-steam reaction acceptable because it is based on NRC-approved methodologies and is evaluated at 102% of the RTP, which bounds the MUR power of 3,709 MWt plus the calorimetric uncertainty listed in Section 2.3 of Attachment 4 to the LAR. The licensee's analysis was implemented under the

Accident / Transient and FSAR Section	Licensee's Bounding Event Determination and/or Reference for NRC-Approved Method	NRC Staff's Review Determination
		provisions of 10 CFR 50.59 in support of the Millstone 3 methods transition approved by NRC (Reference 25).
RCP Shaft Seizure (Locked Rotor Radiological Consequences) (FSAR Section 15.3.3.4)	NRC approval in SPU Reference 4.	The licensee's evaluation of this transient is described in Section II.1.14 of Attachment 4 to the LAR. The licensee stated that this transient is bounded by the RCP shaft seizure transient described in FSAR section 15.3.3, and that the bounding event determination remains valid at uprated conditions. In section 15.3.4 of the FSAR, the licensee stated that the RCP shaft seizure event is bounding because coolant flow decreases faster for this event in the initial stages of the transient. Because this would not be affected by the MUR uprate, NRC staff agrees that the bounding event determination remains valid at uprated conditions and finds this evaluation acceptable consistent with the guidance of RIS 2002-03.
RCP Shaft Break (FSAR Section 15.3.4)	Bounded by event in FSAR 15.3.3.	
Uncontrolled RCCA Bank Withdrawal from a Subcritical or Low Power Startup Condition (FSAR Section 15.4.1)	Analysis performed with NRC- approved methods (Reference 25) and implemented under the provisions of 10 CFR 50.59.	Because the accident analysis is initiated at zero power, the analysis is not affected by the MUR uprate. The NRC staff finds the licensee's evaluation of the transient at uprated conditions acceptable because it is based on NRC-approved methodologies and because the analysis is not affected by the MUR uprate. The licensee's analysis of this transient and core thermal limits were implemented under the provisions of 10 CFR 50.59 in support of the Millstone 3 methods transition approved by NRC (Reference 25).
Uncontrolled RCCA Bank Withdrawal at Power (FSAR Section 15.4.2)	Analysis performed with NRC-approved methods (Reference 25) and implemented under the provisions of 10 CFR 50.59.	This analysis ensures that the high neutron flux and overtemperature delta-T trips provide adequate protection against DNB and fuel centerline melt over a range of reactivity insertion rates and power levels. In its evaluation, the licensee has confirmed that operation at uprated conditions is acceptable with no change to these protection setpoints. The NRC staff finds the licensee's evaluation of the fuel centerline temperature transient at uprated conditions acceptable because it is based on NRC-approved methodologies and is evaluated at 102% of the RTP, which bounds the MUR power of 3,709 MWt plus the calorimetric uncertainty listed in Section 2.3 of Attachment 4 to the LAR. The NRC staff also finds the DNBR analysis acceptable because it is based on NRC-approved methodologies and performed at 101.7% RTP. This power level is appropriate because the Virginia Power Statistical DNBR Evaluation Methodology (Reference 27) takes nominal (rather than bounding) power level as an input, and the licensee conservatively uses a power level that is slightly greater than the nominal uprated power level of 3709 MWt. The licensee's analysis was implemented under the provisions of 10 CFR 50.59 in support of the Millstone 3 methods transition approved by NRC (Reference 25).

Accident / Transient and FSAR Section	Licensee's Bounding Event Determination and/or Reference for NRC-Approved Method	NRC Staff's Review Determination
RCCA Misalignment (FSAR Section 15.4.3)	Analysis performed with NRC- approved methods (Reference 25 and Reference 28) and implemented under the provisions of 10 CFR 50.59.	This analysis was implemented under the provisions of 10 CFR 50.59 in support of the Millstone 3 methods transition approved by NRC (Reference 25). The NRC staff finds these evaluations acceptable because these analyses assume core power levels that are appropriate for uprated conditions, and because they were performed using NRC-approved methodologies.
CVCS Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (Boron Dilution) (FSAR Section 15.4.6)	Analysis performed with NRC- approved methods (Reference 25) and implemented under the provisions of 10 CFR 50.59.	The licensee's analysis was performed at 101.7 percent of the RTP, which does not bound the MUR power of 3,709 MWt plus the calorimetric uncertainty listed in Section 2.3 of Attachment 4 to the LAR. However, the licensee stated that the analysis is not sensitive to an increase to 102 percent of the RTP. The reactivity response to the boron dilution event would be driven primarily by the change in boron concentration, and the insignificant change in coolant temperature due to an increase of 0.3 percent in power would not have a significant effect on the reactivity transient. Therefore, NRC staff agrees with the licensee's determination that the analysis result will not be sensitive to a 0.3 percent increase in power and finds the licensee's evaluation acceptable. The licensee's analysis was implemented under the provisions of 10 CFR 50.59 in support of the Millstone 3 methods transition approved by NRC (Reference 25).
Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (FSAR Section 15.4.7)	Precluded by administrative fabrication and loading procedures.	The licensee described surveillance tests that are performed to detect fuel assembly enrichment errors or loading errors, noting that tests are performed at defined power levels during power ascension following a refueling outage. The licensee explained that the FSAR presents representative cases to demonstrate the high likelihood of detecting core loading errors with the incore flux mapping program, but this is not repeated for individual core loadings. The NRC staff finds this evaluation acceptable because this analysis is not sensitive to the RTP and will not be affected by the MUR power uprate.
Spectrum of RCCA Ejection Accidents (FSAR Section 15.4.8)	NRC approval in SPU Reference 4.	The licensee analyzed this transient at HFP and HZP conditions. The NRC staff finds the licensee's justification for the evaluation at HZP conditions acceptable because the current licensing basis for this transient is not affected by the MUR uprate. The NRC staff finds the licensee's evaluation of the transient at HFP conditions acceptable because it is based on an NRC-approved methodology and is evaluated at 102% of the RTP, which bounds the MUR power of 3,709 MWt plus the calorimetric uncertainty listed in Section 2.3 of Attachment 4 to the LAR.
Spectrum of RCCA Ejection Accidents (Radiological	NRC approval in SPU Reference 4.	

Accident / Transient and FSAR Section	Licensee's Bounding Event Determination and/or Reference for NRC-Approved Method	NRC Staff's Review Determination
Consequences) (FSAR Section 15.4.8.4)		
Inadvertent Operation of the ECCS During Power Operation (FSAR Section 15.5.1)	Analysis performed with NRC-approved methods (Reference 25) and implemented under the provisions of 10 CFR 50.59.	The NRC staff finds the licensee's evaluation acceptable because it is based on an NRC-approved methodology and is evaluated at 102% of the RTP, which bounds the MUR power of 3,709 MWt plus the calorimetric uncertainty listed in Section 2.3 of Attachment 4 to the LAR. The licensee's analysis was implemented under the provisions of 10 CFR 50.59 in support of the Millstone 3 methods transitions approved by NRC (Reference 25).
CVCS Malfunction that Increases Reactor Coolant Inventory (FSAR Section 15.5.2)	NRC approval in SPU Reference 4.	The NRC staff finds the licensee's evaluation acceptable because it is based on an NRC-approved methodology and is evaluated at 102% of the RTP, which bounds the MUR power of 3,709 MWt plus the calorimetric uncertainty listed in Section 2.3 of Attachment 4 to the LAR. The licensee's analysis was implemented in support of the Millstone 3 SPU approved by NRC (Reference 4).
Inadvertent Opening of a Pressurizer Safety or Relief Valve (FSAR Section 15.6.1)	NRC approval in SPU Reference 4. Subsequent analysis performed with NRC-approved methods (Reference 25) and implemented under the provisions of 10 CFR 50.59.	In a letter dated June 2, 2021 (Reference 2), the licensee stated that since the uprate has a negligible effect on the local core conditions (temperature, pressure, and flow), the DNB calculations were performed using these conditions at a conservative power level of 3712 MWt (101.7% RTP) for comparison with the DNB acceptance criteria. This power level is appropriate because the Virginia Power Statistical DNBR Evaluation Methodology (Reference 27) takes nominal (rather than bounding) power level as an input, and the licensee conservatively uses a power level that is slightly greater than the nominal uprated power level of 3709 MWt. The licensee's approach is summarized in the FSAR Section 15.0.3.1. The NRC staff finds the licensee's evaluation of the transient acceptable because it is based on an NRC-approved methodology and the overall methodology results in a conservative calculation of DNBR. The licensee's analysis was implemented under the provisions of 10 CFR 50.59 in support of the Millstone 3 methods transition approved by NRC (Reference 25).
Failure of Small Lines Carrying Primary Coolant Outside Containment (Radiological Consequences)	NRC approval in SPU Reference 4.	The NRC staff finds the licensee's evaluation acceptable because it is based on an NRC-approved methodology and the analysis of the offsite dose from a small line break outside containment (FSAR Section 15.6.2) was performed at 3723 MWt, which is 102% of 3650 MWt and bounds the MUR power of 3,709 MWt plus the calorimetric uncertainty listed in Section 2.3 of Attachment 4 to the LAR. Therefore, the

Accident / Transient and FSAR Section	Licensee's Bounding Event Determination and/or Reference for NRC-Approved Method	NRC Staff's Review Determination
(FSAR Section 15.6.2)		current analysis for small line break outside containment remains bounding for the MUR power uprate.
Steam Generator Tube Failure (FSAR Section 15.6.3)	NRC approval in SPU Reference 4. Subsequent analysis performed with NRC-approved methods (Reference 29) and implemented under the provisions of 10 CFR 50.59.	The NRC staff finds the licensee's evaluation of the transient at uprated conditions acceptable because it is based on NRC-approved methodologies and is evaluated at 102% of the RTP, which bounds the MUR power of 3,709 MWt plus the calorimetric uncertainty listed in Section 2.3 of Attachment 4 to the LAR. The licensee's analysis was implemented in support of the Millstone 3 SPU approved by NRC (Reference 4), and supplemented with an additional evaluation for an issue described in NSAL 07-11 under the provisions of 10 CFR 50.59. NRC staff reviewed NSAL 07-11 and concluded that it does not affect the staff finding on the acceptability of the MUR uprate.
Steam Generator Tube Failure (Radiological Consequences) (FSAR Section 15.6.3.2.2)	NRC approval in SPU Reference 4.	
Best Estimate (BE) Large Break Loss of Coolant Accident (LBLOCA) (FSAR Section 15.6.5.2)	NRC approval in SPU Reference 4.	The licensee's evaluation of the BE LBLOCA is described in Section III.2 of Attachment 4 to the LAR. The licensee stated that the current licensing analysis uses the Automated Statistical Treatment of Uncertainty Method (ASTRUM) evaluation model as approved in the SPU license amendment (Reference 4). The BE LBLOCA analysis assumed a power level of 3,650 MWt and sampled over a 2% calorimetric uncertainty. The licensee stated that an evaluation of the effects of modeling thermal conductivity degradation (TCD) and peaking factor burndown was performed subsequent to the analysis of record. The TCD evaluation resulted in a peak cladding temperature (PCT) penalty and a PCT change from additional post-AOR evaluations. Since the proposed MUR involves an increase to the analyzed power level, the BE LBLOCA analysis is an accident not bounded by the proposed uprated power level. As provided in Section III.2-1 and Table III-1 of Attachment 4 to the LAR, the licensee provided its methodology for demonstrating that no reanalysis is required for the BE LBLOCA analysis of record using the ASTRUM evaluation model. The NRC staff finds the MUR evaluation method for this accident is acceptable as the licensee has determined that a reduction in peaking factors fully offsets the impact of the MUR power increase on the MPS3 ASTRUM BE LBLOCA AOR. Also, the licensee demonstrated there is sufficient margin maintained for the 10

Accident / Transient and FSAR Section	Licensee's Bounding Event Determination and/or Reference for NRC-Approved Method	NRC Staff's Review Determination
		CFR 50.46 PCT, maximum local oxidation, and core-wide oxidation acceptance criteria. The licensee also intends to implement the MPS3 FSLOCA analysis for fuel cycle 22 operation, coincident with the MUR power uprate. The use of the FSLOCA evaluation model at MPS3 was submitted in a separate LAR submittal dated November 5, 2020 (Reference 30). As stated in the LAR, the MPS3 FSLOCA analysis supports the MUR power level as discussed in Section 3.2.2.1 of this SE. The NRC approved the FSLOCA LAR in License Amendment No. 279 dated October 5, 2021 (Reference 31).
Small Break Loss of Coolant Accident (SBLOCA) (FSAR Section 15.6.5.3)	NRC approval in SPU Reference 4.	The licensee's evaluation of this transient is described in Section II.1.25 of Attachment 4 to the LAR. The licensee stated that the current licensing analysis for MPS3 is based on the Westinghouse SBLOCA evaluation model with the NOTRUMP code (References 32, 33, and 34), uses the 10 CFR Part 50, Appendix K methodology with an RTP of 3,650 MWt and applies a 2% calorimetric power measurement uncertainty to the RTP. The NRC staff finds the licensee's evaluation of the transient at uprated conditions acceptable because it is based on NRC-approved methodologies and is evaluated at 102 percent of the RTP, which bounds the MUR power of 3,709 MWt plus the calorimetric uncertainty listed in Section 2.3 of Attachment 4 to the LAR. The licensee's analysis was implemented in support of the MPS3 SPU approved by NRC (Reference 4). The licensee intends to implement NRC-approved WCAP-17642-P-A (PAD5) (Reference 24) for fuel cycle 22 operation, coincident with the MUR power uprate; the impact of the PAD5 implementation on the MPS3 SBLOCA NOTRUMP analysis will be assessed by the licensee as discussed in Section 3.2.2.1 of this SE.
Loss of Coolant Accident (Radiological Consequences) (FSAR Section 15.6.5.4)	NRC approval in SPU Reference 4.	As stated in the LAR, the LOCA event analysis is based upon the alternative source term (AST) as defined in RG 1.183 (Reference 35) with acceptance criteria as specified in either 10 CFR 50.67 or RG 1.183. MPS3 TS 5.3.1 restricts the fuel enrichment to 5.0 weight percent U-235, which is not being changed as a result of the MUR power increase. Fuel assembly exposure is restricted to a lead rod burnup of 62,000 MWD/MTU. As a result of the limited changes in core power, burnup history, and enrichment associated with the MUR Power Uprate, the source term core inventory will remain bounded by the values at the analyzed core power level of 3723 MWt, which is 102% of 3650 MWt. The NRC staff finds the licensee's evaluation of the transient at uprated conditions acceptable because it is based on NRC-approved methodologies and is evaluated at 102 percent of the RTP, which bounds the MUR power of

Accident / Transient and FSAR Section	Licensee's Bounding Event Determination and/or Reference for NRC-Approved Method	NRC Staff's Review Determination
		3,709 MWt. The licensee's analysis was implemented in support of the Millstone 3 SPU approved by NRC (Reference 4).
Radioactive Gaseous Waste System Failure (FSAR Section 15.7.1/ 11.3.3)	NRC approval in Reference 36. Subsequent analysis revisions implemented under the provisions of 10 CFR 50.59.	The waste gas decay tank rupture analysis was part of the original plant licensing basis. To bound the SPU to 3723 MWt, which is 102% of 3650 MWt , each of the isotopes were scaled based on factors determined from the ratio of the primary activity concentrations; therefore, the current analysis bounds the MUR power uprate. The NRC's staff's evaluation of the proposed MUR power uprate on design basis accidents radiological consequence analyses, including the area of radioactive gaseous waste system failure is provided in Section 3.6.2.2.2 of this SE.
Radioactive Liquid Waste System Leak or Failure (Atmospheric Release) (FSAR Section 15.7.2/ 11.2.3.1)	NRC approval in Reference 36. Subsequent analysis revisions implemented under the provisions of 10 CFR 50.59.	The FSAR 15.7.2 and 15.7.3 accidents are defined as an unexpected and uncontrolled postulated rupture of the boron recovery tank which can result in an atmospheric release, as well as a liquid release to the groundwater. The boron recovery tank rupture was part of the original plant licensing basis. In order to bound the SPU to 3723 MWt, which is 102% of 3650 MWt, each of the isotopes were scaled based on factors determined from the ratio of the primary activity concentrations; therefore the current analysis bounds the MUR power uprate. The NRC's staff's evaluation of the proposed MUR power uprate on design basis accidents radiological consequence analyses, including the area of radioactive liquid waste system leak or failure is provided in Section 3.6.2.2.2 of this SE.
Liquid Containing Tank Failure (FSAR Section 15.7.3/ 11.2.3.2)	NRC approval in Reference 36. Subsequent analysis revisions implemented under the provisions of 10 CFR 50.59.	
Design Basis Fuel Handling Accidents (FHA) (FSAR Section 15.7.4)	NRC approval in SPU Reference 4.	As stated in the LAR, the FHA is based upon the AST as defined in RG 1.183 (Reference 35) with acceptance criteria as specified in either 10 CFR 50.67 or RG 1.183. The current FHA dose evaluation was performed using the core inventory described above for the LOCA at 3723 MWt (102% of 3650 MWt), with an assumption of one failed fuel assembly plus 19 fuel rods in the impacted assembly. Therefore, the FHA evaluation remains bounding for the MUR power uprate. No changes to the number of failed fuel rods or assumed radial peaking factor are required to support the MUR. The NRC's staff's evaluation of the proposed MUR power uprate on design basis accidents radiological consequence analyses, including the fuel handling accident is provided in Section 3.6.2.2.2 of this SE.
Natural Circulation Cooldown (FSAR	NRC approval in SPU Reference 4.	The current natural circulation cooldown analysis was performed at 3,650 MWt in support of a previous SPU (Reference 4). The power level assumed in the analysis does not bound the MUR uprated power of 3,709 MWt. In

Accident / Transient and FSAR Section	Licensee's Bounding Event Determination and/or Reference for NRC-Approved Method	NRC Staff's Review Determination
Section 15.2.6, 15.2.7, 15.2.8, 15.3.2)		support of the MUR uprate, the licensee repeated the current licensing analysis using input parameters that reflect a power level of 3,723 MWt. The analysis at uprated conditions meets the acceptance criteria for this analysis. Specifically, it indicates that natural circulation cooling can be established, adequate boron mixing can be achieved, and the system can be cooled and depressurized to conditions that allow initiation of the residual heat removal system in less than the time assumed in the radiological consequence analysis. Because this analysis is performed using a methodology that was previously approved by NRC staff for this application, meets the acceptance criteria, and is evaluated at a power level that bounds the MUR power, NRC staff finds this evaluation acceptable.
Short-Term LOCA Mass and Energy (M&E) Releases (FSAR Section 6.2.1.2)	NRC approval in SPU Reference 4.	The licensee noted in section VI.1.B.iii.a of Attachment 4 to the LAR that the impact of the MUR power uprate on this analysis can be assessed by comparing mass and energy release rates at uprated conditions to release rates at pre-uprate conditions. Double-ended rupture of the pressurizer surge line, pressurizer spray line, and single-ended split of the main feedwater line were considered in this analysis. Double-ended rupture of the residual heat removal line was bounded by double-ended rupture of the pressurizer surge line and was not explicitly considered. The need to consider large primary reactor coolant piping was removed from the design basis of the containment sub-compartments following approval of the leak-before-break methodology for Millstone 3. The licensee provided design conditions in Table 3.0-1 of Attachment 4 to the LAR that reflect operation at 102 percent RTP, which bounds the MUR power level plus the calorimetric uncertainty listed in Section 2.3 of Attachment 4 to the LAR. In section II.1.27.a, the licensee stated that changing conditions result in a small benefit over the current analysis of record. NRC staff notes that while the increase in hot leg temperature presents a small benefit, the decreasing design cold leg temperature without changing the analysis cold leg temperature is a reduction in margin from -4 °F to -3.3 °F for the pressurizer spray line break. The staff finds that (1) the conditions used in the accident analysis are still conservative relative to the design conditions for the MUR power uprate, (2) the design conditions are calculated at a power that bounds the MUR uprated power, and (3) the analyses are performed with the same methodology that was previously approved. Therefore, the NRC staff finds this analysis acceptable.
M&E Release Analyses for	NRC approval in Reference 37	In section 6.2.1.4.3 of the FSAR, the licensee noted that the

Accident / Transient and FSAR Section	Licensee's Bounding Event Determination and/or Reference for NRC-Approved Method	NRC Staff's Review Determination
Postulated LOCAs (FSAR Section 6.2.1.3)	and supplemented by reanalysis implemented under the provisions of 10 CFR 50.59.	analysis is performed at a range of power levels, up to and including 102% of the RTP. The licensee also noted that initial conditions are selected to reflect the initial power for that case and appropriate uncertainties. The NRC staff finds the licensee's evaluation acceptable because it is based on an NRC-approved methodology and is evaluated at 102% of the RTP, which bounds the MUR power of 3,709 MWt plus the calorimetric uncertainty listed in Section 2.3 of Attachment 4 to the LAR. The licensee's analysis was implemented in support of the Millstone 3 SPU approved by NRC (Reference 4).
M&E Release Analyses for Postulated Secondary System Pipe Rupture Inside Containment (FSAR Section 6.2.1.4)	NRC approval in SPU Reference 4.	
Large Break LOCA Long-Term Cooling (FSAR Section 6.3)	NRC approval in SPU Reference 4. The analyses have been supplemented by additional evaluations under the provisions of 10 CFR 50.59.	The licensee performed this analysis to ensure that the core remains subcritical and the core geometry remains coolable following a large break loss-of-coolant accident. In order to keep the core subcritical, the licensee calculated a minimum containment sump boron concentration. This calculation is performed using the licensee's core reload methodology, which explicitly accounts for reactor power. Because the method has been previously approved by the NRC and will reflect the uprated power when reload calculations are performed, the NRC staff finds this acceptable. The calculation of hot-leg switchover time was previously performed at a core power of 3,723 MWt, which is 102 percent of the current RTP. This analysis was previously approved by NRC staff (Reference 4), and supplemented by an additional evaluation for the issue described in NSAL 09-6. NRC staff reviewed this NSAL for information relevant to the MUR uprate. The NRC staff finds this portion of the analysis acceptable because it is based on NRC-approved methodologies and is evaluated at 102 percent of the RTP, which bounds the MUR power of 3,709 MWt plus the calorimetric uncertainty listed in Section 2.3 of the LAR. NRC staff has reviewed the requirements for net positive suction head (NPSH) of containment recirculation pumps, safety injection pumps, charging pumps, and residual heat removal pumps. The NRC staff finds that these pumps would have sufficiently large NPSH margins under the MUR conditions because there are large margins in the current NPSH analysis for these pumps at 102 percent RTP.

Accident / Transient and FSAR Section	Licensee's Bounding Event Determination and/or Reference for NRC-Approved Method	NRC Staff's Review Determination
Anticipated Transient Without SCRAM (ATWS) (FSAR Section 15.8)	NRC approval in SPU Reference 4. Subsequent analysis performed with NRC-approved methods (Reference 38) and implemented under the provisions of 10 CFR 50.59.	In the evaluation, the licensee stated that generic Westinghouse ATWS analyses are reviewed for their continued applicability to Millstone 3 at uprated conditions. The NRC staff finds the licensee's evaluation acceptable because it is based on the methodology used in the current licensing basis, evaluated at 102% of the RTP, which bounds the MUR power of 3,709 MWt plus the calorimetric uncertainty listed in Section 2.3 of Attachment 4 to the LAR. The licensee's analysis was implemented in support of the Millstone 3 SPU approved by NRC (Reference 4) and updated under the provisions of 10 CFR 50.59.
Station Blackout (SBO) (FSAR Section 8.1.8))	NRC approval in SPU Reference 4.	The licensee's evaluation of this transient is described in section V.1.B of Attachment 4 to the LAR. In its evaluation, the licensee noted whether portions of the SBO analysis are affected by the MUR uprate, and confirmed that portions affected by the uprate remain bounding. The licensee indicated the usable condensate inventory for decay heat removal and plant cooldown exceeds the amount required to ensure safe shutdown during the coping duration. Current analyses for effects of loss of ventilation for the turbine driven auxiliary feedwater (AFW) pump room, main steam valve building, and charging pump cubicle remain bounding for uprated conditions. The licensee also confirmed that the turbine driven AFW pump has sufficient capacity to remove decay and sensible heat present in a SBO scenario at MUR uprate conditions. Coping duration, required equipment, alternate AC power sources, class 1E battery capacity, emergency lighting, communications, and heat tracing portions of the analysis are unaffected by the MUR uprate. Because portions of the analysis that are affected by the MUR uprate remain bounding, NRC staff finds this evaluation acceptable.
Analyses to Determine Environmental Qualification Parameters (FSAR Section 3.11)	NRC approval in SPU Reference 4.	The MUR conditions for temperature, pressure, radiation, and humidity and related EQ parameters are bounded by the AORs that are the bases for the existing EQ program requirements. Therefore, the NRC staff finds that the electrical equipment will continue to meet the requirements of 10 CFR 50.49 following implementation of the proposed MUR power uprate. The NRC's staff's evaluation of environmental qualification parameters is provided in Section 3.4.2.8 of this SE.
Safe Shutdown Fire Analysis (Appendix R)	NRC approval in SPU Reference 4.	As stated in the LAR, the proposed MUR power uprate has no impact upon the capability to place Millstone 3 in fire safe shutdown, as defined in the branch technical position (BTP) 9.5-1 compliance documentation for the plant. The NRC's staff's evaluation of safe shutdown fire analysis is provided in Section 3.6.3 of this SE.

Accident / Transient and FSAR Section	Licensee's Bounding Event Determination and/or Reference for NRC-Approved Method	NRC Staff's Review Determination
Spent Fuel Pool Cooling (FSAR Section 9.1.3)	NRC approval in SPU Reference 4.	As stated in the LAR, the loss of SFP cooling analysis is bounding for the MUR Power Uprate. The SFP heat load assumptions included in the Millstone 3 FSAR remain bounding for the proposed MUR power uprate. The NRC's staff's evaluation of spent fuel pool cooling and purification system is provided in Section 3.5.4.2 of this SE.
Internal Flooding (FSAR Section 10.4.5)	NRC approval in SPU Reference 4.	As stated in the LAR, the proposed MUR power uprate does not affect the internal flooding due to failure of non-seismic Category 1 tanks and vessels, and there is no additional leakage from these sources that would affect the associated equipment and floor drain systems. The proposed MUR power uprate does not affect the analyses and design features related to internal flooding due to a circulating water pipe rupture or expansion joint failure. The NRC's staff's evaluation of internal flooding is provided in Section 3.5.7 of this SE.
Vital Area Doses and Shielding (FSAR Section 12.3)	NRC approval in SPU Reference 4.	As stated in the LAR, FSAR Section 12.3 discusses dose consequences associated with vital area access. The associated analysis was performed in accordance with NUREG-0737, Action Item II.B.2, (Reference 39) in order to ensure personnel accessibility after a DBA. The DBA considered for the evaluation of vital areas was the LOCA. The applicable evaluation is based on a core power level of 3723 MWt, which bounds the proposed MUR power uprate. The NRC's staff's evaluation of the proposed MUR power uprate on design basis accidents radiological consequence analyses, including the vital area doses and shielding topic is provided in Section 3.6.2.2.2 of this SE.

3.2.2.1 Planned Analysis Revisions

Westinghouse Performance Analysis and Design Model (PAD5) Implementation

In its LAR, the licensee stated that they intend to implement NRC-approved WCAP-17642-P-A, "Westinghouse Performance Analysis and Design Model (PAD5)," (Reference 24) for fuel cycle 22 operation, coincident with the MUR power uprate. PAD5 implementation involves an LAR submittal, and subsequent NRC approval, to revise TS 2.1.1.2, "Reactor Core Safety Limit," to reflect the fuel melt temperature approved in Reference 24. The licensee's LAR submittal for the proposed PAD5 implementation was submitted on December 8, 2020 (Reference 40) and is under NRC staff review. The implementation of PAD5 will also involve the licensee's reanalysis of select non-LOCA Chapter 15 events at the MUR uprated power level. As indicated in Section III.4-1 of Attachment 4 to the LAR, the events identified for reanalysis to support PAD5 implementation at the MUR uprated power include:

- FSAR Section 15.2.2- Loss of External Load, DNB Case
- FSAR Section 15.2.3- Turbine Trip, DNB Case

- FSAR Section 15.2.6- Loss of Non-Emergency AC Power to the Station Auxiliaries
- FSAR Section 15.2.7- Loss of Normal Feedwater Flow
- FSAR Section 15.2.8- Feedwater System Pipe Break
- FSAR Section 15.4.8- Spectrum of Rod Cluster Control Assembly Ejection Accidents

As presented in Table 1 above, the NRC staff has determined that the current AOR for these specific events support the MUR power uprate. In its LAR, the licensee stated that prior to PAD5 implementation, it will perform a review of each reanalysis and submit them to the NRC for review if it is deemed necessary per the criteria of 10 CFR 50.59. The licensee also stated that with exception of rod ejection, reanalysis of the above listed events will be performed in accordance with the Dominion Energy reload method in VEP-FRD-42-A, “Reload Nuclear Design Methodology” (Reference 41) or the Westinghouse RETRAN transient analysis method in WCAP-14882-P-A (Reference 26). The PAD5 rod ejection reanalysis was completed by the licensee using the methodology of WCAP-7588, Revision 1-A (Reference 42). The completed PAD5 rod ejection analysis was shown to meet all applicable acceptance criteria at the MUR power level. The NRC-approved adoption of Dominion Energy Core Design and Safety Analysis methodologies for MPS3 is provided in Reference 25. Since the analyses revisions would reflect the MUR power level, as identified in the above table for the current analyses of record and because the licensee is obligated to follow appropriate regulatory change processes consistent with the nature of the planned changes, NRC staff finds the licensee’s approach for PAD5 implementation coincident with the MUR power uprate acceptable.

<u>Small Break Loss of Coolant Accident</u>

In its LAR, the licensee stated that they will perform an assessment of the WCAP-17642-P-A (PAD5) (Reference 24) impact on the MPS3 SBLOCA NOTRUMP code analysis in accordance with the 10 CFR 50, Appendix K methodology. As presented in Table 1 above, the NRC staff determined that the current AOR for this specific event supports the MUR power uprate as it is based on NRC-approved methodologies and is evaluated at 102 percent of the RTP, which bounds the MUR power of 3,709 MWt. The NRC staff finds the licensee’s approach acceptable as the PAD5 SBLOCA assessment will continue to support a core power level of 3,723 MWt for the SBLOCA analysis of record and will be performed in accordance with the requirements of 10 CFR 50.46.

<u>Full Spectrum Loss of Coolant Accident</u>

The proposed application of the FSLOCA evaluation model at MPS3 was submitted in a separate LAR submittal dated November 5, 2020 (Reference 30). As stated in the MUR LAR, the MPS3 FSLOCA analysis supports the MUR power level. NRC License Amendment No. 279, dated October 5, 2021 (Reference 31) approved the implementation of the Millstone 3 FSLOCA evaluation model for fuel cycle 22 operation, coincident with the MUR power uprate, by adding WCAP-16996-P-A, Revision 1 (Reference 43), to the list of methodologies approved for reference in the core operating limits report for MPS3. In its safety evaluation, the NRC concluded that the licensee appropriately applied the FSLOCA evaluation model and the analysis results satisfy the 10 CFR 50.46(b)(1) through (b)(4) requirements.

3.2.3 Technical Conclusion

The NRC staff reviewed the current accident and transient analyses discussed in the licensee’s LAR and concludes that they are adequately addressed for the impact of the 1.6 percent power uprate at Millstone 3. The NRC staff also concludes that Millstone 3 will continue to meet the requirements of GDC 4, 10, 15, 16, 27, 33, and 34, as well as 10 CFR 50.46, 10 CFR 50.62, and 10 CFR 50.63 following implementation of the proposed 1.6 percent MUR power uprate. The proposed change is consistent with the guidance of RIS 2002-03.

3.3 Mechanical, Structural, and Material Component Integrity and Design

3.3.1 Reactor Vessel, Reactor Vessel Internals, Reactor Coolant System, and Supports

3.3.1.1 Regulatory Evaluation

Millstone 3 is a Westinghouse 4-loop pressurized water reactor with a dry, sub-atmospheric containment. Ultrasonic flow meters (UFMs) were installed, calibrated, and commissioned for Millstone 3 in 2004. The UFMs use acoustic energy pulses to measure main feedwater temperature and flow rate. These parameters are used to determine reactor thermal power. The UFMs consist of a measuring section containing 16 ultrasonic multi-path transit time transducers, one dual resistance temperature detector, one pressure transmitter installed in each of the four feedwater lines, and an electronic signal processing cabinet.

Nuclear power plants are licensed to operate at a specified core thermal power, referred to as the rated thermal power (RTP). 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 emergency core cooling system (ECCS) analyses for postulated loss of coolant accidents (LOCAs). This requirement provides assurance that instrumentation uncertainties are adequately addressed 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 licensee has proposed to use a power measurement uncertainty of 0.4 percent based on the installation of the Cameron LEFM CheckPlus system. This system provides a more accurate measurement of feedwater (FW) flow than current systems, including those available when 10 CFR Part 50, Appendix K, was issued.

The NRC staff's review of the LAR in the area 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 the structures, systems, and components (SSCs) at Millstone 3 will continue to be adequately maintained following the implementation of the proposed MUR power uprate under normal, upset, emergency, and faulted operating conditions, as applicable. Reasonable assurance of the structural and pressure boundary integrity of the SSCs at Millstone 3 will be provided by demonstrating compliance with the NRC regulations in 10 CFR 50.55a, "Codes and standards," and the general design criteria (GDC) for Millstone 3 as further discussed below.

The GDC which constitute the licensing bases for Millstone 3 are those described in the FSAR, Chapter 3.1, "Conformance with NRC General Design Criteria," and in applicable FSAR sections. The NRC staff's assessment of the Millstone 3 LAR in the areas of mechanical engineering and IST considered the following NRC regulations:

- The regulation at 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 during any condition of normal operation, including anticipated operational occurrences.

- GDC 30, "Quality of reactor coolant pressure boundary," as it relates to the reactor coolant pressure boundary being designed, fabricated, erected, and tested to the highest quality standards practical.

The acceptance criteria are based on continued conformance with the requirements of the above criteria listed in FSAR Section 3.1. The design and licensing bases for Millstone 3 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 provides reasonable assurance that compliance with the applicable regulations will be maintained upon implementation of the proposed MUR power uprate.

The NRC's acceptance criteria for reviewing the safety-related valve analysis are based on 10 CFR 50.55a, and the acceptability of Motor-Operated Valves (MOVs) is based on meeting the requirements of Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance" (Reference 44); GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves" (Reference 45); and GL 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power Operated Gate Valves" (Reference 46).

The Code of Record for Millstone 3 is the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code), 2012 Edition for the fourth 10 year IST Program at Millstone 3, in compliance with the requirements of the 10 CFR 50.55a. The IST Program provides reasonable assurance of the operational readiness of pumps, valves, and dynamic restraints to perform their safety functions.

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 in the area of mechanical engineering covers the structural and pressure boundary integrity of the piping, components, and supports, which make up the nuclear steam supply system (NSSS) and the balance-of-plant (BOP) systems. The mechanical engineering review scope also includes an evaluation of other new or existing SSCs that 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 Millstone 3 pressure-retaining components and their supports and the reactor vessel (RV) internals. The NRC staff's review also considered the impact of the proposed MUR power uprate on postulated high energy line break (HELB) locations and corresponding dynamic effects resulting from the postulated HELBs, including pipe whipping and jet impingement. A review of the impact of the MUR power uprate on flow induced vibration (FIV) was also performed. The NRC staff's review focused on verifying that the licensee has provided reasonable assurance of the structural and pressure boundary integrity of the Millstone 3 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 at Millstone 3 will increase the RTP level from 3,650 MWt to 3,709 MWt. In accordance with the 10 CFR 50, Appendix K, requirements, the licensee notes in Section IV, "Mechanical/Structural/Material Component Integrity and Design," of Attachment 4, "RIS 2002-03 Requested Information," of the LAR that the current ECCS analyses of record (AOR) are based on a maximum analytical thermal power level of 102 percent of RTP (3,650 MWt), which bounds the proposed MUR power uprate licensed thermal power level of 3,709 MWt.

3.3.1.2.1 Power Uprate Evaluation Parameters and Design Bases

As stated in its LAR, the NSSS design parameters are the fundamental parameters used as input in the NSSS analyses. The design parameters are established using conservative input assumptions to provide bounding conditions used in NSSS analyses. The design parameters include primary and secondary side system conditions (thermal power, temperatures, pressures, and flows) that are used as inputs to the NSSS analyses and evaluations. These input parameters were revised to account for the increase in analyzed core power from 3,650 MWt to 3,723 MWt. Core power was conservatively increased by 2 percent in the analysis to bound the MUR power uprate value.

In Table 3.0-1, "NSSS Design Parameters for MPS3 MUR Power Uprating-Cases 1 through 4" and Table 3.0-2, "NSSS Design Parameters for MPS3 MUR power uprate – Cases 5 and 6,"in Section 3.3 of Attachment 1 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 (3,723 MWt). This power level corresponds to the power level following the implementation of the MUR power uprate (i.e., 3,709 MWt) plus the revised uncertainty of 0.4 percent.

As shown in the tables, there is no change in the reactor coolant system (RCS) operating pressure (2250 pounds per square inch absolute (psia)) as a result of the Millstone 3 MUR power uprate. Also, there is no change in thermal design flow after implementation of the MUR power uprate. Other parameter changes are small as noted below. The implementation of the MUR power uprate would yield a RCS hot leg temperature (T_{hot}) of 615.7 degree Fahrenheit (°F), a 0.6 °F increase from the current temperature of 615.1 °F, and a cold leg temperature (T_{cold}) of 547.3 °F (a drop of 0.7 °F from the current temperature of 547.9 °F), resulting in no change to the average RCS temperature. The main steam (MS) pressure increases by 12 psi to 954 psia at the MUR power uprate conditions and the MS steam flow increases from 15.1E+06 lb/hr to 15.41E+06 lb/hr at the MUR power uprate conditions. Steam temperature increases by 1.5 °F from 528.7 °F to 530.2 °F and the final FW temperature would increase by 2.6 °F from 445.3 °F to 447.9 °F as a result of MUR power uprate implementation.

The information related to the structural qualification of SSCs at Millstone 3 is contained in Chapter 3 of the FSAR. The FSAR describes the design criteria applicable to the Millstone 3 SSCs, including loads, load combinations, and acceptance criteria stipulated by the applicable codes of record for these SSCs. In Section IV.1.A, among other locations in the LAR, the licensee notes that implementation of the MUR power uprate does not change current operating transients. As such, loads, stresses, and fatigue values resulting from these transients that are used in the structural evaluations of SSCs are not affected. Similarly, the proposed MUR power uprate has no effect on the deadweight and seismic loads of existing SSCs. Therefore, the NRC staff has determined that the existing AORs for these SSCs remain valid.

The functional description of the RCS, including the RV, Reactor Coolant Pumps (RCPs), RCS piping, and Steam Generators (SGs), is discussed in Chapter 5 of the Millstone 3 FSAR. Chapter 10 of the Millstone 3 FSAR provides the design basis information for the secondary side systems, including the main steam, feedwater, and condensate systems.

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 AOR, the discussion should cover the type of confirmatory information identified in Section II (i.e., accidents and transients for which the existing analyses of record bound plant operation at the proposed uprated power level). For components that are not bounded by existing AOR, a detailed discussion should be provided.

The evaluations discussed in Section IV of RIS 2002-03 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 a 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. The codes of record for Millstone 3 are documented in Table IV-4 of Attachment 4 to the LAR.

The pressure-retaining components and component supports, including piping and pipe supports, which must be evaluated in support of a MUR power uprate include the following: the RV, including the RV shell, RV nozzles and supports; 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, and 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. Furthermore, Section IV.1.B of RIS 2002-03 indicates that the SSCs, affected by implementation of a MUR power uprate, need to be evaluated. Any changes related to the power uprate in the following areas should be identified and evaluated for the following:

- stresses
- cumulative usage factors
- flow induced vibration
- changes in temperature (pre- and post-uprate)
- changes in pressure (pre- and post-uprate)
- changes in flow rates (pre- and post-uprate)
- high-energy line break locations
- jet impingement and thrust forces

Section IV.1.B.i and Section IV.1.A of Attachment 4 to the LAR discuss stresses in NSSS components and BOP piping (NSSS interface systems, safety-related cooling water systems, and containment systems). The revised design conditions for the NSSS components and BOP piping were reviewed by the licensee for impact on the existing AOR. No changes in RCS design or operating pressure were made as part of the MUR power uprate. For Class 1 components and piping, the stresses and cumulative usage factors were determined to be acceptable for a 60-year plant life for each of the components. In summary, the licensee's evaluations have concluded that the proposed MUR power uprate has no adverse impact on Class 1 fatigue AOR.

The NRC staff reviewed BOP piping as discussed in Section IV.1.A.v of Attachment 4 to the LAR. BOP piping includes NSSS interface systems, secondary side power systems (e.g., main steam, feedwater), safety-related cooling water systems, and containment systems. The MUR power uprate operating conditions for the BOP piping were reviewed by the licensee for impact on the existing piping and supports design basis analyses. Thermal, pressure, and flow rate change factors were used in determining piping systems acceptability for MUR power uprate conditions. The BOP piping systems remain acceptable for MUR power uprate conditions. Based on these considerations, the NRC staff concludes that all pressure-retaining components including piping and pipe supports remain bounded at MUR power uprate conditions for Millstone 3.

The NRC staff determined the licensee's assessments of the pressure-retaining components and component supports is acceptable based on the following: (1) the licensee's approach to disposition SSCs as unaffected by the proposed power uprate is consistent with RIS 2002-03; (2) the licensee's confirmation that the existing AORs for all of the applicable SSCs remain bounding when considering the plant parameter changes at the MUR power uprate level provides reasonable assurance 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 magnitude of plant parameter changes, as documented in Table 3.0-1 and Table 3.0-2 of Attachment 1 to the LAR, is generally minor and supports the licensee's assessment, which concludes that all pressure-retaining components remain bounded.

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 implementation of the proposed MUR power uprate. Affected components and supports refer to those for which their AOR is not bounded at MUR power uprate conditions. Pressure-retaining components and their supports generally remain unaffected by the implementation of a MUR power uprate based on the fact that they have been analyzed at conditions that are more limiting than those which 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 implementation of the MUR power uprate, identified above, affect the loads included in the AOR for the component and its supports.

Based on its evaluations of the impact of MUR power uprate implementation on the components identified above, the licensee stated that the existing AORs related to the structural and mechanical qualifications of the following SSCs are unaffected by the proposed MUR power uprate at Millstone 3 for the RV, RV nozzles and RV supports; the pressure-retaining portions of the CRDMs; NSSS piping, pipe supports and branch nozzles associated with the RCS; BOP piping and supports; SGs, including their supports, the SG shells, and 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. Based on these considerations, the NRC staff concludes that all pressure-retaining components and supports, including BOP piping and pipe supports, remain bounded at MUR conditions.

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 Millstone 3 RV internals. Section IV.1.A.ii of Attachment 4 to the LAR addresses reactor core support structures and vessel internals. The revised design conditions for the MUR power uprate were evaluated for impact on the existing RV internals analysis, including a 2 percent increase from 3,650 MWt to 3,723 MWt, which bounds the MUR power increase.

The MUR power uprate design conditions do not affect the current design bases for seismic and LOCA loads. The flow induced vibration (FIV) stress levels on the core barrel assembly and upper internals are low and remain well below the material high-cycle fatigue endurance limit. Therefore, the MUR power uprate design conditions do not affect the structural margin for FIV.

Evaluations were performed by the licensee to demonstrate that the structural integrity of reactor internal components are not adversely affected by the MUR power uprate design conditions. For reactor internal components including the lower core plate and the upper core plate, the stresses and cumulative fatigue usage factor of the current analyses remain bounding at MUR power uprate design conditions. The NRC staff's evaluation of the various RV internal components are provided in section 3.3.2.2 of this SE.

Based on a review of the evaluations performed for Millstone 3 of the RV internals, upper and lower core plate, baffle barrel region, and core support structures, the NRC concludes that these reactor internal components continue to meet their design criteria at the MUR power uprate design conditions.

The structural evaluation of Control Rod Drive Mechanisms (CRDMs) is addressed in Section IV.1.A.ii of Attachment 4 to the LAR. The updated design conditions were reviewed for impact on the existing CRDM design basis structural analyses. CRDMs are subjected to T_{cold} temperatures and RCS pressures. The maximum T_{cold} from the MUR power uprate design parameters is 555.8 °F. The maximum operating temperature used in the AOR is 622.6 °F. As a result, the AOR remains bounding and applicable. No changes in RCS design or operating pressure were made as part of the MUR power uprate. The temperature and pressure transients are not affected by the MUR power uprate. Therefore, the original transient analysis remains bounding and applicable to the MUR power uprate conditions. The stresses and cumulative fatigue usage factor of the existing analyses remain bounding at MUR power uprate design conditions for CRDMs. The NRC staff's evaluation of the structural integrity of the CRDM under MUR power uprate conditions is provided in Section 3.3.2.2 of this SE.

As described in Section IV.1.A.ii of Attachment 4 to the LAR, the existing bottom mounted instrumentation (BMI) guide tubing/flux thimble design bounds the MUR power uprate because there is no change to thermal stresses or pressure stresses. The NRC staff's evaluation of the structural integrity of the BMI guide tubing/flux thimble under MUR power uprate conditions is provided in Section 3.3.2.3 of this SE.

Based on a review of the licensee's assessments, the NRC staff concludes that the design basis analyses of the RV internals and core support structures remain unaffected and bounding following implementation of the MUR power uprate at Millstone 3.

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

Section IV.1.A.iv of Attachment 4 to the LAR discusses the acceptability of NSSS piping, pipe supports and branch nozzles. The MUR power uprate was evaluated by the licensee for impact on the AOR for the reactor coolant loop piping, including the loop bypass line and the pressurizer surge line, primary equipment nozzles (reactor pressure vessel inlet and outlet, SG inlet and outlet, and RCP suction and discharge), primary equipment supports (reactor pressure vessel nozzle supports and neutron shield tank, SG columns and lateral snubbers, RCP columns and lateral snubbers, and pressurizer supports, reactor coolant loop branch nozzles (accumulator and charging line), and Class 1 auxiliary piping systems attached to the reactor coolant loop. There are no significant changes to the reactor coolant loop thermal analysis, LOCA analysis, main steam line break (MSLB) analysis, and reactor coolant loop piping system fatigue evaluations. The existing design transients remain valid for the uprated conditions. The T_{hot} and T_{cold} variations are conservative and bounding.

There were no changes to any existing NSSS design transient parameters. There are no significant changes to the pressurizer surge line operating conditions. Therefore, the AOR for reactor coolant loop piping system remains applicable for the MUR power uprate conditions. The reactor coolant loop displacements at the Class 1 auxiliary line connections to the reactor coolant loop, Class 1 auxiliary lines, primary equipment nozzle qualification, branch nozzle qualification, and primary equipment supports loads are unchanged. The maximum primary and secondary stresses remain valid and maximum usage factors are evaluated by the licensee where appropriate based on the code of record listed in Section IV.1.D of Attachment 4 to the LAR. Based on a review of the information noted above, the NRC finds NSSS piping, pipe supports and branch nozzles are acceptable for MUR power uprate.

<u>Design Transient Applicability</u>

In Section II.2.1 of Attachment 4 to the LAR, the licensee provided its evaluation for the applicability of NSSS design transients for the Millstone 3 MUR power uprate conditions. NSSS design transients were specified in the original design analysis of NSSS components cyclic behavior. The selected transients are conservative thermal/hydraulic representations of plant operations and postulated operating events and are included in the various component design specifications. The NSSS design transients depict variations in fluid temperature, pressure, and flows that are seen by major RCS components. NSSS design transients are used to perform fatigue analyses of the affected components to provide confidence that the component is appropriate for its application over the 60-year plant license period. The existing design transients were evaluated for their continued applicability at MUR power uprate conditions.

The full power design parameters upon which the existing NSSS design transients are based were compared by the licensee to the design parameters for the MUR power uprate. This comparison showed that the maximum hot leg and minimum cold leg temperatures are within 0.6 °F/-0.7 °F of the corresponding values used in the existing design transients. The minimum secondary side steam pressure at the MUR power uprate is about 8 psi lower than in the current licensing basis (CLB) transients. These differences are considered within the analytical tolerances. The maximum feedwater temperature is slightly higher (2.6 °F) at MUR conditions as compared to the CLB conditions; however, this small difference is not significant to the design transients. It is also noted that the design transient parameter responses are provided as changes from the initial value; hence, applying MUR initial conditions to the existing responses further reduces the parameter differences identified above.

The NRC reviewed the information regarding design transients in Section II.2.1 of Attachment 4 to the LAR. As discussed above, since the evaluated full power plant design parameters are within the allowable ranges or tolerances as compared to the design parameters for the MUR power uprate, the NRC staff finds that the existing design transients used in the component fatigue evaluation remain applicable at MUR conditions for use in the various NSSS component evaluations, and is therefore, acceptable.

Auxiliary Equipment Design Transients

In Section II.2.2 of Attachment 4 to the LAR, the licensee provided evaluation of the auxiliary equipment design transients to verify their continued applicability to the NSSS design parameters approved for the Millstone 3 MUR power uprate.

The only transients that could potentially be affected are the temperature transients that are impacted by full load T_{cold}. All of these transients are based on an assumed full load T_{cold} of 560 °F. This NSSS design temperature was selected to ensure that the resulting design transients would be conservative for a wide range of NSSS design temperatures. A comparison of the MUR NSSS design temperature range for T_{cold} (536.7 °F to 555.8 °F) with the T_{cold} reference value used to develop the design transients indicates the range of NSSS design temperatures approved for the MUR power uprate is less than the reference design value.

Since the actual temperature transients (i.e., the change in temperatures from the MUR T_{cold} value to the lower auxiliary system related temperatures or vice versa) are less severe than the design temperature transients, the NRC staff finds that existing auxiliary equipment design transients are conservative and remain bounding for the MUR power uprate, and is therefore, acceptable.

3.3.1.2.5 Balance-of-Plant Piping (NSSS including Interface Systems, Secondary Side Power Systems, Safety-Related Cooling Water Systems, and Containment Systems

The licensee analyzed conditions under the proposed MUR uprate against the existing design basis analyses for BOP systems which include NSSS interface systems, secondary side power systems, safety-related cooling water systems and containment systems. The MUR operating conditions were reviewed for impact on the existing piping and supports design analysis. The licensee used a thermal change factor, pressure change factor and flow rate change factor in determining the acceptability for piping systems under the MUR power uprate conditions. The change factor was based on the ratio of the MUR Value/Analyzed Value for each parameter. Where the ratio was less than or equal to 1.0, the pre-MUR power uprate condition was considered acceptable because it is either the same or envelops the post MUR condition. Where the ratio was greater than 1.0, the licensee stated an evaluation was done to document specific system acceptability. The licensee stated that all BOP systems evaluated remain acceptable and will continue to satisfy the design basis requirements.

The licensee also analyzed the operating conditions under the proposed MUR uprate against the existing design basis analyses for the reactor coolant loop piping including the loop bypass line, pressurizer surge line, primary component nozzles, primary component supports, reactor coolant loop branch nozzles, and Class 1 auxiliary piping systems attached to the reactor coolant loop. This analysis determined the existing design transients remain valid for the uprated conditions. The T_{hot} and T_{cold} variations in the design analysis are conservative and remain bounding. Therefore, the analysis of record remains applicable to the power uprate conditions. The reactor coolant loop displacements and support loads are unchanged by the power uprate operating conditions.

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 NSSS piping, pipe supports and branch nozzles will not result in an increased risk of damage or failure as a result of the uprate, and is therefore, acceptable.

3.3.1.2.6 Steam Generator

The structural integrity of the steam generator (SG) is addressed by the licensee in Section IV.1.A.vi.2 of Attachment 4 to the LAR. The structural evaluation focused on the potential changes to the structural analysis inputs for the SG components, both primary and secondary side, resulting from an increase in the analyzed power from 3,666 MWt to 3,739 MWt for the MUR power uprate. The analyzed power at MUR power uprate represents approximately a 2 percent increase in current NSSS power of 3,666 MWt. Millstone 3 has defined in-process parameter limitations for the MUR power uprate that will reduce impact on SG component structural integrity.

Seismic input, LOCA loads, faulted and test transients, and deadweight loads used in AOR shows that the MUR analyzed power increase of 3,739 MWt from the current analyzed power of 3,666 MWt results in either insignificant changes or no change from the current AOR. The AOR's thermal transients remain bounding for MUR power uprate conditions with respect to the current SG component stress and fatigue levels. There is no change to the AOR stress and fatigue results for the SG primary-side components.

Regarding SG secondary side components, a comparison of the MUR power uprate parameters to those in the AOR show that the steam pressure/temperature and minimum feedwater temperature for MUR power uprate conditions are unchanged, or are enveloped by the parameters used in the AOR. The slight increase feedwater temperature is bounded by the design temperature of the piping. The ASME B&PV code design pressure requirements are satisfied.

The NRC staff's evaluation of the SG tubes, secondary side internal support structures, shell, nozzles, and blowdown system is provided in Section 3.3.3 of this SE.

The cumulative usage factors were determined to be acceptable for a 60-year plant life for each of the components, except the SG secondary side manway bolting. SG secondary side manway bolting is treated as a Class 1 component and a replacement frequency is used to ensure the cumulative usage factor remains less than 1.0. The MUR proposed power uprate has no adverse impact upon the SG secondary side manway bolting fatigue AOR because SG operating pressure/temperature decreases.

In summary, evaluations have concluded that MUR power uprate causes no adverse impact on Class I fatigue AOR. Based on MUR power uprate SG pressure and temperature parameters, the existing Millstone 3 AOR remains applicable and bounding for MUR power uprate conditions and the SG components will continue to meet the applicable ASME B&PV code fatigue and stress limits.

3.3.1.2.7 Reactor Coolant Pumps

Section IV.1.A.vii of Attachment 4 to the LAR addresses Reactor Coolant Pumps (RCPs). The MUR power uprate conditions were reviewed for impact on the RCP design basis analysis. There are no changes in RCS design or operating pressure as a result of the MUR power uprate. The maximum vessel inlet temperature considered in the existing structural design analysis of the RCPs is more limiting or conservative for the MUR power uprate. The MUR power uprate conditions remain bounded by the original design conditions or previously evaluated conditions. The existing NSSS design transients that have been evaluated for Millstone 3 remain applicable for the MUR power uprate conditions.

The RCP motors were also evaluated for horsepower loading at continuous hot operation, starting ability of the motor, and loads on the thrust bearings. The RCP motors are acceptable for the loads calculated at MUR power uprate RCS conditions. The stresses and cumulative fatigue usage factor of the existing analyses remain bounding at MUR power uprate design conditions.

The revised RCS conditions are acceptable for the RCP with respect to ASME B&PV code structural integrity. The original code of record listed in Section IV.1.D remains unchanged. Therefore, the revised MUR power uprate conditions remain bounded by the original or previously evaluated design conditions, and RCPs are acceptable for MUR power uprate.

3.3.1.2.8 Pressurizer Shell, Nozzles, and Surge Line

Section IV.1.A.viii of Attachment 4 to the LAR addresses the impact of MUR power uprate on the pressurizer shell, nozzles, and surge line. The MUR power uprate operating conditions were reviewed by the licensee for impact against the existing pressurizer design basis analysis. The limiting pressurizer conditions occur when the RCS pressure is high and the RCS T_{hot} and

T_{cold} are low. No changes were made in the RCS design or operating pressure as part of the MUR power uprate. The minimum T_{hot} and T_{cold} values from the design parameter cases were used in the pressurizer evaluation. At the normal operating pressure of 2250 psia, the revised T_{hot} and T_{cold} temperature differences for normal operation are bounded by the existing analysis.

The NSSS design transients did not change and were enveloped by the existing design transients. Pressure fluctuations during the MUR power uprate transients are the same as in the existing evaluations. The maximum pressure within each load category (normal, upset, emergency, faulted, and test) has not changed from the value used in the original evaluations. Therefore, the pressurizer shell, nozzles and surge line meet the stress/fatigue analysis requirements for plant operations at MUR power uprate conditions. The ASME B&PV code of Record is listed in Section IV.1.D and remains unchanged.

<u>Pressurizer Surge Line Thermal Stratification</u>

Section IV.1.B.iv.2 of Attachment 4 to the LAR addresses the potential for thermal stratification in the pressurizer surge line. The pressurizer surge line, which is part of the NSSS, connects the reactor coolant loop to the pressurizer. The thermal stratification analysis of the pressurizer surge line accounts for the thermal expansion of the piping resulting from the top-to-bottom temperature difference imposed on the piping by the stratified fluid. Stratification of the fluid occurs because of the higher temperature in the pressurizer with respect to the lower temperature in the reactor coolant loop. For the MUR power uprate, the operating parameters and design transients for the NSSS were compared by the licensee to those considered in the previous thermal stratification analysis in AOR in response to NRC Bulletin 88-11, "Pressurizer Surge Line Stratification," (Reference 47). Because there is no appreciable change to the parameters or transients, the effect of the MUR power uprate on the thermal stratification analysis of the pressurizer surge line is not significant. Therefore, the results from the previous thermal stratification analysis in AOR remain valid for the MUR power uprate.

3.3.1.2.9 Reactor Coolant System Analysis for Alloy 600

The licensee analyzed three areas of the primary coolant pressure boundary containing Alloy 600 material, RV head penetrations, RV hot leg nozzles and bottom mounted instrumentation (BMI) nozzles for impact from the MUR uprate. These three areas are subjected to the highest temperature and would therefore bound all remaining Alloy 600 material in the primary system with respect to susceptibility to primary water stress corrosion cracking. The analysis showed that the AOR and the pre-MUR uprate conditions remain bounding for these areas.

Since the changes due to the MUR power uprate will continue to meet the existing design basis analyses, the NRC staff finds that the effects on Alloy 600 material in the reactor coolant pressure boundary will not result in an increased risk of damage or failure, and is therefore, acceptable.

3.3.1.2.10 Stresses

The breadth of the NRC staff's review of the stresses were whether the current analyses of stresses on the different RCS systems and the BOP systems discussed in IV.1.A.v.remain bounding for the plant as a result of the MUR. The licensee confirmed that there were no changes in the RCS design or operating transient conditions that were made as a result of the MUR 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's evaluation of stress and fatigue factors of the various RV internal components are provided in Section 3.3.2 of this SE.

The NRC staff finds that the licensee evaluated stresses for the RCS piping and BOP piping will not exceed previously evaluated conditions due to the MUR and, therefore, are acceptable.

3.3.1.2.11 Cumulative Usage Factors

The NRC staff reviewed the NSSS components, piping and interface systems to determine if the revised design conditions due to the MUR power uprate would impact the existing design basis analyses. As indicated in Section 3.3.1.2.6 above, the cumulative usage factors were determined to be acceptable for a 60-year plant life for each of the components with exception of the SG secondary side manway bolting. SG secondary side manway bolting is treated as a Class 1 component and a replacement frequency is used to ensure the cumulative usage factor remains less than 1.0. The SG secondary side manway bolting cumulative usage factor is managed currently by replacement on a required frequency and is not impacted by the MUR uprate conditions. The NRC staff determined that the revised conditions as a result of MUR remain bounded by the current analyses, and therefore, the postulated transients previously evaluated for Millstone 3 do not need to be revised. Additionally, this means that the current stress and fatigue values (including cumulative usage factors) remain valid with the implementation of the MUR power uprate, and therefore, are acceptable.

3.3.1.2.12 Postulated Pipe Ruptures and Associated Dynamic Effects

The licensee provided a discussion of high energy line break (HELB) locations in Section IV.1.B.vii of Attachment 4 to the LAR. The Millstone 3 FSAR Section 3.6 describes the high energy line break analysis. MUR power uprate operating temperatures, pressures, and mass flow rates were compared to the analyzed conditions. The review showed that the total pipe stresses are not significantly impacted by the MUR power uprate; therefore, there are no new or revised pipe break locations as a result of MUR power uprate. Based on a review of the information provided in the LAR regarding HELB, the NRC finds that the existing design basis for pipe break, jet impingement, and pipe whip remains valid for the MUR power uprate. The NRC staff concludes that the changes due to the MUR power uprate are not significant enough to necessitate adjustments to the HELB program, and therefore, the previously conducted analyses remain bounding and acceptable.

The high and moderate energy discussion is provided by the licensee in Section VII.6.B of Attachment 4 to the LAR. The high and moderate energy break program ensures that systems or components required for safe shutdown or important to safety are not susceptible to the consequences of high and/or moderate energy pipe breaks. FSAR Section 3.6, "Protection Against Dynamic Effects Associated with the Postulated Ruptures of Piping," describes the high and moderate energy line break analysis. High-energy pipe breaks are analyzed for piping for which the maximum operating pressure exceeds 275 pounds per-square-inch gauge (psig) and the maximum operating temperature exceeds 200 °F. Through-wall leakage cracks are postulated in moderate energy piping (located outside containment only) for which both the maximum operating pressure is less than or equal to 275 psig and the maximum operating temperature is less than or equal to 200 °F.

The proposed MUR power uprate does not result in any new or revised high or moderate energy line break locations. The high and moderate energy line break analysis is not affected. Area temperatures and pressures resulting from high and moderate energy line breaks and internal flooding conditions resulting from high or moderate energy line breaks remain valid at MUR power uprate conditions.

Leak Before Break (LBB) Evaluation

The existing Millstone 3 LBB analyses justified eliminating large primary loop pipe rupture from the Millstone 3 structural design basis. The applicable pipe loadings, normal operating pressure, and temperature parameters at MUR power uprate conditions were used to evaluate the impact of the proposed MUR power uprate on the LBB analysis. The licensee stated LBB acceptance criteria are satisfied for primary loop piping at MUR power uprate conditions and the existing analysis of record's conclusions remain valid. The NRC staff has reviewed the licensee's evaluations and determined that the licensee's conclusion is reasonable, given the small magnitude in temperature and pressure increases, which accompany MUR power uprate implementation. Correspondingly, as previously discussed, these small changes generally have no impact on pressure-retaining components such as piping. Therefore, the dynamic effects of RCS primary loop piping breaks are not considered in the Millstone 3 structural design basis at MUR power uprate conditions.

3.3.1.2.13 Safety-Related Valves, Pumps, and Dynamic Restraints

Safety-Related Valves

The NRC staff reviewed the licensee's analyses of the impact of the MUR power uprate on safety-related valves at Millstone 3. The staff also evaluated whether any design changes and any plant-specific evaluations are required as a result of MUR power uprate. As described in Section 3.3.1.1 of this SE, the NRC's acceptance criteria for reviewing the safety-related valve analysis are based on 10 CFR 50.55a, and the acceptability of MOVs is based on meeting the requirements of GL 89-10 (Reference 44), GL-96-05 (Reference 45), and GL 95-07 (Reference 46).

In the LAR, the licensee reviewed the impact of the proposed MUR power uprate on the existing design basis analyses for safety-related valves. No changes in RCS flow, design, or operating pressure were made 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 no safety-related valves required a change to their design or operation as a result of the MUR power uprate.

The pressurizer code safety valves, power-operated relief valves, and block valves located on top of the pressurizer provide overpressure protection for the RCS. The changes due to the MUR power increase that could potentially impact the pressurizer valves are RCS mass and reactor power (including RCP heat). The RCS mass does not significantly change due to the MUR power increase, based on the small changes in T_{hot} and T_{cold} of the RCS. The MUR power uprate is bounded by the current design basis event transient analyses in Section II of Attachment 4 to the LAR, and thus, there is no adverse impact on the pressurizer overpressure protection valves from the MUR power uprate. Based on this review, it was determined that the current AOR for the pressurizer overpressure protection valves remains bounding for the MUR power uprate conditions. Other safety-related valves were reviewed as part of the system that contains those valves. As discussed in Sections IV.1.A.ix and VI.1, operating conditions for interfacing systems will reflect only a small to no change under MUR power uprate conditions. Based on these reviews, it was determined that the AOR for interfacing system valves remain bounded at MUR conditions.

The MOV Program for Millstone 3 was updated to reflect the recommendations of GL 89-10 (Reference 44) and GL 96-05 (Reference 45), and is not impacted by the MUR power uprate. The implementation of the MUR power update does not change the regulatory requirements applicable to the MOV program or change the scope of the program. The systems that contain MOVs 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 MOV Program. No MOVs will be added or deleted from the 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 programs.

The air-operated valve (AOV) program for Millstone 3 is not impacted by the MUR power uprate. The systems that contain 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 or deleted from the 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 its review of the licensee's evaluations, the NRC staff concludes that the performance of existing safety-related valves is acceptable with respect to the MUR power uprate at Millstone 3.

<u>Safety-Related Pumps</u>

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.

Based on review of the licensee's evaluations, the NRC staff concludes that the performance of existing safety-related pumps is acceptable with respect to the MUR power uprate at Millstone 3.

<u>Inservice Testing Program</u>

In the LAR, the licensee described the review of the IST program for safety-related pumps and valves, dynamic restraints (snubbers) at Millstone 3 during MUR power uprate conditions. The staff review of dynamic restraints (snubbers) based on the MUR power uprate is described below. The code of record for Millstone 3 is the ASME code for Operation and Maintenance of Nuclear Power Plants (OM code), 2012 Edition for the fourth 10 year IST Program at Millstone 3, in compliance with the requirements of the 10 CFR 50.55a (Reference 48). The IST Program provides reasonable assurance of the operational readiness of pumps, valves, and dynamic restraints to perform their safety functions.

Inservice Examination and Testing Program for Dynamic Restraints (Snubbers)

The inservice examination and testing program for snubbers assesses their operational readiness within the scope of the ASME OM code. For the MUR power uprate, there will be no significant changes to operating conditions, or the design basis requirements that would affect component performance, test acceptance criteria, or reference values. Therefore, the existing snubber program will not be impacted by the MUR power uprate. Based on review of the licensee's evaluation, the NRC staff concludes that the snubber program will be acceptable for the MUR power uprate conditions.

Safety-Related Snubbers (Dynamic Restraints)

The NRC staff reviewed the licensee's safety-related snubbers impacted by load increases due to the MUR power uprate. The licensee stated in an e-mail dated May 18, 2021 (Reference 49), that the term "pipe support" applies generically for all types of pipe supporting elements at Millstone 3. The term "pipe supports" includes, snubbers, spring hangers, and rigid supports (e.g., struts, fabricated anchors, box frames, racks, etc.). Thus, the NRC staff considers that snubbers were considered in the MUR assessment for the adequacy of pipe supports for potential load increases and for the impact of larger travel due to higher operating temperatures, and the conclusions in Section IV.1.A.iv and Section IV.1.A.v of the Millstone 3 LAR are also applicable to snubbers. Therefore, the existing snubbers will not be impacted by the MUR power uprate. Based on review of the licensee's evaluation, the NRC staff concludes that the 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, to maintain the safe shutdown condition, or to mitigate the consequences of an accident are acceptable for the MUR power uprate conditions.

In summary, the NRC staff determined that there are no significant changes to operating conditions or the design basis requirements that would affect component performance, test acceptance criteria, or reference values. The Millstone 3 IST Program, in accordance with the ASME OM code, includes applicable pumps, valves, and snubbers. The evaluation of Millstone 3 IST Program for a MUR power uprate concluded that the MUR power update will not affect the current IST Program. Based on review of the licensee's evaluation, the NRC staff concludes that the IST Program at Millstone 3 is acceptable for the MUR power uprate conditions.

Inservice Inspection Program

Pursuant to 10 CFR 50.55a(g)(4), the inservice inspection (ISI) program contains the Millstone 3 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 Millstone 3 FSAR Section 5.2.4 while Class 2 and 3 components are found in FSAR Section 6.6. The NRC staff reviewed the Millstone 3 ISI program to determine if any of the changes to the plant due to the implementation of the MUR 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 increase in pressure and temperature incur any additional stresses that would require additional or more frequent examinations of the same or additional components. The NRC finds that the changes due to the implementation of the MUR are not significant enough to require any changes to the ISI program. Based on its review of the licensee's evaluations, the NRC staff has determined that the ISI program will continue to meet the regulatory requirements of 10 CFR 50.55a upon implementation of the MUR power uprate. Therefore, the NRC staff concludes that the LAR is acceptable with respect 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. Additionally, the NRC staff reviewed the licensee's assessment of the effects on the Millstone 3 HELB AORs, including associated dynamic effects. Based on the review above, the NRC staff concludes that the LAR is acceptable with respect to the structural integrity of the applicable SSCs affected by the MUR power uprate. This acceptance is based on the licensee's demonstration that specific regulatory requirements, related to the review of mechanical/structural/material component integrity and design, will continue to be satisfied following implementation of the MUR power uprate. Specifically, the licensee demonstrated that: (1) the structural and pressure boundary integrity pressure retaining components and supports, including piping and pipe supports, at Millstone 3 are not affected by the proposed MUR power uprate, as evidenced by the fact that their AORs are unaffected; (2) the RV internals at Millstone 3 also remain unaffected, when considering the impact of MUR power uprate implementation on the FIV effects and structural integrity of the RV internals; (3) the Millstone 3 AORs related to the postulation of HELB locations, including dynamic effects associated with these postulated pipe ruptures, remain unaffected by the proposed MUR power uprate; (4) the performance of existing safety-related pumps and valves are acceptable with respect to the MUR power uprate at Millstone 3; and (5) the existing IST Program will not be impacted by the MUR power uprate at Millstone 3 and remains acceptable.

The analyses and evaluations performed by the licensee demonstrate that all acceptance criteria continue to be met. Millstone 3 requires minimal plant modifications to safely operate at the uprated conditions. Based on these considerations, the NRC staff concludes that there is reasonable assurance that the structural integrity of SSCs at Millstone 3 will be adequately maintained following implementation of the MUR power uprate. The staff also concludes that the MUR power uprate will not preclude the capability of these SSCs to perform their intended functions.

3.3.2 Reactor Vessel Integrity

In Section IV.1.A of Attachment 4 to the LAR, the licensee addressed the impacts of the MUR power uprate on the integrity of the reactor vessel (RV), reactor vessel internal, and core support structures. The licensee’s discussion and the NRC staff’s evaluation of these impacts are addressed below.

The NRC staff's review of RV integrity focuses on the potential effects of the MUR power uprate on the service conditions for RV internal and core support structure components to verify that the MUR power uprate will not have an adverse impact on the structural integrity of the RV components. The primary guidance for the MUR power uprate LAR is outlined in RIS 2002-03, “Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications.” Specifically, Section IV, “Mechanical/Structural/Material Component Integrity and Design,” of RIS 2002-03 provides 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 systems, structures, and components.

The NRC staff also reviewed the potential impact of the MUR power uprate on pressurized thermal shock (PTS) calculations, RV pressure-temperature (P-T) limits, Charpy upper-shelf energy (USE) evaluations, and the RV surveillance capsule withdrawal schedules. The NRC staff's review was conducted in accordance with the guidance in RIS 2002-03 to verify that, following implementation of the MUR power uprate, licensee's RV integrity analyses will continue to meet the requirements of 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events"; 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements"; and 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

3.3.2.1 Reactor Vessel, Nozzles and Supports

3.3.2.1.1 Regulatory Evaluation

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

- 10 CFR 50, Appendix A, 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.

- 10 CFR 50, Appendix A, 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 during any condition of normal operation, including anticipated operational occurrences.

- Regulations in 10 CFR 50.55a require the RV be constructed, designed, and analyzed in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel code (ASME code), Section III.

3.3.2.1.2 Licensee's Evaluation

Section IV.1.A.i of Attachment 4 to the LAR discusses the licensee's evaluation of the stress and fatigue usage factors of the RV at MUR power uprate conditions. The licensee stated that the T_{cold} range used in the current analysis of record was 537.4 °F to 556.4 °F while the T_{hot} temperature range evaluated was 605.6 °F to 622.6 °F. The licensee compared the MUR power uprate and current analysis of record operating conditions and determined that the lower range of T_{cold} decreased by 0.7 °F from 537.4 °F to 536.7 °F, while the upper range of T_{hot} increased by 0.6 °F from 622.6 °F to 623.2 °F. The licensee stated that the small changes in T_{cold} and T_{hot} from the values in the current analysis of record due to the MUR power uprate only affects the unit loading and unit unloading (15 percent to 100 percent power) transients.

According to the licensee, during the plant heatup and cooldown transients, the fluid temperatures in the RV vary between the ambient and zero load temperature independent of the normal operating temperatures. The licensee indicated that temperatures during the hydrostatic tests are also independent of the normal operating temperatures. For the remainder of the transients occurring during reactor operation, fluid temperature in each region of the RV varies about the normal operating temperature such that there is a negligible effect due to the small increase in the 100 percent steady-state operating temperature.

The licensee compared the transient thermal stresses created by the unit loading and unit unloading transients for two representative RV wall thicknesses under the current analysis of record and the MUR power uprate conditions. The chosen wall thicknesses pertained to the nozzle beltline region and the RV outlet nozzle reinforced region to address areas exposed to T_{cold} and T_{hot}, respectively. The licensee indicated that there was an insignificant change in the transient thermal stresses for both wall thicknesses between the analysis of record and MUR power uprate conditions. The licensee further indicated that given that the transient pressure stresses and external mechanical load stresses did not change because of the MUR power uprate, the stress and fatigue results for the current analysis of record evaluation remain bounding for the MUR power uprate.

The licensee stated that the faulted condition loads will not change due to the MUR power uprate. Therefore, the faulted condition stress evaluation performed for the current analysis of record remains bounding for MUR power uprate conditions.

According to the licensee, the RV continues to meet the stress and fatigue limits specified in Section III of the ASME code at MUR power uprate conditions. There are no significant changes to any weights or support locations. The licensee stated that there are insignificant changes to the NSSS design and operating parameters, and no changes to the NSSS design transients. The licensee reported that deadweight, thermal, seismic or loss-of-coolant accident LOCA loads in the reactor coolant loop piping did not significantly change the loads on the RV inlet and outlet nozzles under the MUR power uprate.

3.3.2.1.3 Staff Evaluation and Conclusion

The NRC staff determined that MUR power uprate conditions result in insignificant changes in T_{cold} and T_{hot}. As such, MUR power uprate conditions do not significantly affect the thermal stresses in the RV wall, including nozzles attached to the RV. The NRC staff further determined that the MUR power uprate causes insignificant change to the nuclear steam supply system design and operating parameters, pressure stresses, and no changes to the design transients. The NRC staff finds that there are no significant changes to the structural integrity of the RV under MUR power uprate conditions. Because of the above, the NRC staff finds that the licensee has demonstrated reasonable assurance that the structural integrity of RV, nozzles and supports will be maintained for MUR power uprate conditions.

3.3.2.2 Reactor Core Support Structures and Vessel Internals

3.3.2.2.1 Regulatory Evaluation

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

- 10 CFR 50, Appendix A, 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.

- 10 CFR 50, Appendix A, 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 during any condition of normal operation, including anticipated operational occurrences.

- Regulations in 10 CFR 50.55a require the RV be constructed, designed and analyzed in accordance with the ASME code, Section III.

3.3.2.2.2 Licensee’s Evaluation

Section IV.1.A.ii of Attachment 4 to the LAR discusses the impact of MUR power uprate conditions on various RV internals, including a 2 percent increase from 3,650 MWt to 3,723 MWt, which bounds the MUR power uprate. The licensee’s evaluation results are summarized as follows:

<u>Hydraulic Lift Forces and Pressure Losses</u>

Section IV.1.A.ii.c of Attachment 4 to the LAR discusses the licensee’s evaluation of the hydraulic lift forces on the various RV internal components to demonstrate that the reactor internal assembly remains seated and stable for the applicable MUR power uprate conditions. The licensee’s evaluation indicates that the downward force on various RV internal components remains essentially unchanged, and that RV internals would remain seated and stable at MUR power uprate conditions.

<u>Baffle Joint Momentum Flux and Fuel Rod Stability</u>

Section IV.1.A.ii.d of Attachment 4 to the LAR explains that baffle jetting is a hydraulically induced instability or fuel rod vibration caused by a high-velocity water jet. This jet is created by high-pressure water being forced through gaps between the baffle plates that surround the reactor core. The licensee stated that there is no significant change to the pressure differential across the baffle plate, baffle gap width, and fuel assembly model response due to MUR power uprate conditions. The licensee concluded that the baffle joint momentum flux does not change because of the MUR power uprate.

<u>Mechanical Evaluation</u>

Section IV.1.A.ii.e of Attachment 4 to the LAR states that the MUR power uprate design conditions do not affect the current design bases for seismic and LOCA loads. The licensee further stated that the stress levels caused by the flow induced vibration (FIV) on the core barrel assembly and RV upper internals are low and remain well below the material high-cycle fatigue endurance limit. The licensee indicated that MUR power uprate conditions do not affect the structural margin for FIV.

In the June 2, 2021 submittal (Reference 2), the licensee explained that it used the FIV evaluation in the SPU submittal to justify the MUR power uprate increase. The FIV evaluation in the SPU submittal involves scaling the structural response of the FIV according to analytical and experimental formulations based on parameters such as flow rate (mechanical design flow or "MDF"); and vessel inlet and outlet temperatures. The licensee explained that the change in input parameters is shown to be negligible (< 1 percent) so the FIV evaluation for the SPU program was considered to be applicable to the MUR power uprate program, as these parameters are the only change noted in this evaluation between the two programs.

The licensee stated that the SPU evaluation included the RV internal components that are limiting with regards to FIV. These components consist of the core barrel in the lower internals assembly and the guide tubes in the upper internals assembly. The licensee stated that for other RV internal components such as the lower radial restraints, upper core plate alignment pins, lower support plate, and the lower support columns, the vibratory response is extremely small. The licensee calculated stresses by scaling previously generated FIV stresses to what would be expected after SPU implementation. For conservatism, the licensee scaled the values based on the hot functional flow rates, which are typically about 4 percent higher than the MDF rate.

The licensee considered the seismic and LOCA analyses for the SPU program to be applicable to the MUR power uprate program. The licensee stated that the seismic inputs and plant model parameters from the SPU program are also applicable to MUR power uprate conditions. The licensee noted that the assumptions in the LOCA force calculations bound the changes found in the MUR program. The licensee stated that the requirements in the ASME code (1998 Edition with 2000 Addenda, Section III, and Section II, Part D) combined with measured data, form the basis for the acceptance criteria for mechanically induced stresses/strains produced by FIV. The licensee further stated that although the components analyzed do not constitute a pressure boundary, it assessed the acceptability of the components impacted by the MUR power uprate relating to alternating stresses for high-cycle fatigue. The licensee calculated and compared alternating stresses according to the ASME code, Section III rules on high-cycle fatigue or measured experimental data on strains limits in the SPU submittal.

Structural Evaluation

Section IV.1.A.ii.f of Attachment 4 to the LAR states that for RV internal components, the stresses and cumulative fatigue usage factor of the current analyses remain bounding at MUR power uprate conditions. The licensee's evaluations demonstrate that the structural integrity of RV internal components is not adversely affected by MUR power uprate conditions. The licensee evaluated specific RV internals as discussed below.

In the June 2, 2021 submittal (Reference 2), the licensee explained that it evaluated the structural integrity of the reactor internal components, which was previously used in the SPU program, as discussed in its response to NRC's request for additional information (NVIB-RAI-2).

The licensee stated that as part of the 10-year visual examinations, it has not identified cracking in any Millstone 3 RV internal components.

Upper and Lower Core Plate Structural Analysis

Section IV.1.A.ii.g of Attachment 4 to the LAR states that in general thermal design transients, heat generation rates, and operating conditions affect thermal loads on the upper and lower core plates. For the MUR power uprate, the thermal design transients and heat generation rates remain applicable in the analysis of record because MUR power uprate conditions are bounded by the operating conditions in the current analysis of record. The licensee indicated that the maximum primary plus secondary stress intensity and cumulative usage factor of the upper and lower core plate remain acceptable. The licensee reported that the upper and lower core plates remain structurally adequate under MUR power uprate conditions.

In the June 2, 2021 submittal (Reference 2), the licensee stated that the upper and lower core plate evaluation performed in the SPU submittal bounds the upper core plate (UCP) and lower core plate (LCP) under MUR power uprate conditions. The licensee qualified the structural integrity of the UCP and LCP by determining the driving inputs which produce the stresses on the LCP and UCP. The licensee used the LCP and UCP geometry and material, core plate supports, heat generation rates and loads, fluid-thermal loads, thermal design transients (including number of cycles) and mechanical loads in the stress analysis for the LCP and UCP. For the MUR power uprate, the licensee determined that the above inputs have not changed, the change observed is negligible, or that the observed change is bounded by the analysis of record. The original LCP and UCP evaluation is documented in SPU LAR Attachment 5, Section 2.2.3 (Reference 51). Based on similarities in design and thermal loading, the licensee used the evaluation for the LCP and UCP at the similar plant in the current analysis for the SPU program. The licensee used the allowable stresses from Subsection NG of the 1974 edition of the ASME Code, Section III, which is consistent with the original evaluation.

Baffle-Barrel Region Evaluations

Section IV.1.A.ii.h of Attachment 4 to the LAR states that the baffle-barrel regions inside of the RV consist of a core barrel with installed former baffle plates. Former plates are bolted to the baffle and core barrel to restrain baffle plates motion. The licensee stated that the bolts on the former plates are subjected to primary and secondary loads. The primary loads are deadweight, hydraulic pressure differentials, LOCA and seismic loads. The secondary loads are preload and thermal loads resulting from RCS temperatures and gamma heating rates.

The licensee evaluated the maximum displacement of the baffle former bolts at MUR power uprate conditions. This displacement is caused by the temperature difference between the baffle and barrel regions, which is influenced by the power in the fuel assemblies adjacent to the baffle plates. The licensee's original analysis assumed that fresh fuel assemblies were loaded adjacent to the baffle. The licensee stated that power on the peripheral fuel assemblies is less than the initial power distribution because only irradiated assemblies are currently loaded in the peripheral core locations. The licensee explained that the core power distribution effect (i.e., lower power levels of peripheral fuel assemblies) offsets the increased loads due to gamma heating rates, resulting in a temperature difference less than the existing analysis of record. The licensee concluded that the existing thermal and structural analysis of the baffle-barrel region results remain bounding for the MUR power uprate design conditions.

In the June 2, 2021 submittal (Reference 2), the licensee stated that the baffle bolt evaluation in the SPU program was used to assess the structural acceptability of the baffle former bolts for the MUR power uprate. The SPU evaluation determined the cumulative fatigue damage resulting from the thermal loading. The licensee compared the cumulative fatigue damage factor to the allowable factor as discussed in its response to request for additional information (NVIB-RAI-4).

The licensee stated that it performs a VT-3 examination of RV core support barrel former baffle plates and baffle bolting during each 10-year inservice inspection (ISI) interval. The first interval examination was performed during the 3R05 refueling outage in June 1995, the second interval examination was performed during the 3R11 refueling outage in April 2007, and the third interval examination was performed during the 3R17 refueling outage in April 2016. The licensee stated that it did not find relevant indications on the core support barrel baffle plates or baffle bolting during any of these examinations.

3.3.2.2.3 Staff Evaluation and Conclusion

The NRC staff notes that in the June 2, 2021 submittal (Reference 2), the licensee explained that SPU involved a rated thermal power increase of 7 percent. The subject MUR power uprate LAR further increases the rated thermal power level by approximately 1.6 percent. The licensee stated that the MUR power increase considers changing the operating power plus uncertainty but remains bounded by the SPU operating power plus uncertainty. The licensee used the SPU evaluation results to justify the adequacy of various RV internal components and analyses under MUR power uprate conditions.

The NRC staff evaluated various RV internal components as discussed below.

Hydraulic Lift Forces and Pressure Losses

The NRC staff finds that the hydraulic lift forces and pressure losses of the RV internal components are acceptable because the hydraulic lift forces and downward forces on the various RV internal components remain stable and unchanged under MUR power uprate conditions.

Baffle Joint Momentum Flux and Fuel Rod Stability

Based on the information provided, the NRC staff finds that the baffle plate and fuel rods are acceptable under the MUR power uprate conditions because there is no significant change to the pressure differential across the baffle plate, baffle gap width, and fuel assembly model response due to MUR power uprate conditions.

Mechanical Evaluation

The NRC staff reviewed how the licensee evaluated FIV of the RV internals components such as core barrel assembly and upper internals under MUR power uprate conditions. The NRC staff determined that the existing SPU evaluation is applicable to MUR power uprate conditions and that the stresses on the RV internals as a result of FIV are within the allowable stresses of the ASME Code, Section III. Based on the above, the NRC staff has determined that the licensee has demonstrated that FIV under MUR power uprate conditions does not significantly affect the RV internals and that there is reasonable assurance of structural integrity of the RV internal components under MUR power uprate conditions.

Structural Evaluation

The NRC staff reviewed the impact of MUR power uprate conditions on the structural integrity of the RV internal components including the lower core support plate atypical region, lower support columns, and core barrel nozzle weldments. The NRC staff notes that for RV internal components, the licensee has demonstrated that the stresses and cumulative fatigue usage factor of the current analyses remain bounding for MUR power uprate conditions. The NRC staff has determined that the RV internals are bounded by the current analysis of record at MUR power uprate conditions based on the fact that the RV internals have been previously evaluated at a power level, which is greater than the proposed MUR power uprate power level. Additionally, a comparison between the RCS operating parameters before and after MUR power uprate implementation suggests that there should be a minimal impact on the loads used in the evaluation of the RV internals for structural integrity. The NRC staff determined that no abnormal loads (i.e., transient and seismic) are changing as a result of the MUR power uprate. The NRC staff finds that the structural integrity of RV internal components is not adversely affected by MUR power uprate conditions. Therefore, the NRC staff finds that the licensee has demonstrated reasonable assurance of the structural integrity of the RV internal components under MUR power uprate conditions

Upper and Lower Core Plate Structural Analysis

The NRC staff reviewed the impact of MUR power uprate conditions on structural integrity of the LCP and UCP in the RV. The NRC staff finds that the licensee has demonstrated the reasonable assurance of the structural integrity of the LCP and UCP under MUR power uprate conditions because the applied loads and resulting stresses of the LCP and UCP do not change significantly under MUR power uprate conditions.

Baffle-Barrel Region Evaluations

The NRC staff determined that the baffle bolt evaluation in the SPU program is applicable to the baffle former bolts for the MUR power uprate because the evaluation inputs have not changed or changed insignificantly. The NRC staff further determined that the cumulative fatigue usage factor in the SPU evaluation is well below the limit of 1.0 per the ASME Code, Section III. This result is applicable to MUR power uprate conditions because the SPU evaluation bounds MUR power uprate conditions. In addition, the licensee has not detected relevant indications in the core support barrel baffle plates or baffle bolting during any of the previous inservice examinations. Therefore, the NRC staff finds that the licensee has demonstrated reasonable assurance of the structural integrity of the baffle barrel components under MUR power uprate conditions.

3.3.2.3 Control Rod Drive Mechanisms and Bottom Mounted Instrumentation

3.3.2.3.1 Regulatory Evaluation

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

- 10 CFR 50, Appendix A, 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.

- 10 CFR 50, Appendix A, 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 during any condition of normal operation, including anticipated operational occurrences.

- Regulations in 10 CFR 50.55a require the RV be constructed, designed and analyzed in accordance with the ASME Code, Section III.

3.3.2.3.2 Licensee's Evaluation

Control Rod Drive Mechanisms (CRDMs)

Section IV.1.A.iii of Attachment 4 to the LAR discusses the licensee's evaluation of the updated design conditions (design parameters and NSSS design transients) for impact on the existing CRDM design basis structural analyses. According to the licensee, the code of record for the construction is the ASME Code, Section III, 1974 Edition through the Summer 1974 addenda. The licensee stated that CRDMs are subjected to T_{cold} temperatures and RCS pressures which are the only design parameters considered in the CRDM analysis. According to the licensee, the maximum T_{cold} from the MUR power uprate is 555.8 °F. The maximum operating temperature used in the analysis of record is 622.6 °F.

The licensee reported that the AOR remains bounding and applicable. The licensee did not change the RCS design or operating pressure as part of the MUR power uprate. As such, the licensee stated that the RCS temperature and pressure transients are not affected by the MUR power uprate. The licensee indicated that the original transient analysis remains bounding and applicable to MUR power uprate conditions. The licensee further stated that the stresses and cumulative fatigue usage factor of the existing analyses remain bounding at MUR power uprate conditions.

In the June 2, 2021 submittal (Reference 2), the licensee stated that the control rod evaluation determined the maximum mechanical load acting on the guide tubes. The licensee compared this load to the allowable load on the guide tubes. The licensee assessed the predicted lateral load/displacement against drop tests that applied lateral loads/deflections to the guide tube to confirm control rod insertion. These tests considered the effects of both temporary (elastic) and permanent (plastic) deformation. The licensee considered the applicable loadings--mass flow and acoustic loads, system loads, and safe shutdown earthquake (SSE) loads, which were used in the SPU program. Millstone 3 uses 17x17 array of 96-inch style guide tubes. The licensee indicated that the allowable loads and acceptance criteria for the guide tube and control rod insertion met for the SPU analysis which bound MUR power uprate conditions.

Bottom Mounted Instrumentation (BMI)

Section IV.1.A.iii.a of Attachment 4 to the LAR evaluates the BMI guide tubes for stresses associated with MUR power uprate conditions and compared the applied stresses to the allowable stresses in accordance with the ASME Code, Section III. The licensee stated that the applied stresses are affected by RV core inlet temperature, RV pressure, and NSSS design transients. The range of RV core inlet temperatures for the MUR power uprate is 536.7 °F to 555.8 °F, which is lower than the RV core inlet temperature in the existing analysis. These RV core inlet temperatures are bounded by the BMI guide tube design temperature of 560 °F.

The licensee stated that thermal and pressure stresses of the BMI guide tube do not increase under MUR power uprate conditions. The RCS pressure for the MUR power uprate is 2250 psia, which is bounded by the BMI guide tube design pressure of 2500 psia. The licensee did not change the NSSS design transients as part of the MUR power uprate. The licensee concluded that the existing BMI guide tubing/flux thimble design bounds MUR power uprate conditions with no changes to the applicable methodology, results, or margin of safety.

In the June 2, 2021 submittal (Reference 2), the licensee stated that the BMI guide tubes stress analysis is based on the design temperature of 560 °F which bounds the pre- and post-MUR power uprate core inlet temperature of 536.7 °F to 555.8 °F. The design pressure of 2500 psia for the BMI guide tube bounds the RCS pressure for MUR power uprate conditions of 2250 psia. In addition, the NSSS design transients were not changed as part of the MUR power uprate program.

3.3.2.3.3 Staff Evaluation and Conclusion

Control Rod Drive Mechanisms (CRDMs)

The NRC staff noted that CRDMs are subjected to T_{cold} temperatures and RCS pressures. According to the licensee, the maximum T_{cold} from the MUR power uprate design parameters is 555.8 °F. The maximum operating temperature used in the analysis of record is 622.6 °F. As stated above, the MUR power uprate did not change the RCS design or operating pressure significantly. The NRC staff determined that the RCS temperature and pressure transients are not significantly affected by the MUR power uprate. The NRC staff further determined that the original transient analysis remains bounding and applicable to MUR power uprate conditions. Therefore, the stresses and cumulative fatigue usage factor of the existing analyses remain bounding and applicable for MUR power uprate conditions. Based on the above, the NRC staff finds that the MUR power uprate does not affect the structural integrity of CRDMs and that the licensee has demonstrated reasonable assurance of structural integrity of the CRDMs under MUR power uprate conditions

Bottom Mounted Instrumentation (BMI)

The NRC staff reviewed the licensee's stress analysis for the BMI guide tubes/flux thimble using MUR power uprate conditions and compared the applied stresses to the allowable stresses per Section III of the ASME code. The applied stresses on the BMI guide tubes/flux thimble are affected by three parameters---vessel core inlet temperature, reactor pressure, and NSSS design transients. Because the temperature and pressure used in the stress analysis are bounding, the NRC staff determined that thermal stress and pressure stress of the BMI guide tube/flux thimble do not increase associated with MUR power uprate conditions. Based on the above, the NRC staff finds that the licensee has demonstrated reasonable assurance of the structural integrity of the BMI guide tubing/flux thimble under MUR power uprate conditions.

3.3.2.4 Pressurized Thermal Shock (PTS) Calculations

3.3.2.4.1 Regulatory Evaluation

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

- Regulations in 10 CFR 50.61, "Fracture toughness requirements for protection against pressurized thermal shock events," provides the requirements, methods of evaluation, and safety criteria for PTS assessments. Regulations in 10 CFR 50.61 require PTS evaluations to ensure that adequate fracture toughness exists for RV beltline materials in PWRs to protect against failure during a PTS event. Fracture resistance of RV beltline materials during PTS events is evaluated by calculating the nil-ductility temperature (RTNDT) for PTS, identified as RTPTS. 10 CFR 50.61(a) defines RTPTS as the RTNDT evaluated for the "EOL [end-of-life] Fluence" for each RV beltline material using the calculation procedures required by 10 CFR 50.61(c). 10 CFR 50.61(a) defines "EOL fluence" as the neutron fluence projected for a specific RV beltline material at the clad-base metal interface on the inside surface of the RV at the location where the material receives the highest neutron fluence on the expiration date of the operating license.

- 10 CFR 50.61(b)(1) requires that PWR licensees have projected values of RTPTS accepted by the NRC for each RV beltline material.

- 10 CFR 50.61(c)(2) requires that RTPTS calculations for RV beltline materials incorporate credible RV surveillance material test data that are reported as part of the RV materials surveillance program required by 10 CFR Part 50, Appendix H.

- The PTS screening criteria are the values of RTPTS for the RV beltline materials above which the plant cannot continue to operate without justification and approval by the NRC pursuant to 10 CFR 50.61(b). 10 CFR 50.61(b) specifies that the PTS screening criteria are 270 °F for plates, forgings, and axial weld materials and 300 °F for circumferential weld materials. A PWR licensee may demonstrate compliance with 10 CFR 50.61 requirements by showing that their RTPTS values are less than the PTS screening criteria at the expiration of the operating license.

- The RV beltline is applicable to all ferritic RV materials with projected neutron fluence values greater than 1 x 1017 n/cm2 (E > 1.0 MeV). For PTS evaluations, this fluence threshold remains applicable for the duration of the licensed operating period. The NRC staff's basis for this fluence threshold is provided in in Regulatory Issue Summary (RIS) 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components." (Reference 50)

- 10 CFR 50.61(b) requires that licensees update their PTS evaluations whenever there is a significant change in operating conditions that affect the projected values of RTPTS.

Therefore, for power uprate amendment applications, the NRC staff reviews the impact of the changes in operating conditions on the licensee's calculation of the RTPTS values for all RV beltline materials.

3.3.2.4.2 Licensee's Evaluation

Section IV.1.C.i of Attachment 4 to the LAR presents PTS screening calculations for the RV beltline and extended beltline materials using the neutron fluence values at the end of the current 60-year license. The licensee determined that all the beltline materials will continue to meet the PTS screening criteria in 10 CFR 50.61. The licensee indicated that the limiting RTPTS value of 130 °F applies to lower shell plate B9820-2. The licensee explained that this is a change from the analysis of record that had a limiting RTPTS value of 133 °F pertaining to intermediate shell plate B9805-1. The licensee stated that the PTS screening calculations result in RTPTS values that are consistent with those documented in the RV integrity analysis of record. The licensee further stated that the MUR power uprate has no impact on 10 CFR 50.61 compliance and that the RV will remain within its PTS screening criteria limits after implementation of the MUR power uprate.

In the June 2, 2021 submittal (Reference 2), the licensee provided a historical background and noted that the analysis of record for RTPTS is contained in Table 2.1.3-4 of Attachment 5 in the SPU LAR (Reference 51). The limiting RTPTS value is also shown in Table 5.2-7 of the

Millstone 3 FSAR. Section 2.1.3.2.4 in Attachment 5 of the SPU LAR (Reference 51) shows that the limiting material is intermediate shell plate B9805-1 because its RTPTS value was calculated using the RG 1.99, Revision 2 (Reference 52), Position 1.1 chemistry factor (i.e., chemistry factor calculated without the use of surveillance data), in lieu of the Position 2.1 chemistry factor (chemistry factor calculated from credible surveillance data). The licensee stated that the most limiting (highest) RTPTS value for the analysis of record at 54 effective full power years (EFPY) for intermediate shell Plate B9805-1 is 133 °F based on Position 1.1 chemistry factor. Table 2.1.3-4 of Attachment 5 in the SPU LAR also presents an RTPTS value of 111 °F for intermediate shell plate B9805-1 using Position 2.1 chemistry factor. The licensee stated that at the time, the SPU LAR Attachment 5 conservatively reported RTPTS of 133 °F for intermediate shell plate B9805-1, instead of the 111 °F value. According to the licensee, if the SPU report had elected to take credit for the credible surveillance data, intermediate shell plate B9805-1 would no longer have been the most limiting material. Instead, the limiting RTPTS value would have been 130 °F which is associated with lower shell plate B9820-2.

For the subject MUR power uprate LAR, the licensee re-calculated the RTPTS for RV shell material as shown in Table 1-1 in the June 2, 2021 submittal (Reference 2). The licensee used neutron fluence value derived from MUR power uprate conditions to obtain RTPTS values. The licensee stated that the peak 54 EFPY neutron fluence value increased from a value of 2.70 x 1019 n/cm2 (SPU conditions) to 2.72 x 1019 n/cm2 (MUR power uprate conditions). The licensee reported that because the RTPTS values are reported as whole numbers, this slight increase in neutron fluence did not change the highest calculated RTPTS values. Specifically, the RTPTS values corresponding to intermediate shell plate B9805-1 using Position 1.1 (133 °F), intermediate shell plate B9805-1 using Position 2.1 (111 °F), and lower shell plate B9820-2 (130 °F) are unchanged from the neutron fluence increase. The licensee stated that the Position 2.1 chemistry factor for intermediate shell plate B9805-1 for both the SPU and MUR power uprate conditions use the same surveillance data in Westinghouse Report, WCAP-16629-NP, Revision 0, "Analysis of Capsule W from the Dominion Nuclear Connecticut Millstone 3 Reactor Vessel Radiation Surveillance Program," September 2006. The licensee reported that the limiting RTPTS value for MUR power uprate conditions is 130 °F, which is associated with lower shell plate B9820-2.

3.3.2.4.3 Staff Evaluation and Conclusion

The NRC staff recognized that the licensee calculated the limiting reference temperature for PTS (RTPTS) value of 130 °F for lower shell plate B9820-2 based on the neutron fluence for MUR power uprate conditions. The NRC staff acknowledged that the change in the limiting RTPTS value and RV plate material between the analysis of record and the subject MUR power uprate submittal is the result of using credible surveillance data, and is not a result of changes in neutron fluence projections, surveillance data availability, or material properties. The NRC staff notes that the lower the RTPTS value, the less damage to the RV shell material during a PTS event.

The NRC staff verified that the licensee correctly recalculated the RTPTS values for RV beltline materials in accordance with 10 CFR 50.61(c) and RG 1.99, Revision 2 (Reference 52) as shown in in Table 1-1 in the June 2, 2021 submittal (Reference 2). The NRC staff determined that the limiting RTPTS value for MUR power uprate conditions is less than the applicable PTS screening criteria of 270 °F for plates, forgings, and axial weld materials and 300 °F for circumferential weld materials in 10 CFR 50.61. As such, the NRC staff verified that the RTPTS values for all RV beltline materials will continue to remain less than the applicable

PTS screening criteria. Therefore, the NRC staff finds that the RV beltline materials will continue to meet the PTS requirements of 10 CFR 50.61 under MUR power uprate conditions.

3.3.2.5 Heatup and Cooldown Pressure/Temperature Limit Curves

3.3.2.5.1 Regulatory Evaluation

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

- Regulations in 10 CFR Part 50, Appendix G, require pressure-temperature (P-T) limits be established for the RCS to protect the reactor coolant pressure boundary (RCPB) against brittle fracture during normal operation, anticipated operational occurrences, and hydrostatic tests. Section IV.A.2 of 10 CFR Part 50, Appendix G requires that the P-T limits for operating reactors be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of the ASME Code, Section XI, Appendix G.

- RIS 2014-11, “Information On Licensing Applications For Fracture Toughness Requirements For Ferritic Reactor Coolant Pressure Boundary Components,” (Reference 50) provides additional NRC recommendations for evaluation of P-T limits in licensing applications, including guidance for considering the higher stress effects for complex geometry components, such as RV nozzles. The definition of the “RV beltline” is applicable to all ferritic RV materials with projected neutron fluence values greater than 1 x 1017 n/cm2 (E > 1.0 MeV).

- RG 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” (Reference 52) contains methodologies determined by the NRC staff to be acceptable for determining the increase in transition temperature (i.e., adjusted RTNDT for RV beltline materials) and the decrease in USE resulting from neutron irradiation.

  Regulatory Guide 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” provides state-of-the-art calculations and measurement procedures that are acceptable to the NRC staff for determining RV neutron fluence (Reference 53).

3.3.2.5.2 Licensee’s Evaluation

Section IV.1.C.iii of Attachment 4 to the LAR states that the current heatup and cooldown curves are given in Figures 3.4-2 and 3.4-3 of the Technical Specifications (TS). The licensee developed the heatup and cooldown curves using the methodology in the ASME Code, Section XI, Appendix G, 1995 Edition, as augmented by ASME Code, Case N-640, “Alternative Reference Fracture Toughness for Development of P-T Limit Curves.” The licensee stated that beltline material properties degrade with radiation exposure, and this degradation is measured in terms of the adjusted reference nil-ductility temperature, which includes a reference nil-ductility temperature shift. The licensee calculated the adjusted reference temperature (ART) per RG 1.99, Revision 2 (Reference 52) for RV beltline materials for neutron fluence values corresponding up to a cumulative operating time of 32 effective full power year (EFPY), to provide consistency with the current operating time of the TS RCS pressure/temperature limit curves. The licensee stated that the current heatup and cooldown curves and cold overpressure protection system setpoints are bounding through 32 EFPY with the MUR power uprate and do not require update, because fluence values for all RV analyses are bounded by the analysis of record up to 32 EFPY. The licensee further stated that it plans to develop new pressure/temperature curves for beyond 32 EFPY and will submit the revised curves in a separate license amendment request in the future.

3.3.2.5.3 Staff Evaluation and Conclusion

The NRC staff notes that the ASME Code, Section XI, Appendix G specifies a procedure for calculating P-T limits that is based on linear elastic fracture mechanics (LEFM). The critical material property used in the P-T limit calculation is the fracture toughness (KIC). As specified in Paragraph G-2210 of the ASME Code, Section XI, KIC is an exponential function of the difference in metal temperature at the postulated crack tip and the reference nil-ductility temperature (RTNDT) for the ferritic RV material. Section IV.A of 10 CFR Part 50, Appendix G requires that the values of RTNDT for RV beltline materials used in the P-T limit calculations account for the effects of neutron radiation and incorporate credible RV surveillance material test data that are reported as part of the RV materials surveillance program required by 10 CFR Part 50, Appendix H.

Neutron irradiation of RV beltline materials will increase their RTNDT values, thereby causing a rightward shift in the KIC curve and a corresponding rightward shift (more restrictive) in the P-T limit curve. Given the shift of the P-T limit curve caused by neutron irradiation, the reactor coolant system is operated at lower pressures and/or higher temperatures (i.e., below and/or to the right of the curve) in order to maintain the required safety margins for protection of the RV material against brittle fracture per 10 CFR Part 50, Appendix G.

For the proposed MUR power uprate conditions, the NRC staff verified that the licensee calculated appropriate adjusted RTNDT values, consistent with the methods in RG 1.99, Revision 2 (Reference 52), for the limiting RV beltline shell materials and the RV inlet and outlet nozzles. The NRC staff further verified that the licensee's adjusted RTNDT for MUR power uprate conditions demonstrate that the P-T limits were constructed based on the limiting RV beltline shell material that bounds all relevant RV materials. The NRC staff verified that the licensee correctly addressed the impact of MUR power uprate conditions for the existing P-T limits at Millstone 3. Therefore, the NRC staff finds that the licensee's existing P-T limits will remain in compliance with 10 CFR Part 50, Appendix G for MUR power uprate conditions through 32 EFPY.

3.3.2.6 Low-Temperature Overpressure Protection

3.3.2.6.1 Regulatory Evaluation

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

- Regulations in 10 CFR Part 50, Appendix G, require P-T limits be established for the RCS to protect the reactor coolant pressure boundary (RCPB) against brittle fracture during normal operation, anticipated operational occurrences, and hydrostatic tests. Section IV.A.2 of 10 CFR Part 50, Appendix G requires that the P-T limits for operating reactors be at least as conservative as limits obtained by following the methods of analysis and the margins of safety of the ASME Code, Section XI, Appendix G.

- RIS 2014-11 (Reference 50) provides additional NRC recommendations for evaluation of P-T limits in licensing applications, including guidance for considering the higher stress effects for complex geometry components, such as RV nozzles. The definition of the “RV beltline” is applicable to all ferritic RV materials with projected neutron fluence values greater than 1 x 1017 n/cm2 (E > 1.0 MeV).

- RG 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” (Reference 52) contains methodologies for determining the increase in transition temperature (i.e., adjusted RTNDT for RV beltline materials) and the decrease in USE resulting from neutron irradiation.

- RG 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” provides state-of-the-art calculations and measurement procedures that are acceptable to the NRC staff for determining RV neutron fluence (Reference 53).

3.3.2.6.2 Licensee’s Evaluation

Section IV.1.C.iv of Attachment 4 to the LAR states that the cold overpressure protection system (COPS), also known as the low-temperature overpressure protection system (LTOPS), provides RCS pressure relief capability during low-temperature operation. The licensee stated that TS 3.4.9.3 requires cold overpressure protection of the RCS system in Mode 4 when the RCS cold leg temperature is less than or equal to 226 °F, at the Mode 5 and Mode 6 operation. The licensee further stated that the full power design parameters have no impact on the LTOPS transients. According to the licensee, the only potential impact on the LTOPS setpoint determination would be if the uprate resulted in higher RV neutron fluence and more restrictive P-T limit curves. The licensee indicated that the current LTOPS setpoints are bounding through 32 EFPY with MUR power uprate conditions and do not require to be changed because the neutron fluence values and P-T limit curves are bounded by the analysis of record.

3.3.2.6.3 Staff Evaluation and Conclusion

The NRC staff notes that the TS requirements ensure LTOP system pressure relief capacity is not affected by the MUR power uprate. However, the LTOP system applicability temperature is determined based on the adjusted RTNDT for the liming RV beltline material, per the ASME Code, Section XI, Appendix G. Thus, the EFPY term for the LTOP system applicability temperature should be consistent with that used to determine limiting adjusted RTNDT value used in the P-T limit curves. The NRC staff finds that the licensee’s LTOP system applicability temperature, consistent with that used for the limiting adjusted RTNDT values and P-T limits, ensures that the applicability temperature is valid for MUR power uprate conditions. Therefore, the NRC staff finds that the LTOP system at Millstone 3 will continue to meet the requirements of 10 CFR Part 50, Appendix G for MUR power uprate conditions.

3.3.2.7 Upper Shelf Energy (USE) Calculation

3.3.2.7.1 Regulatory Evaluation

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

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," requires certain fracture toughness and Charpy upper-shelf energy (USE) be maintained for RV beltline materials against failure. Appendix G also requires RCS P-T limits be implemented to protect against brittle fracture. Specifically, Section IV.A.1 of 10 CFR Part 50, Appendix G, requires that RV beltline materials have Charpy USE in the transverse direction for base material and along the weld for weld material greater than or equal to 75 foot-pounds (ft-lbs) in the unirradiated condition. The rule also requires that RV beltline materials maintain Charpy USE greater than or equal to 50 ft-lbs throughout the operating life of the RV, unless it is demonstrated in a manner approved by the NRC that lower values of USE would provide margins of safety against fracture equivalent to those required by the ASME Code, Section XI, Appendix G. The analysis to demonstrate acceptable margins of safety against fracture is often referred to as an equivalent margins analysis.

Section IV.A of 10 CFR Part 50, Appendix G, requires that the USE values for RV beltline materials account for the effects of neutron radiation and incorporate credible RV surveillance material test data that are reported as part of the RV materials surveillance program required by 10 CFR Part 50, Appendix H. The NRC's guidelines for calculating the effects of neutron radiation on the USE values for the RV beltline materials are provided in RG 1.99, Revision 2 (Reference 52).

3.3.2.7.2 Licensee's Evaluation

Section IV.1.C.v of Attachment 4 to the LAR states that the USE of the RV beltline materials is evaluated to ensure compliance with 10 CFR 50, Appendix G. The licensee stated that the limiting RV beltline material is lower shell plate B9820-2. The licensee has withdrawn and tested three in-vessel surveillance capsules. The licensee has also withdrawn two additional capsules, but they have not been tested. The licensee stated that one surveillance capsule, Capsule Z, is remaining in the RV. As a validation of the current analysis of neutron exposure, the licensee used the measured reaction rates in conjunction with the current neutron spectra for each withdrawn capsule as input to the NRC-approved least squares dosimetry evaluation methodology. The licensee indicated that for neutron fluence (E >1.0 MeV), the best estimate to calculated ratios span a range from 1.04 to 1.05 with an average best estimate to calculated ratios of 1.04 +/- 0.9 percent (one sigma) for the three sets of capsule data. These comparisons fall well within the +/- 20 percent criterion specified in RG 1.190 (Reference 53), thus validating that the current calculations are applicable to the RV. The licensee provided the end-of-life Charpy USE results in Table IV-3 of the MUR power uprate submittal, where the decrease in Charpy USE due to peak end-of-life fluence at the ¼ T location (T is the wall thickness of RV beltline shell) is calculated from the trend curves in RG 1.99, Revision 2, Figure 2 (Reference 52). The licensee stated that the ¼ T USE values for the RV beltline and extended beltline materials under MUR power uprate conditions exceed 50 ft-lb thus continuing to meet the 10 CFR 50, Appendix G, acceptance criterion at the end of the current 60-year license period.

3.3.2.7.3 Staff Evaluation and Conclusion

For power uprate applications, the NRC staff reviews the impact of the increase in power on the licensee's calculation of projected USE values and, if applicable, the equivalent margins analysis results for the RV beltline materials. As such, the NRC staff verified that the licensee correctly recalculated the projected USE values for all RV beltline materials to address MUR power uprate conditions by incorporating 32 EFPY neutron fluence values that were generated using the new 3-D neutron fluence model. The NRC staff verified that the licensee's 32 EFPY USE calculations for MUR power uprate conditions demonstrate that all RV beltline materials will maintain USE values greater than 50 ft-lbs throughout the operating life of the RV. Therefore, the NRC staff finds that the licensee's USE evaluation for the RV beltline materials will continue to meet the requirements of Section IV.A.1 of 10 CFR Part 50, Appendix G, for MUR power uprate conditions.

3.3.2.8 Neutron Fluence Evaluation

The neutron fluence evaluation is addressed in Section IV.1.C.ii of Attachment 4 to the LAR.

3.3.2.8.1 Regulatory Evaluation

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.

RG 1.190, which provides state-of-the-art calculations and measurement procedures that are acceptable to the NRC staff for determining pressure vessel fluence. (Reference 53).

3.3.2.8.2 Licensee's Evaluation

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.

The fluence evaluation is addressed in Section IV.1.C.ii of Attachment 4 to the LAR. The licensee stated that the fast neutron exposure evaluations for the Millstone 3 reactor vessel are performed using a series of fuel cycle-specific forward transport calculations using a three-dimensional flux synthesis technique. The MUR power uprate neutron fluence values for the

Millstone 3 beltline and extended beltline RV materials are provided in Tables IV-1 and IV-2 of Attachment 4 to the LAR.

A comparison of the analysis of record (AOR) data is provided in Table IV-1 and indicates that for 54 EFPY of plant operation, there is a small increase in the maximum projected neutron exposure at the 15, 30, and 45 degree azimuthal locations compared to the AOR fluence projections. The zero degree azimuthal location experienced no significant increase in exposure. The maximum neutron fluence (E > 1.0 MeV) projected for each of the beltline materials, based on 54 and 60 EFPY of operation for the MUR power uprate at 3,723 MWt is provided in Table IV-2. The MUR power uprate extended beltline material exposures are within 9 percent of the AOR exposures, with the maximum percentage difference occurring for the Intermediate Shell to Nozzle Shell Circumferential Welds at the 45 degree azimuthal angle. The overall change in fluence projections resulting from the MUR power uprate are in all cases less than 9 percent greater than the AOR.

The licensee also stated that these calculations were performed in accordance with WCAP-14040-A, Revision 4 (Reference 54). These methods have been approved for use by the NRC staff and adhere to RG 1.190 (Reference 53).

3.3.2.8.3 Staff Evaluation and 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. 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.

3.3.2.9 Surveillance Capsule Withdrawal Schedule

3.3.2.9.1 Regulatory Evaluation

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

Regulations in Appendix H to 10 CFR Part 50, "Reactor Vessel Material Surveillance Program Requirements," requires the material surveillance program be implemented in the RV to monitor changes in the fracture toughness properties of ferritic materials in the RV beltline region that result from exposure of these materials to neutron irradiation and the thermal environment. Fracture toughness test data are obtained from testing of material specimens exposed to neutron irradiation in surveillance capsules, which are withdrawn periodically from the RV. The test data are used to evaluate fracture toughness of RV material to demonstrate compliance with 10 CFR 50.61 and 10 CFR Part 50, Appendix G.

Regulations in 10 CFR Part 50, Appendix H, require the surveillance program complying with American Society for Testing and Materials (ASTM) E185, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." For each capsule withdrawal, the test procedures and reporting requirements must meet the requirements of ASTM E185-82 to the extent practicable for the configuration of the specimens in the capsule.

The NRC published recommendations for RV material surveillance programs for the 60-year renewed license in Section XI.M31 of NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report" (Reference 55). GALL Revision 2 recommends that the RV surveillance program should have at least one capsule with a projected neutron fluence equal to or exceeding the 60-year peak RV wall neutron fluence prior to the end of the period of extended operation.

3.3.2.9.2 Licensee's Evaluation

Section IV.1.C.vi of Attachment 4 to the LAR states that the Millstone 3 RV material surveillance program has a withdrawal schedule to periodically remove surveillance capsules from the RV and to monitor the RV materials condition under actual operating conditions. The licensee stated that the withdrawal schedule is consistent with ASTM E 185-82.

By letter dated October 2, 2006 (Reference 56), the licensee submitted the analysis for the third surveillance capsule documented in WCAP-16629-NP, "Analysis of Capsule W from the Dominion Nuclear Connecticut Millstone 3 Reactor Vessel Radiation Surveillance Program." The licensee stated that this report shows that the surveillance capsule monitoring program requirements are satisfied through end of life (EOL), including MUR power uprate neutron fluence projections. The licensee further stated that the current capsule withdrawal schedule remains valid for MUR power uprate conditions. The licensee concluded that there are no changes necessary to the capsule withdrawal schedule as shown in Millstone 3 FSAR Section 5.3.1.6.

3.3.2.9.3 Staff Evaluation and Conclusion

The NRC staff evaluated the impact of the MUR power uprate conditions on the licensee's RV surveillance capsule withdrawal schedule. The NRC staff verified that the licensee has withdrawn and tested three in-vessel surveillance capsules as stated in Section IV.1.C.v, "Upper Shelf Energy (USE) Calculation," of Attachment 4 to the LAR. The licensee has also withdrawn two additional capsules which have not been tested. The NRC staff noted that one surveillance capsule (Capsule Z) is remaining in the RV. Based on the review of WCAP-16629-NP, the NRC staff finds that the withdrawn surveillance capsules have been tested in accordance with ASTM E 185-82. The NRC staff verified that the surveillance capsule withdrawal schedule remain acceptable for meeting the mandated criteria in 10 CFR 50, Appendix H and the recommended criteria in GALL Revision 2 (Reference 55) considering the increase in the neutron fluence and RTNDT shift for the limiting RV beltline materials under MUR power uprate conditions. Therefore, the NRC staff finds that the RV material surveillance program will continue to meet the requirements of 10 CFR Part 50, Appendix H for MUR power uprate conditions.

3.3.3 Chemical Engineering and Steam Generator Integrity

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.

3.3.3.1 Chemical and Volume Control System

3.3.3.1.1 Regulatory Evaluation

The chemical and volume control system (CVCS) provides a means for: (1) maintaining water inventory and quality in the RCS, (2) supplying seal-water flow to the Reactor Coolant Pumps 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 control of primary water chemistry. The NRC's review criteria are based on 10 CFR 50 Appendix A GDC 14, "Reactor coolant pressure boundary." GDC 14 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." Specific review criteria are contained in the Standard Review Plan, NUREG-0800, Section 9.3.4, “Chemical and Volume Control System, Section 9.3.4, (PWR). (Reference 57) (SRP).

Additional information regarding the design of the CVCS system can be found in the MPS3 FSAR Section 9.2.1, “Chemical Volume and Control System.

3.3.3.1.2 Licensee Description

In Table II-1, “FSAR Accidents, Transients and Other Analyses,” the licensee states, the current CVCS Malfunction FSAR analysis was performed at NSSS power level of 3,739 MWt, which is bounding of the MUR proposed power level of 3,709 MWt. The analysis was approved by letter dated August 12, 2008, “Millstone Power Station, Unit No. 3 – Issuance of Amendment 242 in support of the SPU (Reference 4).

3.3.3.1.3 Staff Evaluation and Conclusion

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed MUR on the CVCS and reviewed the previous analysis performed and finds that the previous analysis is bounding of the current proposal of 3,709 MWt. Additionally, as stated in the NRC staff’s SE approving the SPU (Reference 4), the licensee has adequately addressed changes in the temperature of the reactor coolant and their effects on the CVCS. The NRC staff further concludes that the licensee has demonstrated that the CVCS will continue to be acceptable and will continue to meet the requirements of GDC-14 following implementation of the proposed MUR power uprate. Therefore, the NRC staff finds the proposed MUR acceptable with respect to the CVCS.

3.3.3.2 Steam Generator Blowdown System

3.3.3.2.1 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 during plant operation. 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's review criteria for the SGBS are based on 10 CFR 50 Appendix A, GDC 14, as it requires that the reactor coolant pressure boundary (RCPB) be designed to have an extremely low probability of abnormal leakage, of rapidly propagating fracture, and of gross rupture. SRP Section 10.4.8, "Steam Generator Blowdown System," provides review guidance for the NRC staff (Reference 58). Additionally, the SGBS is described in Section 10.4.8, "Steam Generator - Blowdown Processing System," of the Millstone 3 FSAR.

3.3.3.2.2 Licensee Description

In its LAR, the licensee stated that Millstone 3 will continue to operate the SGBS per the plant chemistry program following implementation of the MUR power uprate, with no change in blowdown flowrate. Additionally, the licensee states that the effect of any potential impurity increase in the SGBS due to MUR power uprate will be evaluated under the plant chemistry program. Under MUR power uprate conditions, the SGBS operating temperatures and pressures change insignificantly and remain bounded by existing design parameters.

3.3.3.2.3 Staff Evaluation and Conclusion

The NRC staff reviewed the impacts from the proposed MUR power uprate on the SGBS and the effects on secondary water chemistry control. The NRC staff also reviewed FSAR Sections 10.3.5 and 10.4.8. The NRC staff finds the impacts from the proposed MUR power uprate on the SGBS ability to maintain secondary water chemistry acceptable because the secondary water chemistry program incorporates the EPRI secondary water chemistry guidelines as stated in FSAR Section 10.3.5. Additionally, 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 NRC staff finds there is reasonable assurance that the SGBS will be able to achieve the necessary flow rates to maintain secondary water chemistry.

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.

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

3.3.3.3.1 Regulatory Evaluation

SG tubes constitute a large part of the reactor coolant pressure boundary. As a result, their integrity is important to the safe operation of the 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 Millstone 3 TSs. Millstone 3 TS 3.4.5, "Steam Generator (SG) Tube Integrity," and 6.8.4(g), "Steam Generator (SG) Program," govern the SG inspections for Millstone 3. Details of the Millstone 3 SGs can be found in FSAR Section 4.3.2, "Steam Generators." Specific review criteria for this topic are contained in the SRP, Section 5.4.2.1, "Steam Generator Materials," for the SG materials, and Section 5.4.2.2, "Steam Generator Program," for the SG program (Reference 59). Additionally, RIS 2002-03 recommends 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 fracture toughness of the ferritic materials is adequate, (3) the design of the steam generator limits the susceptibility of the materials to degradation and corrosion, (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.

3.3.3.3.2 Licensee Description

The Millstone 3 Model F SGs contain Alloy 600 TT tubing and corrosion resistant stainless steel 405 alloy tube support plates with a four lobe hole quatrefoil design. In its LAR, the licensee stated, the thermal-hydraulic evaluation concluded that the Millstone 3 SG thermal-hydraulic operating characteristics remain acceptable at MUR power uprate conditions. It also stated that the existing Millstone 3 analysis of record remains applicable and bounding for MUR power uprate conditions and the SG components will continue to meet the applicable ASME B&PV code fatigue and stress limits. The proposed MUR power uprate conditions (e.g. steam flow, temperature, pressure) are described in Table IV-6, "MPS3 CHECKWORKS Fluid Parameters and Wear Rate Comparison – Pre and Post MUR power uprate." This table shows a slight increase in certain temperatures and flow rates from the current operating conditions.

3.3.3.3.3 Staff Evaluation and Conclusion

The NRC staff evaluated the material provided by the licensee and determined that the changes in operating conditions at the proposed MUR power uprate conditions would be relatively small. The changes in operating conditions for the proposed MUR power uprate are described in Table IV-6 of Attachment 4 to the LAR. Further, the new operating temperatures and pressures are typical of those used by other plants, which the NRC staff has already approved for use. Similar SGs have operated successfully under these conditions.

With respect to the SG materials, the NRC staff has determined that the materials used in the SG remain acceptable, the fracture toughness of the ferritic materials is adequate, the design still 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, the NRC staff has determined that the impact of the power uprate on SG tube vibration and fatigue 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 SG 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 to manage such degradation, since the SG Program still requires the licensee 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 by the NRC staff to have a small, if any, effect on the structural limits for the tubes. Since the tube repair criterion is determined from the structural limit, it may also be slightly affected by the MUR power uprate conditions. Although this analysis was not reviewed by the NRC staff in detail, the NRC staff has determined that the tube repair criteria remain valid under the MUR power uprate conditions. This determination is based on NRC staff's approval of similar 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 effect of implementation 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 implementation of the MUR power uprate. Therefore, the NRC staff concludes that the LAR is acceptable with respect to the SG tube material and program.

3.3.3.4 Flow-Accelerated Corrosion

3.3.3.4.1 Regulatory Evaluation

Flow-accelerated corrosion (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 the FAC program at Millstone 3 is based on the Electric Power Research Institute (EPRI) NSAC-202L recommendations (Reference 60). Section 19.2.1.10, "Flow Accelerated Corrosion," of the licensee's FSAR provides additional detail on the Millstone FAC program and the basis for the program.

3.3.3.4.2 Licensee Description

In Section IV.1.E.iii, "Flow Accelerated Corrosion Program," of Attachment 4 to the LAR, the licensee states that the established FAC program at Millstone 3 meets the intent of the EPRI's recommendations and best practices. The licensee also stated that Millstone 3 utilizes CHECWORKS™ Steam/Feedwater Application computer code to assist in monitoring FAC susceptible piping and components. Additionally, the licensee provided sample results from the updated CHECWORKS™ model considering the proposed MUR power uprate conditions. The licensee also stated that no significant impact to component service life has been identified based on the reviews conducted to determine the impact of increased component wear rates. No additional secondary system lines were identified as requiring monitoring for FAC wear as a result of the MUR power uprate. The remaining service life for the modeled FAC susceptible components will continue to be monitored and documented in accordance with the FAC monitoring program.

3.3.3.4.3 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 staff finds the basis for the FAC program acceptable because, as per Chapter 19 of the Millstone 3 FSAR, it is consistent with Revision 0 of the Generic Aging Lessons Learned (GALL) Report, dated July 2001, which is not impacted by the MUR power uprate (Reference 55). The GALL Report recommends use of EPRI NSAC-202L, "Recommendations for an Effective Flow-Accelerated Corrosion Program," which the NRC staff found acceptable as a basis for a FAC program. The staff also recognize that although the 5th point extraction steam piping shows a large percent increase in wear, the actual value of these changes (mils/year) are relatively small with respect to the component thickness. The licensee also stated that these wear rates are within the industry guidelines, and the increase in wear rates anticipated as a result of the MUR power uprate will not significantly alter the remaining service life. Additionally, there are no planned changes to the 3R22 (fall 2023) and 3R23 (spring 2024) FAC examination and replacement schedules. Therefore, the NRC staff finds that the FAC program will provide reasonable assurance that components susceptible to FAC will be managed appropriately post MUR power uprate implementation. The NRC staff concludes the proposed MUR power uprate is acceptable, with respect to the impacts on the FAC Program.

3.3.3.5 Rapidly Propagating Fatigue Cracks in Steam Generator Tubes

3.3.3.5.1 Regulatory Evaluation

Section IV.1.F of Attachment 1, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," to RIS 2002-03, states that licensees should address if the effect of the power uprate on steam generator (SG) tube high cycle fatigue is consistent with NRC Bulletin (BL) 88-02, "Rapidly Propagating Fatigue Cracks in Steam Generator Tubes," dated February 5, 1988 (Reference 61).

NRC BL 88-02 was addressed to all licensees of Westinghouse designed reactors with SGs that utilize carbon steel support plates. The bulletin described a SG tube rupture (SGTR) event at

North Anna Power Station, Unit No. 1 that was caused by rapidly propagating fatigue cracks due to high cycle fatigue. The bulletin described actions for the addressees to take in order to minimize the potential for a SGTR, such as the one that occurred at North Anna Power Station, Unit No. 1.

3.3.3.5.2 Licensee Description

In Section IV.1.F of its LAR, the licensee stated that the Millstone 3 Model F SGs contain Alloy 600 thermally treated (TT) tubing and stainless steel 405 alloy tube support plates with a four lobe hole quatrefoil holes. The quatrefoil hole configuration results in reduced potential for contaminant concentration at tube support plate intersections by reducing the crevice area in which contaminants can concentrate. Additionally, the licensee states that since the stainless steel 405 alloy tube support plates in the Millstone 3 model SGs are designed to inhibit the introduction of corrosion products, the support condition (i.e., denting) necessary for high cycle fatigue should not occur. The licensee also states without the lockup of tubes at the first (i.e., topmost) tube support plate that is caused by denting, the high cycle fatigue mechanism is unlikely unless severe circumferential cracking is present. The licensee concludes that since no cracking has been observed in the Millstone 3 Model F SGs the high cycle fatigue failure mode described in NRC BL 88-02 is not a concern.

3.3.3.5.3 Staff Evaluation and Conclusion

The NRC staff reviewed the licensee's evaluation of the effect of implementation of the MUR power uprate on SG tube integrity with respect to SG tube high cycle fatigue. In its LAR, the licensee states that Millstone 3 SGs contain Alloy 600 TT tubing and corrosion resistant stainless steel 405 alloy tube support plates with a four lobe hole quatrefoil holes. The licensee further states that because of the design differences between the SGs addressed in BL 88-02 and the SGs at Millstone 3, the criteria in BL 88-02 for high cycle fatigue concerns are not satisfied. The NRC staff finds the licensee’s evaluation acceptable because NRC BL 88-02 was applicable to SGs with carbon steel support plates, which Millstone 3 does not have. Therefore, the NRC staff has reasonable assurance that SG tube integrity will be maintained with respect to SG tube high cycle fatigue concerns.

3.3.3.6 Coatings Program

3.3.3.6.1 Regulatory Evaluation

Protective coating systems (paints) provide a means for protecting the surfaces of facilities and equipment from corrosion and contamination from radionuclides and 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 loss-of-coolant accident (DBA LOCA) conditions, considering temperature, pressure, radiation, and chemical effects on the emergency core cooling system .

Applicable regulatory requirements for protective coating systems are found in:

- 10 CFR Part 50, Appendix B, “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 structures, systems, and components.

- 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 62).

The licensee stated that the containment coatings program is described in FSAR Section 6.1.2.1, “Protective Coating.” This section states that American National Standards Institute (ANSI) standard N101.2, “Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities,” is the licensing basis for Service Level I coatings in the Millstone 3 containment.

3.3.3.6.2 Licensee Description

LAR, Section VII.6.D, “Coatings Program,” states, there were no changes to the containment analysis that would require a change to the containment design pressure or temperature. Since the containment design pressure and temperature limits were used to qualify the service level 1 containment coatings, and these limits are not changing, the service level 1 containment coatings remain qualified under MUR power uprate conditions.

3.3.3.6.3 Staff Evaluation and Conclusion

The NRC staff has reviewed the information provided by the licensee as well as the Millstone 3 FSAR with regards to the 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. The coating qualifications continue to bound the predicted conditions in containment after a DBA LOCA at the proposed MUR power uprate conditions. Therefore, the 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 staff has also determined that the protective coatings continue to meet the requirements of 10 CFR 50, Appendix B, as well as the testing requirements of ANSI N101.2-1972, “Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities” as described in Section 6.1.2.1 of the licensee’s FSAR (Reference 62).

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 Millstone 3 FSAR.

3.4.2 Technical Evaluation

The proposed MUR power uprate would allow Millstone 3 to operate at a 1.6 percent higher reactor thermal power level, increasing core power output from 3,650 MWt to 3,709 MWt. The electrical equipment design information is provided in Section V, "NRC Regulatory Issue Summary 2002-03 Topic: Electrical Equipment Design." of Attachment 4 to the LAR. The NRC staff reviewed the licensee's evaluation of the impact of the proposed MUR power uprate on the following electrical system attributes:

- AC Distribution System
- Power Block Equipment
- Direct Current (DC) System
- Emergency Diesel Generators (EDGs)
- Station Blackout
- Switchyard
- Grid Stability
- Environmental Qualification (EQ) Program

3.4.2.1 AC Distribution System

Section 8.3.1.1 of the FSAR states that the onsite AC distribution system consists of normal and Class 1E systems with the normal system providing power to station service transformers (SSTs), 6.9 and 4.16 kilovolts (kV) buses, and 480 Volt AC (VAC) load centers and motor control centers (MCCs). The 120 VAC instrument power system receives power from 480 VAC MCCs. The normal SSTs, which are usually supplied from the main generator (MG), feed power to the onsite AC distribution system, and if the MG trips with its output breaker open, back feed power to that same system from the 345 kV switchyard through the main transformer (MT). If the normal SSTs are not available, then the reserve SSTs supply power to the onsite AC distribution system. The Class 1E system consists of two 4.16 kV buses with each supplying power to the redundant safety related loads. Each Class 1E bus supplies four safety related 480 VAC load centers which supply the safety related MCCs with no cross connections between Class 1E systems at either 4.16 kV or 480 VAC levels.

Section V.1.E of Attachment 4 to the LAR indicates, the electrical load changes for the MUR power uprate affect only 6.9 kV and 4.16 kV, non-safety related motors. Based on licensee evaluations, their brake horsepower (BHP) load changes were bounded by the BHP load values used in the electrical distribution systems analyses of record (AOR). In addition, the licensee determined that the 6.9 kV and 4.16 kV systems loading, voltage, and short circuit current levels are still within their existing equipment capabilities after considering those motor load changes. Only negligible load changes will be made at the 480 VAC MCCs to support the MG upgrade with no electrical load changes at the 120 VAC level.

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 distribution system has adequate capacity to operate at MUR power uprate conditions since all its existing electrical equipment at the 6900, 4160, 480, and 120 VAC levels will continue to operate within their design ratings.

3.4.2.2 Power Block Equipment

The Millstone 3 turbine-generator converts the thermal energy of steam produced in the steam generators into mechanical shaft power and then into electrical energy. The MG is presently rated at 1354.7 mega volt-ampere (MVA), 24 kV, 60 Hertz (Hz), 0.925 power factor, 1800 revolutions per minute (rpm) at 75 pound per square inch gauge (psig) hydrogen pressure. In its letter dated June 2, 2021 (Reference 2), in response to a request for additional information (EEEB-RAI-9), the licensee identified that the increase in RTP due to the MUR power uprate results in an increase in the MG electrical power output of 21 megawatt electric (MWe). The licensee also clarified that the MG is nearing end of life expectancy with an extensive upgrade being made to it, instead of needed repairs to allow its continued safe operation. The upgraded MG will be rated for 1500 MVA at a power factor of 0.95 and will operate well below its upgraded capacity since Millstone 3 will be limited by its licensed RTP for the MUR power uprate. An upgrade will also be made to the isolated phase bus duct (IPBD) and any tap buses as indicated in Section V.1.F.ii of Attachment 4 to the LAR for the MUR power uprate.

Per Section V.1.F of LAR, the MT consists of three single-phase transformers and has a maximum design rating of 1500 MVA which can accommodate full power output capability of MG after its upgrade. The normal and reserve SSTs after considering downstream 6.9 and 4.16 kV motor load changes for MUR power uprate conditions will still be within their maximum design ratings.

Based on its review of the licensee's LAR, the NRC staff concludes that the MG and the IPBD based on their upgrades, and the MT and the normal and reserve SSTs based on their existing ratings, are adequately sized for the MUR power uprate and that they will operate well within their capacities.

3.4.2.3 DC System

The DC system, as described in FSAR Section 8.3.2, provides a reliable source of continuous power for control, instrumentation, and emergency lighting. The normal DC equipment consists of two 125 Vdc batteries, two operating battery chargers, one spare swing battery charger, two distribution switchboards, and five distribution panel boards. In Section V.1E of Attachment 4 to the LAR, the licensee identified that there will only be minor load changes to the DC system to support control power needs for the MG field breaker replacement.

Based on its review of the LAR and FSAR, the NRC staff concludes that since only minor load changes are being made to the DC system, it will continue to operate within its capabilities at MUR power uprate conditions.

3.4.2.4 Emergency Diesel Generators

The emergency diesel generator (EDG) system, as described in FSAR Section 8.3.1.1.3, consists of two independent 4.16 kV, 3-phase, 60 Hz, diesel engine-driven synchronous generators with each supplying one of the two Class 1E, 4.16 kV buses. In Section V.1.A, "Emergency Diesel Generators," of Attachment 4 to the LAR, the licensee evaluated each EDG's electrical loading at MUR power uprate conditions and determined that there are no load additions or modifications including no changes to its load sequences or durations, and its capacity and capability are bounded by its loading AOR.

Based on the review of the licensee's LAR and the EDG system design requirements at the MUR power uprate levels, the NRC staff concludes that the EDGs continue to meet the requirements of GDC 17 and have adequate capability to operate at MUR power uprate conditions.

3.4.2.5 Station Blackout

The licensee reviewed the potential impact of the MUR power uprate on the SBO licensing bases as described in LAR Section V.1.B.including associated plant equipment, Alternate AC (AAC) power source, SBO 8-hour coping duration and assessment, and auxiliary feedwater system flow rate requirements. Several factors determine the coping duration, but none of them were identified by the licensee as being affected by the MUR power uprate. In its LAR, section V.1.B, the licensee indicates there are no changes in the electrical loading requirements of associated systems/components including the AAC power source and its loading analysis for an SBO event at MUR power uprate conditions.

Based on its review of the licensee's LAR, the NRC staff finds that the proposed MUR power uprate will have no impact on the coping duration of any equipment required for an SBO event. Therefore, the NRC staff concludes that Millstone 3 will continue to meet the requirements of 10 CFR 50.63 upon implementation of the MUR power uprate.

3.4.2.6 Switchyard

In LAR Section V.1.G, "Switchyard Interface," the licensee commits to performing an Interconnection System Impact Study in accordance with Reference V.1, “Docket # ER19-2421-000, Effective Date 03/19/2020, Schedule 22, Large Generator Interconnection Procedures.” In its supplement dated June 2, 2021, for EEEB-RAI-10 (Reference 2), the licensee stated that Independent System Operator – New England (ISO-NE, the contractor) will evaluate switchyard functionality as part of that study. The licensee does not identify any switchyard load changes in the LAR or required modifications to switchyard components except for potential protection scheme modifications that may impact switchyard due to MG upgrade. The present output of MG is 1354.7 MVA with the MT presently rated for 1500 MVA. The MG will be upgraded to a 1500 MVA rating with only 21 MWe of that planned increase being due to MUR power uprate. That modest increase in power for the MT and switchyard is about 1.55 percent using present output of MG as base loading ((21MWe/1354.7MVA) x 100) which is a minimal change at the 345 kV level for both current and power requirements. Since the licensee did not identify any required modifications to switchyard components for that minimal change, the switchyard and its components will operate within their ratings.

Based on its review of the licensee's LAR and response to EEEB-RAI-10, the NRC staff concludes that there are no planned changes to switchyard components to address the negligible load changes noted above, and therefore the switchyard and its components will continue to operate well within their capability for MUR power uprate conditions.

3.4.2.7 Grid Stability

3.4.2.7.1 Technical Evaluation

In Section V.1.D, "Grid Stability," of Attachment 4 to the LAR, the licensee stated:

> An Interconnection System Impact Study will be performed in accordance with the processes of ISO-NE Schedule 22, Large Generator Interconnection Procedures. The Interconnection System Impact Study will consist of a short circuit analysis, a stability analysis, a power flow analysis (including thermal analysis and voltage analysis), a system protection analysis and any other analyses that are deemed necessary by the System Operator (ISO-NE) in consultation with the Interconnecting Transmission Owner (Eversource). The Interconnection System Impact Study will evaluate the impact of the proposed interconnection on the reliability and operation of the New England Transmission System.

In its response to EEEB-RAI-10, the licensee further described that ISO-NE will perform a system impact study which will include the following: a steady state analysis, stability analysis, a short circuit analysis, and a voltage and reactive power performance of the bulk power system.

Each of the analyses in that impact study are explained at a high level as to their intended purpose as follows:

(1) the steady state analysis will verify no significant adverse impact upon reliability or operating characteristics of bulk power system at MG maximum output during summer operation;

(2) the stability analysis will verify no significant adverse impact on stability, reliability, or operating characteristics of bulk power system under maximum MG output during winter operation;

(3) the short circuit analysis will verify that the fault ratings of electrical equipment in the switchyard and the grid in immediate vicinity of Millstone 3 are not exceeded and will identify any necessary upgrades to them for short circuit contributions from all generating facilities that are directly connected to New England Transmission System and interconnected through affected systems that have any impact on the Millstone 3 interconnection request; and

(4) the voltage and reactive power performance of the bulk power system will evaluate the compliance of the voltage control capability with the requirements of ISO-NE Operating Procedure No.14 - “Technical Requirements for Generators, and Demand Response Resources, Asset Related Demands and Alternative Technology Regulation Resources.”

ISO-NE will perform the analyses for that impact study as stated above with the sole purpose being to validate the capability of Millstone 3 and its surrounding grid for the MUR power uprate conditions based on conclusive results. Upon completion of the study, the licensee stated that it would confirm the effects of grid resiliency, switchyard functionality, and GDC-17 compliance. As stated by the licensee, if the impact study produces unsatisfactory results under the MUR uprated conditions, the MUR LAR will not be implemented, and the licensee will inform the NRC of this outcome and describe the expected actions for resolution.

3.4.2.7.2 Regulatory Commitment

The licensee submitted the following regulatory commitment in Attachment 4 to the LAR:

> Commitment: DENC will complete an Interconnection System Impact Study, including a grid stability analysis, as described in Attachment 4, Section V.I.D of the MPS3 MUR Power Uprate LAR submittal.
>
> Schedule Completion Date: The Interconnection System Impact Stability Study will be completed prior to implementation of the MUR Power Uprate for MPS3.

The NRC staff does not rely on regulatory commitments in its basis for the regulatory approval of the proposed change. Regulatory commitments specify the items for which the licensees volunteer to perform in support of their licensing applications. Regulatory commitments do not require prior NRC approval of subsequent changes and, therefore, they are not enforceable licensing requirements.

3.4.2.7.3 Technical Conclusion for Grid Stability

Based on its review of the licensee's LAR and the licensee's response to EEEB-RAI-10, the NRC staff has determined the licensee's approach for appropriately implementing the MUR power uprate based on the validated results of a completed system impact study as proposed in its LAR is acceptable regarding grid stability.

3.4.2.8 EQ Parameters

3.4.2.8.1 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)(2), which requires that humidity during design basis accidents must be considered.

- 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 non-safety-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.

3.4.2.8.2 Technical Evaluation

The NRC staff reviewed the Millstone 3 LAR to ensure that the EQ of electrical equipment important to safety remained bounded as a result of the proposed MUR power uprate. Section 8.3.1.1.4, "Design Criteria," of the Millstone 3 FSAR notes that the EQ test program for demonstrating the capability of Class 1E equipment to function throughout its qualified life is in accordance with the Institute of Electrical and Electronic Engineers (IEEE) Standard 323-1974, "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations," as endorsed by RG 1.89, "Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants," Revision 1 (Reference 63).

The licensee's response to Section II.1 of RIS 2002-03 is provided in Section II.1, "Accidents and Transients Bounded by the Proposed Uprated Power Level," and Table II.1-1, "FSAR Accidents, Transients, and Other Analyses," of Attachment 4 to the LAR. In the LAR, the licensee noted that the values of 3,723 MWt and 3,739 MWt corresponds to 102 percent of the current rated power of 3,650 MWt and 102 percent of an assumed nuclear steam supply system power of 3,666 MWt, respectively. These values remain the bounding power level for the MUR power uprate conditions (i.e., 3,709 MWt) when uncertainty is applied.

In Section II.1.31, "Analyses to Determine Environmental Qualification Parameters," and Section V.1.C, "Environmental Qualification (EQ) of Electrical Equipment," of Attachment 4 to the LAR, the licensee noted that it evaluated the effects MUR power uprate conditions for temperature, pressure, radiation, and humidity and related EQ parameters.

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. The NRC staff confirmed that the current design basis analyses of record were performed at 102 percent of either the current rated power of 3,650 MWt or an assumed NSSS power of 3,666 MWt, respectively, which bounds the MUR power uprate. Therefore, there is no impact on the EQ of electrical equipment with respect to temperature, pressure, radiation, and humidity and related EQ parameters due to the MUR power uprate.

3.4.2.8.3 Technical Conclusion for EQ Parameters

Based on its review of the LAR and the Millstone 3 FSAR, the NRC staff concludes that the MUR power uprate will not adversely impact the EQ of electrical equipment at Millstone 3 since the existing qualification for electrical equipment important to safety remains adequate and bound the MUR power uprate conditions.

Therefore, the NRC staff concludes that implementation of the proposed MUR power uprate will have no adverse impact on the Millstone 3 EQ Program or its ability to continue to meet 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 in the LAR and its supplement and determined that Millstone 3 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 based on its electrical engineering evaluations with the only credence given to the licensee's commitment to perform a grid stability study being that the MUR power uprate implementation will not commence if the study's results are unsatisfactory.

3.5 System Design

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 3,709 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. The NRC staff reviewed the impact of the MUR power uprate on the following major plant systems:

- NSSS interface systems,
- containment systems
- safety-related cooling water systems,
- SFP storage and cooling systems
- radioactive waste systems
- engineered safety features (ESF) heating, ventilation, and air conditioning
- flooding protection, and
- high and moderate energy line breaks.

The NRC staff's review focused on verifying that the licensee has provided reasonable assurance that plant systems will continue to operate safely at the MUR power uprate conditions. The NRC staff evaluated the LAR for conformance with the guidance provided in the SRP and in RIS 2002-03.

3.5.1 NSSS Interface Systems

The NSSS interface systems include the main steam, steam dump, condensate, feedwater, and auxiliary/emergency feedwater systems. The design parameters are established using conservative input assumptions to provide bounding conditions used in NSSS analyses. As described in Section 3.1 of Attachment 4 to the LAR, the design parameters include primary and secondary side system conditions (thermal power, temperatures, pressures, and flows) that are used as inputs to the NSSS analyses and evaluations. These parameters were revised to account for the increase in analyzed core power from 3,650 MWt to 3,723 MWt. Core power was conservatively increased by 2 percent in the analysis to bound the MUR power uprate value.

Main Steam (MS) System

The MS system is described in FSAR Section 10.3. The MS system is not only piping from steam generators to the main turbine, but also the main steam safety valves, the main steam isolation valves (MSIV) and the moisture separator reheaters.

The licensee stated that component parameters are bounded by the original design equipment ratings, or by the original design considerations for off-normal operation. It is stated that the MS system components have sufficient design and operational margin to accommodate the MUR power uprate.

For MS piping, the licensee evaluated pressures, temperatures, and velocities. The licensee determined that system pressures and temperatures are bounded by piping design parameters during MUR power uprate conditions. The flow velocities are bounded by industry design guidelines except for the main steam piping to turbine bypass valves, which exceed the industry recommended velocity limits by less than 5 percent. To address this velocity limit, it was noted that the MS piping to the turbine bypass valves is normally isolated and this section of piping is included in the FAC program. The FAC program is described in Section IV.1.E.iii of Attachment 4 to the LAR and is designed to monitor the rate of wear in piping and components. This program currently monitors parameters of the main steam, cold reheat, and hot reheat piping, which will continue after implementation of the MUR.

A total of five ASME B&PV code MSSVs are located on each main steam line outside reactor containment and upstream of the MSIVs. MSSV lift setpoints are determined by steam generator design pressure and the ASME B&PV code. Since the SG design pressure will not change with the MUR power uprate, the existing MSSV setpoints are unaffected. Main steam overpressure events have been analyzed at a power level of 3,723 MWt and the MSSVs are adequate for the MUR power uprate.

The MSIVs will close during accident conditions. The closure time of the MSIVs is not affected under MUR power uprate conditions, since the valve and operator designs are based on the worst-case break flow that the valve experiences.

Based on the discussion above, the NRC staff finds that the MS system remains capable of performing its function following the MUR power uprate.

Turbine Bypass System

The turbine bypass function is accomplished by the main steam pressure relieving valves, main steam pressure relieving bypass valves and turbine bypass to condenser system. The main steam pressure relieving valves and main steam pressure relieving bypass valves are described in Section 10.3 of the FSAR. The turbine bypass system is described in FSAR Section 10.4.

The MS system is provided with automatically actuated main steam pressure relieving valves and main steam bypass valves to control steam pressure at hot standby zero load operation and to remove reactor coolant system (RCS) stored energy following a turbine trip.

The turbine bypass to condenser system is designed to discharge a percentage of rated main steam flow directly to the main condenser, bypassing the turbine. The turbine bypass system creates an artificial load by dumping steam to the main condenser. Atmospheric dump valves (ADV) from each steam generator are provided to limit or control secondary pressure whenever the condenser is out of service.

The main steam pressure relieving valves and main steam pressure relieving bypass valves were evaluated and existing design supports cooldown capability at MUR power uprate operations. The licensee evaluated the turbine bypass to condenser and concluded that existing nine condenser dump valves have sufficient margin and continue to satisfy the requirements at MUR power uprate conditions.

Based on the discussion above, the NRC staff finds that the turbine bypass system remains capable of performing its function following the MUR power uprate.

Condensate and Main Feedwater Systems

The condensate and main feedwater systems, along with the condensate polisher demineralizers are described in FSAR Section 10.4.7 and 10.4.6.

The condensate system consists of three parallel, 50 percent capacity, condensate pumps. Normally, two condensate pumps are operating at full load, delivering water to the main feedwater pump's suction header. Three low pressure heater drain pumps (one per string) are normally operating at full load. The licensee determined MUR power uprate results in a condensate flow increase of approximately 1.6 percent. The licensee evaluated operating requirements and concluded the condensate system remains acceptable at MUR conditions.

The main feedwater system consists of three parallel, 50 percent nominal capacity, main feedwater pumps, two of which are variable speed turbine driven (TD) pumps and one is a constant speed motor driven (MD) pump. Two pumps are in operation at full load conditions. Main feedwater flow is controlled by feedwater regulating valves on each pump discharge.

As shown in FSAR Section 10.4.7.5, a speed controller automatically controls speed of the turbine driven feed pumps. Automatic/manual control selectors are located on the main control board. Inputs to the speed controller are steam flow, main steam header pressure, and feedwater pumps discharge header pressure. The LAR indicated that since a speed controller automatically controls the speed of the turbine driven feed pumps, the MUR power uprate requires an adjustment to the MSS/FWS differential pressure input setpoint. With these adjustments, the licensee indicated that the turbine driven feedwater pumps will provide the necessary flow in both configurations (two TD pumps or one TD pump/one MD pump) for MUR conditions. The licensee clarified that setpoint changes result in the feedwater regulating valve position remaining essentially unchanged from the pre-MUR position. Adequate main feedwater pump performance is available at MUR power uprate conditions and the low suction pressure alarm and trip setpoints will be adjusted for MUR flow conditions.

The LAR states that parameter changes resulting from the MUR power uprate do not exceed component design specifications or cause any adverse conditions that would challenge system operability. Main feedwater isolation valves, feedwater regulating valves, and feedwater regulating bypass valves will continue to provide containment isolation capability.

The licensee determined the MUR power uprate results in increased feedwater flow of approximately 1.8 percent. The existing NSSS accident analysis was completed at 102 percent of current licensed core power, which bounds the MUR power uprate.

The licensee evaluated operating requirements related to the increased flow and concluded MUR feedwater system capabilities remain sufficient to perform its function. The licensee indicated a comparison between operating requirements, including piping pressures, temperatures and flow increase, for MUR power uprate conditions and current conditions demonstrates that the condensate system capabilities remain acceptable at MUR power uprate conditions.

The licensee evaluated operating requirements related to the increased flow and concluded MUR feedwater system capabilities remain sufficient to perform its function. The condensate and main feedwater systems operating parameters will not significantly change at MUR power uprate conditions, therefore, the staff finds that the condensate and main feedwater systems remain capable of performing its functions following the MUR power uprate.

Auxiliary Feedwater System

The auxiliary feedwater (AFW) system is described in FSAR Section 10.4.9 and the system includes the demineralized water storage tank (DWST), which is a dedicated safety grade AFW pump suction source. The AFW system consists of two MD pumps, one TD pump and the associated piping and valves necessary to connect the DWST to the pump suctions, and the pump discharges to the feedwater system.

Three analyzed accidents crediting AFW were evaluated at 3,650 MWt for impact by the MUR power uprate as follows:

1. A Station Blackout (SBO) DWST required inventory and turbine driven auxiliary feedwater (TDAFW) pump capability verification analysis had been performed based upon a 3650 MWt licensed reactor power level. This analysis was revised using a 3,723 MWt (102 percent RTP) reactor power level and the results continue to show that the current TS 3.7.1.3, "Demineralized Water Storage Tank" limiting condition of operation is bounding relative to SBO event functional requirements, and that the TDAFW pump's capacity remains adequate relative to SBO event functional requirements.

2. FSAR Section 10.4.9 discusses the Best Estimate Loss of Normal Feedwater (BELONF) analysis, which supports the AFW system reliability analysis. This analysis was revised based on 3,723 MWt (102 percent RTP), along with DWST temperature change (based on operating experience), and concluded MUR Uprate has acceptable impact on analysis results.

3. An AFW system analysis determined the power level that can be supported by the motor driven AFW pumps during normal plant startup and shutdown evolutions and had been based upon decay heat and power history associated with a 3,650 MWt licensed reactor power level. This analysis was revised based on 3,709 MWt and concluded the motor driven AFW pumps capability during startup and shutdown is acceptable for MUR power uprate conditions.

Section V.1.B.v of Attachment 4 to the LAR indicates that analyses at MUR power uprate conditions show that the TDAFW pump will have sufficient capacity to support SBO scenario decay and sensible heat removal requirements following implementation of the MUR power uprate.

The limiting DWST required inventory was performed at 3,723 MWt (102 percent of 3,650 MWt) reactor power level and determined to be unaffected by MUR power uprate.

Based on the above, the staff finds that the AFW system remains capable of performing its function following the MUR power uprate.

NRC Staff Conclusion Regarding the NSSS System

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 determined that an 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 FSAR Section 6.2. The containment systems function to limit peak containment temperature and pressure during LOCA and main steam line break (MSLB) accidents. The containment systems also function to limit radiological release during various accidents.

Containment Isolation System

The containment structure is described in FSAR Section 6.2.1.1 and the containment isolation system is described in FSAR Section 6.2.4. The licensee indicated that their evaluation concluded that the MUR power uprate has no impact upon the containment structure and containment isolation system.

As the containment systems described in the LAR is not impacted by implementation of the MUR power uprate, the NRC staff determined that they are acceptable with respect to operation at the proposed MUR power uprate.

Appendix J Program

FSAR Section 6.2.6, "Containment Leakage Testing," defines the performance based testing program and test types. In Section VII.6.C "Appendix J Program," of its LAR the licensee stated:

> A review of the LOCA response analysis confirmed that the analysis was performed at 102 percent of 3650 MWt. Because the LOCA peak pressure analysis is unaffected, peak containment pressure (Pa) at MUR power uprate conditions is unchanged from the current conditions specified in MPS3 TS 6.8.4.f. No changes or modifications are required to the existing Appendix J Program or procedures.

The statement confirms that the current LOCA analysis was performed at 102 percent of 3,650 MWt which bounds the peak power at uprate conditions.

The NRC staff has reviewed the licensee's evaluation of the proposed MUR power uprate on containment systems and finds that the Appendix J Containment Leakage Rate Testing Program is unchanged with the proposed MUR power uprate, and is therefore, acceptable.

3.5.3 Safety-Related Cooling Water Systems

The safety-related cooling water systems include the reactor plant component cooling (CCP) system, service water (SWP) system, and Ultimate Heat Sink (UHS).

3.5.3.1 Reactor Plant Component Cooling System

The reactor plant CCP system is described in FSAR Section 9.2.2.1. The CCP system is a closed loop cooling system that transfers heat from reactor auxiliaries to the SWP system during plant operation and during normal and emergency shutdown/cooldown. The CCP system is designed to provide the cooling requirements for normal plant operation, plant cooldown, spent fuel pool cooling (SFC) and design basis accident (DBA) cooldown.

Section VI.1.C.i of the application states CCP system was evaluated to confirm that the heat removal capabilities are adequate to satisfy the MUR power uprate heat removal requirements during normal plant operation, plant cooldown, and accident cooldown conditions. It was determined that CCP is adequate to satisfy the MUR power uprate heat removal requirements during normal plant operation, plant cooldown, and accident cooldown conditions.

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

3.5.3.2 Service Water System

The SWP is described in FSAR Section 9.2.1. The SWP system provides cooling water for heat removal from the reactor plant auxiliary systems during all modes of operation and from the turbine plant auxiliary systems during normal operation.

The licensee states that each component cooled by the SWP system was evaluated to confirm that the existing flow rate is adequate to satisfy the MUR power uprate heat removal requirements during normal, shutdown, and accident conditions. The evaluations determined that the existing SWP flows will continue to support the heat removal requirements at MUR power uprate conditions. The licensee concludes that the system and component design parameters remain bounding for MUR power uprate operation.

The NRC staff has reviewed the information provided in the LAR regarding the SWP system and, based on the existing design parameters bounding uprate operating conditions, determined that SWP system will continue to perform its safety function upon implementation of the MUR power uprate.

3.5.3.3 Ultimate Heat Sink

The UHS is described in FSAR Section 9.2.5. The UHS is the source of cooling water provided to dissipate reactor decay heat and essential cooling system heat loads after a normal reactor shutdown or a shutdown following an accident. Heat removed from both safety and non-safety related cooling systems during normal operation, shutdown, and accident conditions is discharged via the SWP and Circulating Water System (CWS). Acceptable performance of the UHS is based on the ability to maintain an acceptable inventory of water to accept the design basis heat load at MUR conditions under limiting conditions. The UHS is Long Island Sound, which is effectively an infinite heat sink.

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

3.5.3.4 Staff Evaluation and Conclusion Regarding Cooling Water Systems

The NRC staff reviewed the information and evaluations performed by the licensee showing that the design of the cooling water 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 determined that an MUR power uprate will not challenge the cooling water systems, and concludes that the LAR is acceptable regarding cooling water systems.

3.5.4 SFP Storage and Cooling Systems

Spent fuel pool criticality analyses and fuel pool cooling and purification system impacts are discussed below.

3.5.4.1 Spent Fuel Pool Criticality

The licensee addressed the effects of the MUR power uprate on spent fuel pool criticality in Section VI.1.D.i of Attachment 4 to the LAR.

The licensee stated that the existing spent fuel pool criticality analysis was approved by NRC License Amendment No. 273 (Reference 64). The licensee also stated that the analysis underlying License Amendment No. 273 was performed considering a power level that bounds the MUR power uprate conditions.

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. Therefore, based on its review of the LAR, the NRC staff has determined that the spent fuel pool criticality analysis as approved in License Amendment No. 273 will not be impacted by implementation of the MUR power uprate.

3.5.4.2 SFP Cooling and Purification System

FSAR Section 9.1.3 discusses the cooling requirements for the SFP. The fuel pool cooling and purification system removes decay heat from spent fuel stored in the fuel pool and provides adequate clarification and purification of water in the fuel pool, refueling cavity, and refueling water storage tank. The fuel pool cooling system is safety related and the purification system is non-safety related.

The licensee states that the limiting heat load cases are performed at a power level of 3,723 MWt, which bound the MUR power level. There is no change to the loss of cooling analysis. The MUR power uprate will have no significant impact on the SFC refueling purification or cooling functions. Section II.1 (Item 33) of Attachment 4 to the LAR indicates the spent fuel evaluation has been performed for normal and abnormal conditions to address capabilities of the spent fuel pool cooling system and concluded that loss of SFP cooling analysis is bounding for the MUR power uprate.

As indicated above, the licensee comparison of the operating conditions for the MUR power uprate to the current 3,723 MWt conditions demonstrates that the SFP fuel pool cooling and purification system have sufficient design and operational margin to accommodate the MUR and the systems remain bounded by the existing analyses. Therefore, based on its review of the LAR, the NRC staff has determined that the spent fuel system will not be impacted by implementation of the MUR power uprate.

3.5.5 Radioactive waste systems

The NRC staff reviewed the radioactive waste systems which include the gaseous waste system, liquid waste system, and solid waste system as described in Chapter 11 of the FSAR. Additionally, the NRC staff reviewed the steam generator blowdown system as described in FSAR section 10.4.8. These systems provide the means to control, collect, process, store, recycle, release, and dispose of radioactive waste generated during normal operation. The licensee stated these systems and the various subsystems and components were evaluated for the MUR power uprate.

3.5.5.1 Gaseous Waste System

The gaseous waste processing system is designed to remove fission product gases from the reactor coolant. The major input to the gaseous waste processing system during normal operation is taken from the gas space in the volume control tank.

At MUR power uprate conditions, the required containment, confinement, and filtering capacities of the gaseous waste processing system and the capacities of its various decay and storage tanks are sufficient because the MUR power uprate does not materially affect the system flow rates or gas volumes. The MUR power uprate may increase radioactivity of gaseous waste a maximum of 1.6 percent. However, as discussed by the licensee, no system or component design parameters were exceeded at MUR power uprate conditions. The increase in nuclide concentration does not significantly impact gaseous waste processing system operation, and is therefore, acceptable.

3.5.5.2 Liquid Waste System

The liquid waste processing system is designed to control, collect, process, store, recycle, and dispose of liquid wastes. The system design considers potential personnel exposure and ensures that quantities of radioactive releases to the environment are as low as is reasonably achievable.

The liquid waste processing system collects and processes potentially radioactive wastes for recycle or for discharge. Provisions are made to sample and analyze fluids before they are recycled or discharged. Based on the laboratory analysis, these wastes are either released under controlled conditions via the cooling water system or retained for further processing. Detailed administrative records of all radioactive liquid releases are maintained.

As discussed by the licensee, the liquid waste system functions and the liquid waste volume processed are unaffected by the MUR power uprate. The volume generated during normal operation will not change because of the uprate. Implementing the MUR power uprate will not increase the volume inventory of liquid waste processed by the liquid waste processing system. The concentration of radioactive nuclides in the liquid waste processing system is expected to increase by a maximum of approximately 1.6 percent. This increase in nuclide concentration does not significantly impact liquid waste processing system operation and is therefore acceptable.

3.5.5.3 Solid Waste System

The solid waste system is sized to treat the radioactive solid waste produced during plant operation. The MUR power uprate does not affect the generation of solid waste volumes or the function of the solid waste management system. The MUR power uprate may increase radioactivity of gaseous waste a maximum of 1.6 percent. The potential increase in nuclide concentration does not significantly impact solid waste processing system operation, and is therefore, acceptable.

3.5.5.4 Steam Generator Blowdown System

Similar to the waste management systems, the steam generator blowdown system will continue to operate at the same flowrate but may have an increase in impurities (including a small increase in radioactivity). The licensee indicates that impurities will be evaluated under the plant chemistry program. In addition, like the radwaste systems, the increase in radionuclide concentrations in the steam generator blowdown system would have minor impact on the steam generator blowdown system radionuclide inventory. Since the small increase in radioactivity would not be expected to significantly impact solid waste processing system operation the change is acceptable.

3.5.5.5 Staff Evaluation and Conclusion Regarding Radioactive Waste Systems

The licensee evaluated the waste systems and subsystems and components for the MUR power uprate. No system or component design parameters were exceeded at MUR power uprate conditions. The systems are sized to treat the waste produced during plant operation, and the licensee confirmed that the functions of the waste system and the waste volume processed are unaffected by the uprate. Based on the existing design parameters bounding uprate operating conditions, NRC staff has determined that the waste systems will not be impacted by implementation of the MUR power uprate.

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

The NRC staff reviewed the impact of implementation of the MUR power uprate on the control room ventilation system, the engineering safety features building ventilation system, fuel building ventilation system, and the containment ventilation system. In LAR Section VI.1.F "Engineered Safety Features (ESF) Heating, Ventilation, and Air Conditioning Systems," the licensee described that these systems were evaluated at the expected MUR power uprate conditions and stated that these systems remain capable of performing their intended functions, with no impact on ventilation operation.

In its review, the NRC staff also used specific criteria relevant to the evaluation of ESF HVAC Systems found in the SRP, Section 6.4, "Control Room Habitability System"; Section 6.5.2, "Containment Spray as a Fission Product Cleanup System"; Section 9.4.1, "Control Room Area Ventilation System"; Section 9.4.2, "Spent Fuel Pool Area Ventilation System"; Section 9.4.3, "Auxiliary and Radwaste Area Ventilation System"; Section 9.4.4, "Turbine Area Ventilation System"; and Section 9.4.5, "Engineered Safety Feature Ventilation System."
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.

3.5.6.1 Control Building Ventilation System

The Control Building Ventilation System (LAR Section VI.1.F.i) is described in FSAR Section 9.4.1. The control building ventilation system consists of air conditioning, heating, filtration, and ventilation subsystems which provide a suitable environment for the comfort and safety of personnel within the control room area; and facilitates removal of equipment generated heat except for the chiller and cable spreading areas. The cable spreading area has no forced ventilation and is purged by portable fans. The analysis determined that the control building electrical area ventilation systems (switchgear rooms, battery rooms, cable spreading areas, chiller equipment room) are not impacted by the MUR power uprate conditions, because the heat loads in these areas do not increase. The heat loads (electrical heat loads, lighting, personnel) at MUR power uprate conditions were evaluated.

Loss of Ventilation analyses was performed (LAR Section V.1.B.iv) for the areas outside containment containing SBO equipment (control room, instrument rack room, switchgear rooms) to determine the expected area temperatures during an eight-hour SBO event at current plant conditions. The analysis determined the heat loads in the control room, instrument rack room, and east and west switchgear rooms are not impacted by the MUR power uprate.

3.5.6.2 Engineered Safety Features Building Ventilation System

The ESF Building ventilation system (LAR Section VI.1.F.i) is described in FSAR Section 9.4.5. The ESF Building ventilation system consists of two systems: one normal and one emergency ventilation system. The current limiting case was evaluated at the MUR power uprate conditions and found to have no impact on the ESF normal and emergency ventilation capabilities to perform their function. This was based on having no increase in electrical heat loads and no significant increase in piping system heat loads at the MUR power uprate conditions.

3.5.6.3 Fuel Building Ventilation System

The fuel handling area ventilation system (LAR Section VI.1.F.iii) is described in FSAR Section 9.4.2. The fuel building ventilation system removes heat generated by equipment and water vapor from fuel pool evaporation, prevents moisture condensation on interior walls, and provides a suitable environment for equipment operation and personnel. This system is not credited in the fuel handling accident analysis. The licensee stated that the SFP cooling equipment loads analyses are not impacted by the MUR power uprate. The decay heat is not significantly increased at MUR power uprate conditions and will not impact the limiting case full core off-load temperature limit. The licensee also indicated that the maximum SFP and piping temperatures at MUR power uprate conditions are below the calculated limiting case.

3.5.6.4 Containment Ventilation

The containment structure ventilation systems (LAR Section VI.1.H) are described in Section 9.4.7 of FSAR. The containment ventilation system provides general area cooling and direct cooling to critical components. It also provides the means to purge the containment atmosphere prior to personnel entry during maintenance periods. The containment air cooling system provides air cooling that maintains containment bulk air temperature within the limits specified in TS 3.6.1.5. As indicated in LAR, the containment bulk temperature increase is negligible, and the current design remains acceptable.

It was determined that the containment bulk temperature increase is negligible, and the current design remains acceptable. The MUR power uprate at Millstone 3 will not have any adverse effects on the containment structure ventilation system operation.

3.5.6.5 Staff Evaluation and Conclusion Regarding Ventilation Systems

Based on the discussion above, the NRC staff has determined that the increase in heat loads in the ESF HVAC systems are minimal and bounded by the current analyses and the criteria identified in NRC requirements remain satisfied at MUR power uprate conditions. The NRC staff has also determined the applicable guidance in the SRP for evaluating the increase in heat loads in the control room and on the ventilation systems has been adequately addressed. Therefore, the NRC staff finds the proposed MUR uprate amendment request acceptable regarding ESF HVAC.

3.5.7 Flood Protection

In Section II.1 (Item 34) of Attachment 4 to the LAR, the licensee provided a brief discussion on Internal flooding areas related to the following major flooding concerns:

1. Internal flooding from tanks and vessels outside containment, which could potentially affect safety-related components;
2. Liquids currently entering the equipment and floor drains systems; and
3. CWS flow rate and operating pressures.

The licensee indicated that no increase in the tank size or the amount of fluid in any of the non-Seismic Category I tanks and vessels located in safety-related structures outside the containment, require the addition of any new non-Seismic Category I tanks or vessels, affect the location of existing safety-related equipment required for safe shutdown of the plant, nor does it require the addition of any new safety-related equipment required for safe shutdown.
The MUR power uprate does not add any new equipment or modify existing equipment (e.g., pumps, strainers) that would result in increasing the quantities of liquids currently entering the equipment and floor drains systems.

CWS flow rate and operating pressures are unchanged at MUR power uprate conditions. There are no required modifications to the CWS or the Turbine Building as a result of the MUR power increase that would affect the analyses associated with flooding due to a circulating water pipe rupture or expansion joint failure. The MUR power uprate does not add any safety-related equipment to the Turbine Building.

Therefore, the MUR power uprate does not affect the analyses and design features related to internal flooding due to a circulating water pipe rupture or expansion joint failure.

The NRC staff has reviewed the information provided in the LAR regarding internal flooding. Based on current plant design not being changed in any way to increase flooding concerns from major component failures, the NRC staff finds that the existing flood protection remains adequate for MUR power uprate conditions.

3.5.8 High and Moderate Energy Line Breaks

The impact and consequences of high energy line breaks (HELBs) and moderate energy piping system cracks are postulated to occur outside containment and are discussed in FSAR Section 3.6.1 and LAR Section VII.6.B. The high and moderate energy break program ensures that systems or components required for safe shutdown or important to safety are not susceptible to the consequences of high and/or moderate energy pipe breaks. FSAR Section 3.6, "Protection Against Dynamic Effects Associated with the Postulated Ruptures of Piping," describes the high and moderate energy line break analysis.

Section IV.1.B.vii of Attachment 4 to the LAR, "High Energy Line Break (HELB) Locations" indicates a review was performed to determine the MUR power uprate impact on HELB systems. MUR power uprate operating temperatures, pressures, and mass flow rates were compared to the analyzed conditions. The licensee's review concluded that the total pipe stresses are not significantly impacted, do not result in any new or revised pipe break locations, and the existing design basis for pipe break, jet impingement, and pipe whip remains valid.

The licensee indicated that the MUR power uprate does not result in any new or revised high or moderate energy line break locations and the high and moderate energy line break analysis is not affected. The licensee determined that area temperatures and pressures resulting from high and moderate energy line breaks, and internal flooding conditions resulting from high or moderate energy line breaks, remain valid at MUR power uprate conditions.

Based on current HELB evaluation and lack of new or revised pipe break locations, and the existing design basis for pipe break, jet impingement, and pipe whip remaining valid, the NRC staff concludes that the HELB analysis remains acceptable.

3.5.9 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 Other Considerations

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:

- The regulation at 10 CFR 50.34(f)(2)(i), in part, requires the licensee to provide simulator capability that correctly models the control room.

- The regulation at 10 CFR 50.34(f)(2)(ii), requires the applicant to establish a program, to begin during construction and follow into operation, for integrating and expanding current efforts to improve plant procedures. The scope of the program shall include, applicable to the Millstone 3 MUR power uprate, emergency procedures, human factors engineering, and operator training.

- The regulation at 10 CFR 50.34(f)(2)(iii), requires the applicant to provide, for Commission review, a control room design that reflects state-of-the-art human factor principles prior to committing to the fabrication or revision of fabricated control room panels and layouts.

- The regulation at 10 CFR 50.34(f)(2)(iv), requires a plant safety parameter display system (SPDS) console that displays to operators a minimum set of parameters defining the safety status of the plant, capable of displaying a full range of important plant parameters and data trends on demand, and capable of indicating when process limits are being approached and exceeded.

- RIS 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," provides the NRC staff guidance for reviewing MUR power uprate applications. Specifically, Section VII, "Other," of RIS 2002-03, has a set of statements applicable to a human factors review of MUR power uprate applications.

3.6.1.2 Technical Evaluation

3.6.1.2.1 Operator Actions

In Section VII.1 of Attachment 4 to the LAR, the licensee provided the following:

> Reviews have determined that the existing required operator actions are not affected by the MUR power uprate. There are no reductions in the times for required operator actions. No new manual operator actions must be created and no existing manual actions must be automated or modified.
>
> The MUR power uprate is being implemented under the administrative controls of the plant modification process. The critical operator actions and the associated completion times currently tracked in the MPS3 Time Critical Operator Action program remain valid and unchanged for MUR conditions. The plant modification process ensures that impacted procedures will be revised prior to the MUR power uprate implementation.

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. The NRC staff finds that the operator actions affected by the Millstone 3 MUR power uprate are unchanged by this LAR (i.e., no creation of new actions, deletion of existing actions, or automation of actions). The time available for the existing required operator actions is also not changed.

3.6.1.2.2 Emergency and Abnormal Operating Procedures

In Section VII.2.A of Attachment 4 to the LAR, the licensee provided the following:

> Emergency and abnormal operating procedures were reviewed to determine any MUR power uprate impact. No changes are required to the procedure steps and mitigation actions as a result of the MUR power uprate.
>
> Several setpoints used in the plant emergency operating procedures were based on a core rated thermal power of 3650 MWt. The use of core rated thermal power in developing the setpoints was consistent with the Westinghouse Owners Group background documentation. For the implementation of the MUR power uprate, the setpoints will be changed to reflect a total core power of 3723 MWt, which is 102 percent of 3650 MWt, in order to bound the MUR power uprate.
>
> There are no operator action changes for shutdown risk management due to the MUR power uprate. The time to core boil will decrease due to the MUR power uprate but the method of calculating the time to core boil will remain the same. Therefore, DENC has determined that no emergency and abnormal operating procedure modifications due to MUR power uprate are required to ensure that

> operator actions do not adversely affect defense in depth or safety margins. MPS3 procedures will be revised to include data generated for decay heat at the MUR power level. Operator training on the procedure changes will be provided as part of the MUR power uprate implementation.

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 that the emergency and abnormal operating procedures are acceptable because there are no changes to the associated procedures due to the MUR power uprate, other than updating the setpoint to reflect the new RTP. The NRC staff also finds that the proposed MUR power uprate for Millstone 3 will be in conformance with 10 CFR 50.34(f)(2)(ii) and the proposed MUR power uprate 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 Attachment 4 to the LAR, the licensee provided the following:

> The recommended modifications associated with the MUR power uprate that pertain to control room controls, displays (including the safety parameter display system), and alarms have been identified. These modifications include rescaling of the instruments associated with turbine first stage pressure, instrument loops affected by the MUR power uprate (indicator replacement, calibration span, and/or scaling), and changes to calorimetric algorithm and plant computer points. Therefore, DENC has identified the modifications to control room controls, displays and alarms due to MUR power uprate that are necessary to ensure that changes in operator actions do not adversely affect defense in depth or safety margins. Critical safety function status trees will be reviewed and revised as necessary.

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 licensee has appropriately identified the human system interfaces related to plant safety affected by the Millstone 3 MUR power uprate and will modify them so that these changes do not adversely affect defense in depth or safety margins. The NRC staff also finds the proposed MUR power uprate for Millstone 3 will be in conformance with 10 CFR 50.34(f)(2)(iii) and 10 CFR 50.34(f)(2)(iv).

3.6.1.2.4 Control Room Plant Reference Simulator

In Section VII.2.C of Attachment 4 to the LAR, the licensee provided the following:

> The MUR power uprate is being implemented under the plant modification process administrative controls. Modifications due to MUR power uprate are necessary to ensure that changes in operator actions do not adversely affect defense in depth or safety margins have been identified. The modifications will be performed under the administrative controls of Millstone Power Station. Any required simulator changes resulting from the MUR power uprate will be evaluated, implemented and tested per MPS3 approved procedures. Any required simulator modifications will be completed in time to support operator training prior to MUR power uprate 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 proposed changes to the control room plant reference simulator will not adversely affect defense-in-depth or safety margins. The licensee's proposed changes to the simulator will be completed under the plant modification process administrative controls and are acceptable. The NRC staff also finds the proposed MUR update for Millstone 3 will be in conformance with 10 CFR 50.34(f)(2)(i).

3.6.1.2.5 Operator Training

In Section VII.2.D of Attachment 4 to the LAR, the licensee provided the following:

> The operator training program requires revision as a result of the MUR power uprate. Operator training will be developed, and the operations staff will be trained on the plant modifications, TS and TRM changes, and procedure changes prior to MUR power uprate 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 licensee has appropriately identified the elements of operator training that must be revised and will implement these changes under the administrative controls of the plant modification process so that it reflects the MUR power uprate for Millstone 3 and will not adversely affect defense in depth or safety margins. The NRC staff also finds the proposed MUR power uprate for Millstone 3 will be in conformance with 10 CFR 50.34(f)(2)(ii).

3.6.1.2.6 Modifications

In Section VII.3 of Attachment 4 to the LAR, the licensee provided the following:

> Modifications required to support the MUR with respect to various aspects of plant operator actions, including operator training, will be completed prior to MUR power uprate implementation for MPS3.

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 licensee's treatment of the proposed modifications (i.e., changes to the emergency and abnormal operating procedures; changes to the control room control, displays, and alarms; changes to the simulator; and training of operators) to support the Millstone 3 MUR power uprate is acceptable because the changes will be implemented under the administrative controls of the plant modification process. The process also ensures that all changes will be completed prior to implementation so that any potential issues may be identified and addressed.

3.6.1.2.7 Temporary Operation Above Licensed Full Power Level

In Section VII.4 of Attachment 4 to the LAR, the licensee provided the following:

> Existing MPS3 plant operating procedure related to temporary operation above full steady-state licensed power levels will be revised to reflect the uncertainty in the power level credited in the MUR power uprate application. Precautions will be revised to account for the MUR uprate power level.

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 reflect the uncertainty in the power level credited in the MUR power uprate in addition to revising the precautions. The changes to the procedures are necessary for the operators to take the correct action so that the defense in depth or safety margins of Millstone 3 are not compromised.

3.6.1.3 Conclusion

The NRC staff concludes that the proposed MUR power uprate for Millstone 3 is acceptable because the licensee has appropriately identified and addressed the human factors aspects (i.e., operator actions, procedures, training, human system interfaces, and the simulator) that will be affected by the MUR power uprate. The licensee has stated that those changes will be made prior to implementation so that potential issues may be identified and addressed, and that those changes do not adversely affect defense in depth or safety margins. The NRC staff has determined that the results of the licensee's review of these areas would continue to meet the applicable requirements in 10 CFR 50.34(f)(2)(i)-(iv) and are in conformance with Section VII, Items 1 through 4, of Attachment 1 to RIS 2002-03.

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; (4) GDC-61, insofar as it requires that systems that may contain radioactivity be designed to assure adequate safety under normal and postulated accident conditions, including suitable shielding, containment, confinement, and filtering systems; and (5) 10 CFR 50.34(f)(2)(vii) and Item II.B.2 of NUREG 0737, "Clarification of TMI Action Plan Requirements," (Reference 39), insofar as it relates to plant shielding for spaces/systems which may be used in post-accident operations and to protect safety equipment from the radiation environment. Specific review criteria are contained in SRP Section 11.1 (Reference 65) and SRP Sections 12.1-12.5 (Reference 66).

The staff conducted its evaluation to verify that annual dose are within the applicable 10 CFR 20 annual limit of 100 mrem, and the 40 CFR 190 annual limit of 25 mrem to a member of the public from the reactor fuel cycle, as referenced by 10 CFR 20.1301(e). In addition, the staff conducted a review to ensure that occupational radiation exposure would continue to meet NRC requirements and the "as low as is reasonably achievable" (ALARA) criterion.

3.6.2.1.2 Design-Basis Accident

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[6] 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 excess of 25 rem [Roentgen equivalent man][7] 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.

Regulations in 10 CFR 50.67, “Accident source term,” 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)[8] total effective dose equivalent (TEDE).

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[6] 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.

[7] 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.

[8] 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

> (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 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:

RG 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 67).

RG 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

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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.

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 68).

RG 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 35). 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. Specific criteria for radiological consequence analyses using alternative source terms are contained in SRP Section 15.0.1 (Reference 69).

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.

NRC License Amendment No. 232, dated September 15, 2006 for Millstone 3 (Reference 36) approved the implementation of the Alternative Source Term (full scope for Millstone 3) radiological analysis methodology in accordance with 10 CFR Section 50.67 to perform the radiological consequences analyses of design basis accidents as described in RG 1.183 (Reference 35).

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; and (3) concentrations of all radionuclides other than fission products in the reactor coolant.

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 is activated to become radioactive nitrogen-16 by a neutron-proton reaction as it passes through the neutron-rich core at power. 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.

As discussed by the licensee, during normal operations, the primary coolant specific activity is expected to increase by no more than the percentage increase in power level. Therefore, there will be no significant change in the types or amounts of any effluents released offsite during normal operation. In addition, offsite release concentrations and doses will continue to be within allowable 10 CFR 20 and 10 CFR 50, Appendix I, limits per the MPS REMODCM. In addition, actual, measured doses due to normal effluent associated with the reactor operating at the currently licensed thermal power 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 ALARA. The MUR power uprate will not result in changes to the offsite dose calculation manual where an expected slight increase in long-lived effluent isotopic releases and doses would increase approximately proportional to the MUR power uprate.

Therefore, the NRC staff concludes that the proposed license amendment is acceptable with respect to the radiological effluents from radwaste systems.

3.6.2.2.1.2 Individual or Cumulative Occupational Radiation Exposure

Millstone 3 was designed with sufficient margin for higher-than-expected radiation sources. Specifically, FSAR, Chapter 12, indicates that while the historical plant shielding design was based on a power level of 3,636 MWt (which is slightly lower than the power uprate level of 3,709 MWt), the initial plant shielding design for Millstone 3 was based on an assumed 1 percent fuel defects. This exceeds the current guidance in NUREG-0800 of an assumed 0.25 percent fuel defects. This conservatism of an assumed failed fuel percentage of four times the current guidance far exceeds the slightly lower assumed power level.

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 and ALARA as 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. A review of plant occupational exposure history shows that Millstone didn't have any exposures near the annual occupational dose limits and collective dose was consistent with other reactors. A small increase in power will not have a significant impact in occupational exposure or exposure to members of public and the licensee's programs can continue to be used to meet NRC requirements. Therefore, no change is required in the design basis radiation protection design features for the MUR uprate.

As a result, 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 efficiency of the NRC staff's review, licensees requesting an MUR power uprate should first identify existing DBA analysis of record, which 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. The Millstone 3 accident and safety analyses including loss of coolant accident events are addressed in Chapter 15 of the Millstone 3 FSAR.

In its LAR, the licensee discussed each analysis in support of the MUR, including the assumed core power level in each analysis and whether the analysis remains bounding for the MUR power uprate.[9] As previously discussed, the MUR LAR would increase the Millstone 3 authorized core power level from 3,650 MWt to 3,709 MWt which is an increase of approximately 1.6 percent rated thermal power, based on the use of the LEFM CheckPlus system. In accordance with current licensing basis which incorporates the full implementation of AST, and as documented in FSAR Chapter 15, the dose consequences of environmental releases following a LOCA meet the onsite and offsite dose limits set by 10 CFR 50.67, as modified by RG 1.183, Revision 0 (Reference 35). The inventory of radionuclides in the reactor core available for release into containment following a LOCA is currently based on a core thermal power of 3,723 MWt (102 percent of the current rated thermal power of 3,650 MWt), which bounds operation at MUR operating conditions.

The NRC staff reviewed the impact of the proposed 1.6 percent MUR power uprate on DBA radiological consequence analyses, as documented in Chapter 15 of the FSAR. The NRC staff confirmed that the current licensing basis dose consequence analyses remain bounding at the proposed MUR uprated power level of 3,723 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 areas of review include:

- Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accidents) - FSAR Section 15.6.5.4
- Fuel Handling Accident-FSAR Section 15.7.4
- Major Secondary System Pipe Rupture - FSAR Section 15.1.5.4
- Steam Generator Tube Rupture - FSAR Section 15.6.3.2.2
- Failure of Small Lines Carrying Primary Coolant Outside Containment – Section 15.6.2
- RCCA Ejection Accidents – FSAR Section15.4.8.4
- Reactor Coolant Pump Locked Rotor- FSAR Section 15.3.3.4
- Radioactive Gaseous Waste System Failure - FSAR Sections 15.7.1/11.3.3
- Radioactive Liquid Waste System Leak or Failure – FSAR Sections 15.7.2/11.2.3.1
- Vital Area Doses and Shielding – FSAR Section 12.3

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 has continuous operating online self-diagnostic processes to verify that the digital circuits are operating correctly and within the design basis uncertainty limits. These processes can identify failure conditions that will cause the LEFM CheckPlus to switch from the normal operation to the maintenance mode.[10]

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[9] Also included in Table II-1 of the LAR, “FSAR Accident, Transients and Other Analyses”
[10] See Section 3.1.2 of this SE for a description of the LEFM CheckPlus system in normal and maintenance modes of operation.

The NRC staff confirmed that the applicable current licensing basis dose consequence analyses remain bounding at the proposed MUR uprated power level of 3,709 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. 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 Millstone 3 FSAR, 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, when the UFM System is in "CheckPlus" mode, considering the higher accuracy of the proposed feed water 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 and of the postulated DBA radiological dose consequence analyses at the proposed uprated power level. The NRC staff finds that operating Millstone 3 at the proposed uprated power level will continue to meet the applicable dose limits following implementation of the proposed 1.6 percent MUR power uprate. The NRC staff further finds there is reasonable assurance that Millstone 3, as modified by this approved license amendment, 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. In addition, the staff finds that the plant can continue to meet radiation protection requirements, shielding requirements, and effluent release requirements and including maintaining doses ALARA during normal operation. Therefore, the NRC staff concludes that the proposed license amendment is acceptable with respect to the impact on plant source terms, radiation doses, effluent releases, and the radiological dose consequences of DBAs.

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, nor will it 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 MUR power uprate on the plant's safe-shutdown analysis to ensure that structures, systems, and components (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) Criterion 3, "Fire protection;" and (3) Criterion 5, "Sharing of structures, systems, and components;" of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50. 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.

Criterion 3, of 10 CFR Part 50, Appendix states:

> 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.

Criterion 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 Criterion 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 core power level of 3,709 MWt against the previously analyzed core power level of 3,650 MWt.

The NRC staff reviewed Attachment 4 to the LAR, Sections II and VII, 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 Attachment 4 to the LAR, Sections II, Item 32, "Safe Shutdown Fire Analysis (Appendix R)," the licensee stated that the proposed MUR power uprate does not modify the current alternative shutdown methods used for safe shutdown and does not require any new methods or equipment to perform the required alternative shutdown functions. Further, the licensee stated that the analyses demonstrate that Millstone 3 can be placed in cold shutdown status within 72 hours after reactor trip for the limiting fire scenarios at MUR power uprate conditions. The MUR power uprate does not affect the systems and components used to achieve hot standby and cold shutdown in the event of a fire (e.g., auxiliary feedwater and residual heat removal), and does not affect the relevant support systems (e.g., diesel generators, ventilation systems). The licensee indicated that the operator actions and time limits for operator actions associated with the limiting transients are not changed by the MUR power uprate. The MUR power uprate does not add any heat loads that would affect the current evaluations of ventilation system requirements for post-fire safe shutdown. Accordingly, area temperature considerations should not impact the ability of operators to perform required manual actions at MUR power uprate conditions.

In Attachment 4 to the LAR, Sections VII.6.A, "Fire Protection Program," the licensee stated that an evaluation concluded that the MUR power uprate has no impact upon the fire protection program's effectiveness in: (1) preventing fires from starting; (2) rapid fire detection and suppression; and (3) obtaining safe shutdown as defined in the Millstone 3 fire protection licensing basis. Further, the fire protection organization, including fire brigade is unaffected by the MUR power uprate. There are no changes to the responsibilities, reporting relationships, or fire brigade composition as a result of the MUR power uprate. The licensee stated that the fire protection subsystems, including fire detection and alarm (including fire detectors, manual pull station, and alarm notification alarms), site main fire water system (including fire water main loop, subsystem distribution, fire hydrants, fire pumps, sprinkler systems and deluge systems), carbon dioxide and halon suppression system, portable suppression capabilities (consisting of portable fire extinguishing equipment and manually activated subsystems), life safety egress (including protected exit access, exits and emergency lighting), and communication systems remain unchanged as a result of the MUR power uprate.

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 Millstone 3 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 Millstone 3 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 also reviewed the LAR information concerning the elements of the fire protection program effectiveness. The licensee stated that the proposed MUR power uprate will not impact the fire protection organization, including fire brigade. Further, fire protection subsystems, including fire detection and alarm, site main fire water system, water-based and gaseous fire suppression systems, portable suppression capabilities, life safety egress (including protected exit access, exits, and emergency lighting), and communication systems remain unchanged as a result of the MUR power uprate. The NRC staff finds the licensee information acceptable because the licensee's analysis concluded that the elements of the fire protection systems would not be impacted by 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 1.6 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.

3.7 Changes to the Operating License and Technical Specifications Associated with the MUR Power Uprate

As stated above, the licensee requested changes to the Millstone 3 TSs to allow for an MUR power uprate. The licensee proposed an MUR increase in RTP level from 3,650 MWt to 3,709 MWt, an increase in RTP of approximately 1.6 percent.

The proposed changes would revise the “Maximum Power Level” in the Facility Operating Licenses 2.C.(1) "Maximum Power Level" and TS 1.27 “Rated Thermal Power."

The proposed change would also revise TS 3.7.1.1, Action Statement “a” and TS Table 3.7-1 to update the maximum power levels corresponding to the number of operable MSSVs per SG. In addition, the proposed changes make an editorial correction to TS 2.1.1.1 from “correlations” to “correlation.”

As stated in the LAR, a mark-up of the TS Bases and Technical Review Manual (TRM) pages were provided for information only and will be updated in accordance with the TS Bases Control Program and TRM Control Program, respectively after NRC approval of the proposed license amendment.

3.7.1 Regulatory Evaluation

The NRC staff's guidance for reviewing changes to the TS is contained in 10 CFR 50.36, “Technical Specifications,” where 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: 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 safety evaluation, the licensee proposed changes to the operating license (OL) and TS as discussed below.

The Millstone 3 OL Item 2.C is being revised to increase the maximum core power level from 3,650 MWt to 3,709 MWt.

The definition of RTP in TS 1.27, “Definitions – Rated Thermal Power,” is being changed to account for the increase in reactor core thermal power level as follows:

> RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3709 MWt.

An administrative correction is proposed for Section 2.1, “Safety Limits,” in TS 2.1.1.1, to change the reference to a correlation used for calculating the departure from nucleate boiling ratio from plural to singular because there is only one correlation used for this calculation. The NRC staff finds the proposed revision from ‘correlations’ to ‘correlation’ in TS 2.1.1.1 is an editorial change that has no effect on the requirements of the TS, and is an appropriate correction since the associated correlation (WRB-2M) is a singular correlation, and is therefore, acceptable.

The licensee also proposed to revise TS 3.7.1.1, Action “a,” and Table 3.7-1 in TS 3/4.7.1, “Turbine Cycle Safety Valves” by revising the maximum allowable power (percent of rated thermal power) allowed as restricted by the number of operable MSSVs per SG.

As part of the change the licensee proposed to add a section to the TRM, "Feedwater Ultrasonic Flow Meter (UFM) Calorimetric." The proposed section of the TRM would add requirements for plant operation when the LEFM or the Plant Process Computer (PPC) calorimetric calculation are not functional. The TRM allows operation at the uprated condition with the LEFM not functional for up to 48 hours with some restrictions. If the LEFM cannot be restored to functional within 48 hours, the use of LEFM data as input to the thermal power calorimetric calculations is suspended and the thermal power level is reduced to the pre-uprate value of 3,650 MWt (98.4 percent RTP). The TRM also establishes surveillance requirements (SR) to verify that the LEFM is functional. If the PPC calorimetric calculation is not functional the TRM directs monitoring of alternate power indications to verify power is within the licensed limit. If the PPC calculation cannot be restored to functional by the next required performance of power range channel to heat balance comparison per TS Table 4.3-1, the TRM directs that power be reduced to the pre-uprate RTP of 3,650 MWt (98.4 percent RTP) within one hour. The power range channel to heat balance SR frequency is established by the surveillance frequency control program.

As discussed throughout this SE, the NRC staff has determined that the licensee's proposal to increase the RTP from 3,650 MWt to 3,709 MWt as part of an MUR power uprate is acceptable.

The licensee stated that the Millstone 3 calorimetric power calculation was revised to normalize the feedwater flow rate, feedwater temperature, and feedwater pressure from the existing feedwater instruments based on inputs from the LEFM. In this safety evaluation, the NRC found that the site-specific calculations provided by the licensee, combined with operation of the LEFM as described in the LAR will ensure operation such that the power levels assumed in the analyses will not be exceeded.

The licensee evaluated maximum reactor power levels for various numbers of MSSVs inoperable to assure that the MSSVs remaining operable could perform their safety functions at the uprated conditions. The proposed change provided reduced reactor power levels corresponding to MSSV operability status. The previous allowable power levels for 4, 3, and

2 MSSVs per SG operable were 60.1, 42.8, and 25.5 percent RTP respectively. The proposed values are 59, 41, and 24 percent RTP respectively. As discussed in this safety evaluation, the staff found these values acceptable.

Therefore, the NRC staff has determined that changing the RTP from 3,650 MWt to 3,709 MWt as stated in the OL, and in TS 1.27, Definitions – Rated Thermal Power, is acceptable.

3.7.3 Conclusion

The NRC staff has reviewed the licensee's proposed OL and TS changes associated with the implementation of the 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 associate 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 Millstone 3 from 3,650 MWt to 3,709 MWt and make associated changes to the TSs is acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission’s regulations, the Connecticut State official was notified on September 1, 2021, of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment 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. The NRC staff has determined 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 has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on this finding (April 1, 2021; 86 FR 17211). 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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Date: November 9, 2021

LIST OF ACRONYMS

ACRONYM	DEFINITION
°F	degrees Fahrenheit
10 CFR	Title 10 of the *Code of Federal Regulations*
AC	alternating current
ADP	atmospheric dump valve
AFW	auxiliary feedwater
ALARA	as low as reasonably achievable
ANSI	American National Standards Institute
AOO	anticipated operational occurrence
AOR	analysis of record
AOT	allowed outage time
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
BHP	brake horsepower
BOP	balance-of-plant
BPV	ASME Boiler and Pressure Vessel Code
CCS	component cooling system
CLB	current licensing basis
CLTP	current licensed thermal power
COLR	core operating limits report
COPS	cold overpressure protection system
CRDM	control rod drive mechanism
CT	completion time
CVCS	chemical and volume control system
DBA	design basis accident
dc	direct current
DENC	Dominion Energy Nuclear Connecticut
DNB	departure from nucleate boiling
DNBR	departure from nuclear boiling ratio
ECCS	emergency core cooling system
EDG	emergency diesel generator
EFPY	effective full-power year
EOL	end of life
EPRI	Electric Power Research Institute
EQ	environmental qualification
FIV	flow-induced vibration
ft^2	square feet (feet squared)
FW	feedwater
GDC	General Design Criterion
gpm	gallons per minute
HELB	high energy line break
HVAC	heating, ventilation, and air conditioning
HZP	hot zero power

ACRONYM	DEFINITION
IPBD	isolated phase bus duct
in^2	square inches (inches squared)
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
MCCs	motor control centers
MELB	moderate energy line break
MFW	main feedwater
MG	main generator
Mlbm/hr	million pounds mass per hour
MOV	motor-operated valve
MPS3	Millstone Power Station, Unit No. 3
MS	main steam
MSIV	main steam isolation valve
MSSV	main steam safety valve
MT	main transformer
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
psi	pounds per square inch
psia	pounds per square inch absolute
psig	pounds per square inch gauge
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
REM	Radioactive Effluent Monitoring
RFOL	Renewed Facility Operating License
RG	regulatory guide
RHR	residual heat removal
RIS	Regulatory Issue Summary
RPV	reactor pressure vessel
RTDP	Revised Thermal Design Procedure
RTP	rated thermal power

ACRONYM	DEFINITION
RV	reactor vessel
SBO	station blackout
SE	safety evaluation
SFP	spent fuel pool
SG	steam generator
SGBS	steam generator blowdown system
SGTR	steam generator tube rupture
SLB	steam line break
SRP	Standard Review Plan
SRSS	square-root-sum-of-squares
SSCs	structures, systems, and components
SST	station service transformer
Sv	sievert
TCD	thermal conductivity degradation
T_{cold}	cold leg temperature
T_{hot}	hot leg temperature
TEDE	total effective dose equivalent
TS	technical specification
UFM	ultrasonic flow meter
FSAR	updated final safety analysis report
UHS	ultimate heat sink
USE	Charpy upper-shelf energy
V	Volt

SUBJECT: MILLSTONE POWER STATION, UNIT NO. 3 - ISSUANCE OF AMENDMENT NO. 280 REGARDING MEASURMENT UNCERTAINTY RECAPTURE POWER UPRATE (EPID L-2020-LLS-0002) DATED NOVEMBER 9, 2021

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