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

October 9, 2020

Ms. Cheryl A. Gayheart
Regulatory Affairs Director
Southern Nuclear Operating Company
3535 Colonnade Parkway
Birmingham, AL 35243

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENTS RE: MEASUREMENT UNCERTAINITY RECAPTURE POWER UPRATE (EPID L-2019-LLS-0002)

Dear Ms. Gayheart:

The U.S. Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 230 to Renewed Facility Operating License No. NPF-2 and Amendment No. 227 to Renewed Facility Operating License (RFOL) No. NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2, (Farley) respectively. The amendments are in response to your application dated October 30, 2019, as supplemented by letters dated November 25, 2019, April 22, May 8, May 29, and July 13, 2020.

The amendments revise the RFOLs paragraph 2.C(1) "Maximum Power Level"; Technical Specification (TS) 1.1 "Definitions, 'Rated Thermal Power (RTP),'" TS 2.1.1 "Reactor Core SLs [Safety Limits]," TS 3.4.1 "RCS [Reactor Coolant System] Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and TS 5.6.6 "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," to implement a measurement uncertainty recapture power uprate. Specifically, the amendments authorize an increase in the maximum licensed RTP from 2,775 megawatts thermal (MWt) to 2,821 MWt, which is an increase of approximately 1.7 percent.

The amendments also grant approval for SNC to apply WCAP-18124-NP-A, "Fluence Determination with RAPTOR-M3G and FERRET," to Farley in a limited application to predict fluence for non-beltline reactor vessel material.

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

Sincerely,

/RA/

Shawn A. Williams, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-348 and 50-364

Enclosures:
1. Amendment No. 230 to NPF-2
2. Amendment No. 227 to NPF-8
3. Safety Evaluation

cc: Listserv

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

SOUTHERN NUCLEAR OPERATING COMPANY

ALABAMA POWER COMPANY

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 230
Renewed License No. NPF-2

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

   A. The application for amendment to the Joseph M. Farley Nuclear Plant, Unit 1 (the facility), Renewed Facility Operating License No. NPF-2 (the license) filed by Southern Nuclear Operating Company (the licensee), October 30, 2019, as supplemented by letters dated November 25, 2019, April 22, 2020, May 8, 2020, May 29, 2020, and July 13, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment. Paragraphs 2.C.(1) and 2.C.(2) of the license are hereby amended to read as follows:

   2.C.(1) Maximum Power Level

   Southern Nuclear is authorized to operate the facility at steady state reactor core power levels not in excess of 2821 megawatts (thermal). Prior to attaining the power level, Alabama Power Company shall complete the preoperational tests, startup tests and other items identified in Attachment 2 to this renewed license in the sequence specified. Attachment 2 is an integral part of this renewed license.

   2.C.(2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 230, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

3. This amendment is effective as of its date of issuance and shall be implemented within 180 days from the completion of the Unit 1 refueling outage scheduled for March-April 2021.

FOR THE NUCLEAR REGULATORY COMMISSION

Caroline L. Carusone
Digitally signed by Caroline L. Carusone
Date: 2020.10.09 16:53:37 -04'00'

Caroline L. Carusone, Deputy 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: October 9, 2020

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

SOUTHERN NUCLEAR OPERATING COMPANY

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 227
Renewed License No. NPF-8

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

   A. The application for amendment to the Joseph M. Farley Nuclear Plant, Unit 2 (the facility), Renewed Facility Operating License No. NPF-8 (the license) filed by Southern Nuclear Operating Company (the licensee), dated October 30, 2019, as supplemented by letters dated November 25, 2019, April 22, 2020, May 8, 2020, May 29, 2020, and July 13, 2020,complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment. Paragraphs 2.C.(1) and 2.C.(2) of the license are hereby amended to read as follows:

   2.C.(1) Maximum Power Level

   Southern Nuclear is authorized to operate the facility at steady state reactor core power levels not in excess of 2821 megawatts thermal.

   2.C.(2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 227 are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

3. This amendment is effective as of its date of issuance and shall be implemented within 180 days of the completion of the Unit 2 refueling outage scheduled for October-November 2020.

FOR THE NUCLEAR REGULATORY COMMISSION

Caroline L. Carusone

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Caroline L. Carusone, Deputy 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: October 9, 2020

ATTACHMENT TO JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

LICENSE AMENDMENT NO. 230

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

AND LICENSE AMENDMENT NO. 227

TO RENEWED FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

Replace the following pages of the Renewed Facility Operating Licenses and Appendix "A" Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
License	License
NPF-2, page 3	NPF-2, page 3
NPF-2, page 4	NPF-2, page 4
NPF-8, page 3	NPF-8, page 3
TSs	TSs
1.1-5	1.1-5
2.0-1	2.0-1
3.4.1-1	3.4.1-1
5.6-5	5.6-5

in Houston County, Alabama in accordance with the procedures and limitations set forth in this renewed license;

(2) Alabama Power Company, pursuant to Section 103 of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess but not operate the facility at the designated location in Houston County, Alabama in accordance with the procedures and limitations set forth in this renewed license;

(3) Southern Nuclear, 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;

(4) Southern Nuclear, 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;

(5) Southern Nuclear, 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

(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

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

(1) <u>Maximum Power Level</u>

Southern Nuclear is authorized to operate the facility at steady state reactor core power levels not in excess of 2821 megawatts (thermal). Prior to attaining the power level, Alabama Power Company shall complete the preoperational tests, startup tests and other items identified in Attachment 2 to this renewed license in the sequence specified. Attachment 2 is an integral part of this renewed license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 230, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

(3) Additional Conditions

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

a. Southern Nuclear shall not operate the reactor in Operational Modes 1 and 2 with less than three reactor coolant pumps in operation.

b. Deleted per Amendment 13

c. Deleted per Amendment 2

d. Deleted per Amendment 2

e. Deleted per Amendment 152

Deleted per Amendment 2

f. Deleted per Amendment 158

g. Southern Nuclear shall maintain a secondary water chemistry monitoring program to inhibit steam generator tube degradation. This program shall include:

1) Identification of a sampling schedule for the critical parameters and control points for these parameters;

2) Identification of the procedures used to quantify parameters that are critical to control points;

3) Identification of process sampling points;

4) A procedure for the recording and management of data;

5) Procedures defining corrective actions for off control point chemistry conditions; and

(2) Alabama Power Company, pursuant to Section 103 of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess but not operate the facility at the designated location in Houston County, Alabama in accordance with the procedures and limitations set forth in this renewed license.

(3) Southern Nuclear, 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;

(4) Southern Nuclear, 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;

(5) Southern Nuclear, 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

(6) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

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

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at reactor core power levels not in excess of 2821 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 227, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications.

(3) Deleted per Amendment 144
(4) Deleted per Amendment 149
(5) Deleted per Amendment 144

1.1 Definitions

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates and the Low Temperature Overpressure Protection System applicability temperature, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6.
QUADRANT POWER TILT RATIO (QPTR)	QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2821 MWt.
REACTOR TRIP SYSTEM (RTS) RESPONSE TIME	The RTS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RTS trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC, or the components have been evaluated in accordance with an NRC approved methodology.
SHUTDOWN MARGIN (SDM)	SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: a. All rod cluster control assemblies (RCCAs) are fully inserted except for the single RCCA of highest reactivity worth, which is assumed to be fully withdrawn. However, with all RCCAs verified fully inserted by two independent means, it is not necessary to account for a stuck rod in the SDM calculation. With any RCCA not capable of being fully inserted, the reactivity worth of the RCCA must be accounted for in the determination of SDM; and

(continued)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the departure from nucleate boiling ratio (DNBR) shall be maintained within the 95/95 DNB criterion correlation specified in the COLR.

2.1.1.2 In MODES 1 and 2, the peak fuel centerline temperature shall be Maintained < 5080°F, decreasing by 9°F per 10,000 MWD/MTU.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2735 psig.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified in the COLR. The minimum RCS total flow rate shall be ≥ 258,000 GPM and ≥ the limit specified in the COLR.

APPLICABILITY: MODE 1.

-----------------------------------NOTE-----------------------------------
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

7. WCAP-11397-P-A "Revised Thermal Design Procedure," April 1989

   (Methodology for LCO 2.1.1-Reactor Core Safety Limits, LCO 3.4.1-RCS Pressure, Temperature and Flow Departure from Nucleate Boiling Limits.)

8. WCAP-13749-P-A, “Safety Evaluation Supporting the Conditional Exemption of the Most Negative EOL Moderator Temperature Coefficient Measurement,” March 1997.

   (Methodology for LCO 3.1.3 - Moderator Temperature Coefficient.)

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

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

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. The reactor coolant system pressure and temperature limits, including heatup and cooldown rates and the LTOP System applicability temperature, shall be established and documented in the PTLR for the following:

   LCO 3.4.3, “RCS Pressure and Temperature (P/T) Limits,” and
   LCO 3.4.12, “Low Temperature Overpressure Protection (LTOP) System.”

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in WCAP-14040-A, Revision 4, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” May 2004. WCAP-18124-NP-A, Revision 0, “Fluence Determination with RAPTOR-M3G and FERRET,” July 2018, may be used as an alternative to Section 2.2 of WCAP-14040-A.

c. The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.

(continued)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 230 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-2

AND

AMENDMENT NO. 227 TO RENEWED FACILITY OPERATING LICENSE NO. NPF-8

SOUTHERN NUCLEAR OPERATING COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-348 AND 50-364

1.0 INTRODUCTION

By letter dated October 30, 2019 (Reference 1), as supplemented by letters dated November 25, 2019 (Reference 2), April 22 (Reference 3), May 8 (Reference 4), May 29 (Reference 5), and July 13, 2020 (Reference 6), Southern Nuclear Operating Company (SNC, the licensee) submitted a License Amendment Request (LAR) proposing changes to the Renewed Facility Operating Licenses (RFOLs) and the Technical Specifications (TSs) for Joseph M. Farley Nuclear Plant, Units 1 and 2 (Farley).

Specifically, the licensee proposed to revise the RFOLs paragraph 2.C(1) "Maximum Power Level"; TS 1.1 “Definitions, ‘Rated Thermal Power (RTP),’” TS 2.1.1 "Reactor Core SLs [Safety Limits]," TS 3.4.1 "RCS [Reactor Coolant System] Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," and TS 5.6.6 "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)" to implement a measurement uncertainty recapture power uprate (MUR-PU) that would increase the maximum licensed RTP from 2,775 megawatts thermal (MWt) to 2,821 MWt. The licensee also requested approval to apply WCAP-18124-NP-A, "Fluence Determination with RAPTOR-M3G and FERRET" (Reference 7), to Farley in a limited application to predict fluence for non-beltline reactor vessel material.

On February 4, 2020, the U. S. Nuclear Regulatory Commission (NRC) staff published a proposed no significant hazards consideration (NSHC) determination in the *Federal Register* (85 FR 6231) for the proposed amendments that included the submittal dated October 30, 2019, and the supplement dated November 25, 2019. The supplemental letters dated April 22, May 8, May 29, and July 13, 2020, provided additional information that clarified the application, did not expand the scope of the application as noticed, and did not change the NRC staff's proposed NSHC determination as published in the *Federal Register*.

2.0 BACKGROUND

2.1 Measurement Uncertainty Recapture Power Uprates

Nuclear power plants are licensed to operate at a specified maximum RTP. Appendix K, "[Emergency Core Cooling System] ECCS Evaluation Models," of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, formerly required licensees to assume that the reactor has been operating continuously at a power level at least 1.02 times the licensed power level when performing loss-of-coolant accident (LOCA) and ECCS analyses. This requirement was included to ensure that instrumentation uncertainties were adequately accounted for in the safety analyses. In practice, many of the design bases analyses assumed a 2 percent power uncertainty, consistent with 10 CFR Part 50, Appendix K.

A change to the Commission's regulations in 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 allowed licensees to use a power level less than 1.02 times the RTP for the LOCA and ECCS analyses, but not a power level less than the licensed power level, based on the use of state-of-the art feedwater (FW) flow measurement devices that provide a more accurate calculation of power. Licensees can use a lower uncertainty in the LOCA and ECCS analyses provided that the licensee has demonstrated that the proposed value accounts adequately for instrumentation uncertainties. As there continues to be substantial conservatism in other Appendix K requirements, sufficient margin to ECCS performance in the event of a LOCA is preserved.

However, this change to 10 CFR 50, Appendix K, did not authorize increases in licensed power levels for individual nuclear power plants. As the licensed power level for a plant is contained in its operating license, licensees seeking to raise the licensed power level must submit a LAR, which must be reviewed and approved by the NRC staff.

In order to provide guidance to licensees seeking an MUR-PU on the basis of improved FW flow measurement, the NRC issued Regulatory Issue Summary (RIS) 2002-03, "Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications," dated January 31, 2002 (Reference 8). 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-PU LARs. While RIS 2002-03 does not constitute an NRC requirement, its use aids licensees in the preparation of their MUR-PU LAR, while also providing guidance to the NRC staff to conduct its review. The licensee stated in its LAR that its submittal followed the guidance of RIS 2002-03.

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

The instruments originally installed to measure FW flow rate in existing nuclear power plants were usually a venturi or a flow nozzle, each of which generates a differential pressure proportional to the FW velocity in the pipe. However, errors in the determination of flow rate can be introduced due to venturi fouling and, to a lesser extent, flow nozzle fouling, the transmitter, and the analog-to-digital converter.[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 computer-controlled electronic transducers that do not have differential pressure elements that are susceptible to fouling.

2.2 Implementation of an MUR-PU at Farley

The licensee intends to use UFMs developed by the Cameron International Corporation (Cameron, formerly known as Caldon Ultrasonic Inc. (Caldon)), specifically the leading edge flow meter (LEFM) CheckPlus System, which provides a more accurate measurement of FW flow as compared to the accuracy of the venturi flow meter-based instrumentation originally installed at Farley. Installation of these UFMs to measure FW flow would allow the licensee to operate the plant with a reduced instrument uncertainty margin and an increased power level in comparison to its currently licensed thermal power (CLTP). Based on the use of the Caldon instrumentation to determine core power level with a power measurement uncertainty of approximately 0.3 percent, SNC proposes to reduce the licensed power uncertainty required by 10 CFR 50, Appendix K, by approximately 1.66 percent (hereafter referred to as 1.7 percent).

The Cameron LEFM CheckPlus System was developed over a number of years. Cameron submitted Topical Report, 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 9), 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. ER-80P 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 50, Appendix K, in a safety evaluation (SE) dated March 8, 1999 (ER-80P SE) (Reference 10), which allowed a 1 percent power uprate using the LEFM. NRC staff's SE included four Criterion that should be addressed by licensees incorporating ER-80P in their plant licensing basis.

Following the publication of the changes to 10 CFR 50, Appendix K, which allowed for an uncertainty less than 2 percent, Cameron submitted Topical Report ER-160P, Rev. 0, "Supplement to Engineering Report ER-80P: Basis for a Power Uprate with the LEFM System" (Reference 11). The NRC staff approved a plant-specific precedent using ER-160P by letter dated January 19, 2001 (Reference 12), for a power uprate of up to 1.4 percent at Watts Bar Nuclear Plant, Unit 1.

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

Subsequently, in an SE dated December 20, 2001 (Reference 13), the NRC staff approved a plant-specific precedent using ER-157P, Rev. 5, “Supplement to Engineering Report ER-80P: Basis for a Power Uprate with the LEFM or LEFM CheckPlus System” (Reference 14), for a power uprate of up to 1.7 percent using the CheckPlus system.

Cameron submitted ER-157P, Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or an LEFM CheckPlus System, Rev. 8, (ER-157P, Rev. 8) in May 31, 2008 (Reference 15) and an associated errata on October 15, 2010 (Reference 16). ER-157P, Rev. 8, corrects minor errors in Rev. 5, provides clarifying text, and incorporates 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 safety evaluation for ER-157P, Rev. 8 (Reference 17), dated August 16, 2010 (ER-157P SE), the NRC staff approved ER-157P, Rev. 8, and associated errata, subject to five Criterion that should be addressed by licensees incorporating ER-157P, Rev. 8, in their plant licensing basis

2.3 Regulatory Evaluation

Due to the numerous technical review sections of a MUR-PU, a regulatory evaluation section is included for each technical section rather than consolidated here. Chapter 3, Section 3.1, “Conformance with NRC General Design Criteria [GDC],” of the Farley Updated Final Safety Analysis Report, Revision 29 (UFSAR) (Reference 18), describes how Farley complies with the GDCs.

3.0 TECHNICAL EVALUATION

3.1 Safety Systems

3.1.1 Feedwater Flow Measurement Technique and Power Measurement Uncertainty

3.1.1.1 Regulatory Evaluation

Topical Report ER-80P and its supplement, Topical Report ER-157P, describe the Cameron LEFM CheckPlus System for the measurement of feedwater flow and provide a generic basis for the proposed power uprate. In its application, the licensee proposed to use the Cameron LEFM CheckPlus system to decrease the uncertainty in the measurement of FW flow, thereby decreasing the power level measurement uncertainty from 2.0 percent to a bounding value of 0.34 percent. The licensee developed its application consistent with the guidelines in RIS 2002-03. The regulatory guidance provided in RIS 2002-03, provides a method to satisfy the requirements of 10 CFR 50, Appendix K.

Regulatory Guide (RG) 1.105, Revision 3, “Setpoints for Safety-Related Instrumentation,” (Reference 19) describes a method acceptable to the NRC for complying with the NRC regulations for assuring that setpoints for safety-related instrumentation are initially within and remain within the technical specification limits.

3.1.1.2 Technical Evaluation

The NRC staff's review addressed the proposed plant-specific use of the feedwater flow measurement technique and the power increase gained as a result of implementing this technique in accordance with RIS 2002-03, Attachment 1, Section I, "Feedwater flow measurement technique and power measurement uncertainty," Items A through H. The NRC staff conducted its review to confirm consistency between the licensee's submittal which described the implementation of the proposed feedwater flow measurement device and the NRC staff approved Topical Reports ER-80P and ER-157P, as well as the acceptability of the provided calculations. The licensee's letter dated October 30, 2019, included proprietary and non-proprietary versions of the following Cameron Engineering Reports:

- ER-1180P, Revision 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Farley Unit 1 Using the LEFM ✓+ System,"
- ER-1181P, Revision 1, "Bounding Uncertainty Analysis for Thermal Power Determination at Farley Unit 2 Using the LEFM ✓+ System,"
- ER-1182P, Revision 1 and Revision 1 Errata, "Meter Factor Calculation and Accuracy Assessment for Farley Unit 1" and,
- ER-1183P, Revision 1, "Meter Factor Calculation and Accuracy Assessment for Farley Unit 2".

3.1.1.2.1 Leading Edge Flow Meter Technology and Measurement

As stated in the application, the Cameron LEFM Check and LEFM CheckPlus are advanced ultrasonic systems that accurately determine the volume flow and temperature of feedwater in nuclear power plants that can be used to compute reactor core thermal power.

3.1.1.2.2 Licensee's Response to RIS 2002-03, Attachment 1, Section I

Attachment 1 of RIS 2002-03, provides guidance to licensees on how to address the issues of FW flow measurement technique and power measurement uncertainty in MUR-PU license amendment requests. The following subsections of this SE discuss the licensee's response to these guidelines and the NRC staff's evaluation of the licensee's responses.

3.1.1.2.2.1 Items A, B, and C of Section I, Attachment 1 to RIS 2002-03

Items A and B request the licensee to identify and Reference the documents that form the regulatory basis for the LAR. The licensee stated that the feedwater flow measurement technique at Farley is a Cameron LEFM CheckPlus ultrasonic multi-path transit time flowmeter as described in ER-80P, Rev. 0, and ER-157P, Rev. 8.

Item B requests the licensee to Reference the NRC approval of the proposed feedwater flow measurement technique. The licensee stated that the Cameron LEFM check instruments were reviewed and approved by the NRC in the safety evaluation report (SER) contained in engineering reports ER-80P and ER-157P.

Item C requests a discussion of the plant-specific implementation of the guidelines in the topical reports and the NRC staff's safety evaluation approving the topical report for the feedwater flow measurement technique. The licensee stated that the LEFM CheckPlus UFM is installed in Farley and are operated in accordance with the manufacturer's requirements. The system will be used for continuous calorimetric power determination by direct links with the Farley Integrated Plant Computer. Even though the system is not safety related, it is designed and manufactured in accordance with Cameron's Quality Assurance Program, which is certified to ISO 9001:2015 and supplemented by the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

The licensee stated that the LEFM spool pieces installed in Farley are installed sufficiently upstream of the existing feedwater flow venturis and downstream of any piping components such that no adverse interaction is created. The LEFM CheckPlus UFM system is composed of three metering sections for the feedwater lines (each metering section includes two electronic transmitters, two pressure transmitters, 16 acoustic transducers, RTDs [resistance temperature detectors], and a pressure port), and the system includes two Processing Units, and instrument cables for each transducer in the system. Each of the LEFM spool pieces were calibrated at the Alden Research Laboratory facility using a hydraulic duplicate of the principal hydraulic features of the plant configuration. Calibration tests determined the meter factor calibration constant for each of the Farley Units' LEFM. The meter factor provides a small correction to the numerical integration to account for fluid velocity profile specifics and any dimensional measurement errors. Parametric tests were also performed at the Alden Research Laboratory facility to determine meter factor sensitivity to upstream hydraulics. Copies of the Meter Factor Calculation and Accuracy assessments were provided as attachments 7 and 8 of the LAR and were reviewed by NRC staff who determine that the measurements of flow and temperature were sufficiently accurate.

<u>NRC Staff Conclusions Regarding Items A. B. and C of Section I, Attachment 1 to RIS 2002-03</u>

The NRC staff reviewed the licensee's submittals and finds that the licensee has adequately addressed the plant-specific implementation of the Cameron LEFM CheckPlus system using the NRC-approved topical reports. Therefore, the NRC staff concludes that the implementation of the feedwater flow measurement technique and its implementation follows the guidance in terms A, B, and C of Section I of Attachment 1 to RIS 2002-03 and thus meets the regulatory requirements of 10 CFR 50, Appendix K.

3.1.1.2.2.2 Item D of Section I, Attachment 1 to RIS 2002-03

Item D requests the licensee provide the dispositions of the criteria that the NRC staff stated should be addressed when implementing the feedwater flow measurement uncertainty technique Referenced in ER-80P and ER-157P.

The licensee addressed the four Criterion described in the NRC's ER-80P SE.

<u>Criterion 1, ER-80P SE</u>

Criterion 1, ER-80P SE (Reference 10), states:

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

The licensee stated that the plant maintenance and calibration procedures are revised to incorporate Cameron's maintenance and calibration requirements and would be implemented prior to raising power above the current licensed thermal power (CLTP) of 2775 MWt. The Farley Technical Requirement Manual (TRM) will be revised to address contingencies for non-functional LEFM instrumentation. The LEFM CheckPlus will be installed at Farley during their respective units' refueling outages: Unit 2 during October 2020, and Unit 1 during March 2021. Following installation, testing will include an in-service leak test, comparisons of feedwater flow and thermal power calculated by various methods, and final commissioning testing. The licensee stated that the commissioning process provides final confirmation that the actual field performance meets uncertainty bounds established for instrumentation.

The licensee stated that the Farley LEFM CheckPlus was calibrated in site-specific model test at Alden Research Laboratory. The meter factor reports (attachments 7 and 8 of the LAR), contain the reports from the calibration's tests. The calibration tests included a site-specific model of each of the Units' hydraulic geometry.

Preventive maintenance and required training have been developed and implemented at the Farley site. The preventive maintenance program and the LEFM CheckPlus system continuous self-monitoring feature ensure that the LEFM remains bounded by ER-80P, as supplemented by ER-157P, analysis and assumptions. The NRC staff reviewed the licensee's contingency plans for the plant operation with an inoperable LEFM CheckPlus system.

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

Criterion 2, ER-80P SE

Criterion 2, ER-80P SE, states:

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

The licensee stated that Criterion 2 does not apply to Farley since they do not have the LEFM CheckPlus systems installed at this time. The licensee stated that, after the LEFM CheckPlus is installed and operational, data will be collected comparing the LEFM CheckPlus operating data to the venturi data to verify consistency between thermal power calculation based on LEFM data and other plant parameters.

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

Criterion 3, ER-80P SE

Criterion 3, ER-80P SE, states:

> The licensee should confirm that the methodology used to calculate the uncertainty of the LEFM in comparison to the current feedwater instrumentation

> is based on accepted plant setpoint methodology (with regard to the development of instrument uncertainty). If an alternative methodology is used, the application should be justified and applied to both venturi and ultrasonic flow measurement instrumentation installations for comparison.

The licensee stated that the LEFM CheckPlus uncertainty calculation is based on the American Society of Mechanical Engineers (ASME) Performance Test Code (PTC) 19.1 and Alden Research Laboratory Inc. calibration tests. Cameron performed unit-specific bounding uncertainty analysis for Farley Units 1 and 2 (attachments 5 and 6 to the LAR), which were reviewed by NRC staff.

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

<u>Criterion 4, ER-80P SE</u>

Criterion 4, ER-80P SE, states:

> Licensees for plant installations where the ultrasonic meter (including LEFM) was not installed with flow elements calibrated to a site specific piping configuration (flow profiles and meter·factors not representative of the plant specific installation), should provide additional justification for 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 Criterion 4 is not applicable to Farley since the LEFM system is not yet installed. The calibration factors used for each Units' LEFMs are based on the analysis contained in these reports. The uncertainties in the calibration factor for the spools are based on the Cameron site-specific engineering reports provided in LAR Attachments 5 and 6. Final acceptance of this analysis will occur after the completion of the commissioning process that will verify the bounding calibration.

Based on its review of the licensee's LAR, the NRC staff concludes that the licensee has adequately addressed Criterion 4 in ER-80P SE.

<u>Criterion 1, ER-157P SE</u>

Criterion 1, ER-157P SE (Reference 17), states:

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

The licensee responded that operation above 2775 MWt will be limited to 72 hours if the LEFM CheckPlus system is not functional.

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

Criterion 2, ER-157P SE

Criterion 2, ER-157P SE states:

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

The licensee stated that Farley will treat a single path or single plane failure in the LEFM CheckPlus as a complete system failure.

Based on its review of the licensee's LAR, the NRC staff concludes that since the licensee will not operate the LEFM with only one train, the licensee has adequately addressed Criterion 2 in ER-157P SE.

Criterion 3, ER-157P SE

Criterion 3, ER-157 P SE states:

> An applicant [licensee] with a comparable geometry can Reference the above Section 3.2.1 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.

The licensee stated that Farley will not consider a CheckPlus system with disabled components as a separate category; such a condition will be considered as a non-functional LEFM and the same actions identified in response to Criterion 1 above will be implemented.

Based on its review of the licensee's LAR, the NRC staff concludes that the licensee has adequately addressed Criterion 3 in ER-157P SE.

Criterion 4, ER-157P SE

Criterion 4, ER-157P SE states:

> An applicant [licensee] that requests a MUR with the upstream flow straightener configuration discussed in Section 3.2.2 [in Reference 14] should provide justification for claimed CheckPlus uncertainty that extends the justification provided in Reference 17 [(Reference 20)]. Since the 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 licensee stated that the feedwater piping configurations at Farley do not necessitate or use upstream flow straighteners.

Based on its review of the licensee's LAR, the NRC staff concludes that the licensee has adequately addressed Criterion 4 in ER-157P SE.

Criterion 5, ER-157P SE

Criterion 5, ER-157P SE states:

> An applicant [licensee] 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 18 [(Reference 21)].

The licensee stated that the uncertainty associated with the moisture carryover is on the same order of magnitude as the other calorimetric uncertainties. As discussed in Engineering Report ER-764, "The Effect on the Distribution of the Uncertainty in Steam Moisture Content on the Total Uncertainty in Thermal Power" (Reference 20), the uncertainty associated with steam moisture would not be classified as a "large uncertainty" in the moisture content. Therefore, Criterion 5 does not apply to Farley.

The NRC staff considers this uncertainty in steam moisture content to be small and not a significant factor in the calculation of the total power uncertainty. Based on its review of the licensee's LAR, the NRC staff concludes that the licensee has adequately addressed Criterion 5 in ER-157P SE.

NRC Staff Conclusions Regarding Item D of Section I, Attachment 1 to RIS 2002-03

The NRC staff evaluated the licensee's responses to applicable criteria from ER-80P and ER-157P, Revision 8, as described in this section. The NRC staff reviewed these assessments by the licensee and finds them acceptable. Therefore, the NRC staff concludes that the licensee has adequately addressed the guidance in Item D of Section I of Attachment 1 to RIS 2002-03, and thus meets the regulatory requirements of 10 CFR 50, Appendix K.

3.1.1.2.2.3 Item E of Section I, Attachment 1 to RIS 2002-03

Item E requests the licensee provide a calculation of the total power measurement uncertainty at the plant, identifying explicitly all parameters and their individual contribution to the power uncertainty.

The licensee provided the calculations requested in Item E, in LAR Attachment 5, "Bounding Uncertainty Analysis for Thermal Power Determination at Farley Unit 1 Using the LEFM Check Plus System," and Attachment 6, "Bounding Uncertainty Analysis for Thermal Power Determination at Farley Unit 2 Using the LEFM Check Plus System." Proprietary and non-proprietary versions of Attachment 5 and 6 were provided in the LAR.

Attachments 5 and 6 describe the methodology for the analysis of the uncertainty contribution of the LEFM CheckPlus system to the overall mass flow and thermal power uncertainty of Farley Unit 1 and 2. The LEFM CheckPlus UFM system is permanently installed in Farley Units 1 and 2 and operated in accordance with the manufacturer's requirements. The system is used for continuous calorimetric power determination by direct links with the Farley Unit 1 and 2 Integrated Plant Computer. The system incorporates self-verification features to ensure that the hydraulic profile and signal processing requirements are met within its design basis uncertainty analysis. LEFM CheckPlus system is designed and manufactured in accordance with Cameron's Quality Assurance Program, which is certified to ISO 9001:2015 and supplemented by the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

The bounding analysis for Farley units utilizes actual dimensions of the equipment and plant parameters at operating conditions. All errors and biases for the plant and operating parameters are calculated and combined according to procedures in ASME PTC 19-1-1985 (Reference 22). Actual values for the uncertainties in length measurements, time measurements, and calibration coefficients (meter factor) are employed in the bounding analysis. If the errors or biases individual elements are all caused by a common or systematic boundary condition, the total error is found by summing up all the errors. If the errors in parameters that are independent of each other, then the probability theory requires that the total uncertainty be determined by the root sum squares (RSS) of all the uncertainties for 95 percent confidence and probability value.

Inputs and Uncertainty Items

Attachments 5 and 6 includes the LEFM Check Plus inputs and uncertainty items for the bounding uncertainty items calculations. Appendix A.1 of the attachments 5 and 6 calculates other key dimensions and factors from these inputs, such as, the face-to-face distance between pairs of transducer assemblies, which is used by the LEFM Check Plus for the computation of mass flow and temperature. Appendix A.2 of attachments 5 and 6 calculates the uncertainties in mass flow and temperature as computed by the LEFM CheckPlus. Appendix A.3 of attachments 5 and 6 describes meter factor (calibration) uncertainties.

Cameron performs a calibration test to determine the meter calibration constant or meter factor. The meter factor provides a small correction to the numerical integration scheme to obtain the fluid velocity profile and any dimensional measurement errors. The calibration test was performed at Alden Research Laboratories (Alden), an independent hydraulics laboratory. Calibration included Reference flow rates that were determined by Alden using a weigh tank, fill times, fluid temperature and barometric pressure measurements. All elements of the calibration measurements such as, weigh tank scale, transit time measurements, thermometers and pressure gauges are based on NIST (National Institute of Standards and Technology) standards. Meter factor is determined by comparing flow meter outputs to the Reference flow rates.

For Farley Units 1 and 2, the bounding uncertainty analysis values are listed in the Cameron Engineering Reports submitted with the LAR, as supplemented (Unit 1: ER-1180 P/NP Rev.1 and Rev. 1 Errata; Unit 2: ER-1181 P/NP Rev. 1 and Rev. 1 Errata).

NRC Staff Conclusions Regarding Item E of Section I, Attachment 1 to RIS 2002-03

The NRC staff reviewed the submittal and finds that the licensee properly identified the parameters associated with the thermal power measurement uncertainty, provided individual measurement uncertainties, and calculated the overall thermal power uncertainty. The NRC staff finds that the licensee provided calculations of the total power measurement uncertainty for Farley Units 1 and 2 and identified the parameters and their individual contributions to the overall thermal power uncertainty. Therefore, the NRC staff concludes that the licensee has adequately addressed the guidance in Item E of Section I of Attachment 1 to RIS 2002-03, and thus meets the regulatory requirements of 10 CFR 50, Appendix K.

3.1.1.2.2.4 Item F of Section I, Attachment 1 to RIS 2002-03

Item F of Section I, Attachment 1 to RIS 2002-03, requests that licensees provide information to specifically address the following aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric:

i. Maintaining C

In Attachment 4 of the LAR, Section I.1.F.i, the licensee states that the calibration and maintenance of LEFM CheckPlus will be performed using procedures based on the Cameron LEFM CheckPlus technical manuals.

ii. Controlling Software and Hardware Configuration

In Attachment 4 of the LAR, Section I.1.F.ii, the licensee states the following:

> Cameron's manufacturing and quality programs are certified to ANSI/ISO/ASQC9001 and supplemented by quality assurance criteria for Nuclear Power Plants defined in 10 CFR 50 Appendix B and 10 CFR 21 for Reporting of Defects and Nonconformance. Cameron software is developed and maintained under a Verification and Validation (V&V) program consistent with ASME NQA-1a-1999 Subpart 2.7. After installation, the LEFM CheckPlus system software configuration will be maintained using existing procedures and processes.

iii. Performing Corrective Actions

In Attachment 4 of the LAR, Section I.1.F.iii, the licensee states that LEFM will be monitored by Farley personnel and deficiencies will be documented in accordance with the Farley corrective action program (CAP) and corrective actions will be identified and implemented.

iv. Reporting Deficiencies to the Manufacturer

In Attachment 4 of the LAR, Section I.1.F.iv, the licensee states that deficiencies associated with the vendor's processes or equipment will be reported in accordance with the Farley CAP.

v. Receiving and Addressing Manufacturer Deficiency Reports

In Attachment 4 of the LAR, Section I.1.F.iv, the licensee states that the CheckPlus is under Cameron's V&V program, and procedures are maintained by Cameron for user notification of important deficiencies. Additionally, Farley processes requires applicable deficiencies to be documented and addressed in the Farley CAP.

<u>NRC Staff Conclusions Regarding Item F of Section I, Attachment 1 to RIS 2002-03</u>

Based on its review of the information above, the NRC staff finds that the licensee adequately addressed the calibration and maintenance aspects of the Cameron LEFM CheckPlus System and all other instruments affecting the power calorimetric. Therefore, the NRC staff concludes that the licensee has adequately addressed Item F of Section I of Attachment 1 to RIS 2002-03, and thus meets the regulatory requirements of 10 CFR 50, Appendix K.

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

Items G and H of Section I, Attachment 1 to RIS 2002-03 request that licensees provide a proposed allowed outage time (AOT) for the instrument, along with the technical basis for the time selected, and to propose actions to reduce power if the AOT is exceeded.

In order to address Items G and H, the licensee indicated that if the LEFM system is declared non-functional, either the LEFM system be restored to functional status within 72 hours or the power is to be reduced to no more than 2775 MWt.

In Attachment 4 of the LAR, Section I.1.G, the licensee provides its basis for an AOT of 72 hours. The LAR states that:

- The 72 hours allowed outage time is adequate to provide sufficient time for plant personnel to plan and package work orders, complete repairs, and verify normal system operation within original uncertainty bounds.
- During the AOT, the licensee proposes to use the feedwater flow from the venturis to perform the calorimetric until the LEFM is returned to functional status.
- A review of flow venturi fouling history demonstrates that fouling/de-fouling should not introduce significant error/drift over a 72-hour period.
- The LEFM CheckPlus consists of two redundant planes of transducers and a single path or single plane malfunction results in a minimal increase in feedwater flow uncertainty. For Farley, operators will conservatively respond to a single path or single plane failure in the same manner as a complete system failure.
- Operators routinely monitor other indications of core thermal power, including nuclear instrumentation system power range monitors, loop Δ-temperatures, steam flow, feed flow, turbine first stage pressure, and main generator output.

<u>NRC Staff Conclusions Regarding Items G and H of Section I, Attachment 1 to RIS 2002-03</u>

The NRC staff reviewed the information provided above for the proposed AOT. Based on its review, the NRC staff finds that the licensee provided sufficient justification for the proposed 72-hour AOT and the actions to reduce power level if the AOT is exceeded.

Therefore, the NRC staff concludes that the licensee has provided the information requested by Items G and H of Section I of Attachment 1 to RIS 2002-03, and thus meets the regulatory requirements of 10 CFR 50, Appendix K.

3.1.2 Containment Analysis

3.1.2.1 Regulatory Evaluation

For containment issues the regulation at 10 CFR, Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 4 (GDC 4), "*Environmental and dynamic effects design bases*," states, in part, that, "Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with … postulated accidents, including loss-of-coolant accidents." The NRC staff reviewed the licensee's prediction of conditions in containment during postulated accidents.

GDC 16, "*Containment design*," and GDC 50, "*Containment design basis*," address the requirements for the containment pressure resulting from the discharge of mass and energy (M&E) into the containment as a result of a postulated design-basis LOCA.

GDC 16, "*Containment design,"* states that, "Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require."

GDC 38, "*Containment heat removal*," states, in part, that, "A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels."

GDC 50, "*Containment design basis*," states, in part, that, "The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident."

3.1.2.2 Technical Evaluation

Per Item II.1 and III.1 in Attachment 1 to RIS 2002-03, the NRC staff reviewed the following areas of containment design and analysis for the LAR: Short- and Long-term LOCA containment response analyses; containment response to a main steam line break (MSLB); LOCA at a low power and reduced containment temperature; and minimum containment backpressure analysis.

3.1.2.2.1 Short-Term and Long-Term LOCA Mass and Energy Release and Containment Analysis

The short- and long-term LOCA peak containment pressure analysis is documented in UFSAR Section 6.2.1.3.4.1, "LOCA Mass and Energy Releases." The licensee indicated in Section II.1.D.iii, Item 28, of the LAR, that the large break LOCA long-term mass and energy release analysis demonstrates the ability of the containment safeguards systems to mitigate the consequences of a hypothetical LOCA. The long-term mass and energy release analysis of record (AOR) for Farley assumes a core thermal power of 2830.5 MWt, which corresponds to the licensed core power of 2775 MWt with an additional 2 percent uncertainty, which bounds MUR-PU.

The licensee stated that the short-term LOCA M&E release calculations are performed to support the reactor cavity and loop sub-compartment pressurization analyses, which in turn guarantee that the walls in the immediate proximity of the break location can maintain their structural integrity during the short pressure pulse (generally less than 3 seconds) that accompanies a LOCA. The licensee indicated that the short-term LOCA blowdown transients are characterized by the M&E releases that occur during a subcooled condition, therefore, the AOR bound the MUR-PU conditions.

In addition, the analyzed loop average temperature (another input on the analyses) does not change (583.2 °F) at MUR-PU conditions, therefore, the initial energy content of the RCS fluid assumed in the AOR does not change and the margin of safety is not reduced.

Based on the above, the NRC staff finds that the effects of the MUR-PU on the short- and long-term LOCA M&E release are bounded by the AOR and is, therefore, acceptable.

3.1.2.2.2 Postulated Secondary System Pipe Rupture Outside Containment

The NRC staff reviewed the secondary system pipe rupture M&E release effects to ensure that the safety-related equipment located outside containment are still capable to perform their intended safety function following the MUR-PU.

The licensee stated in Section II.1.D.iii, Item 33, of the LAR, that the MUR-PU only impacted the main feedwater system and portions of the auxiliary feedwater (AFW) system. The feedwater and the AFW water temperature increased from 440°F to 446°F, this increase in temperature will increase the break effluent enthalpy and associated energy release. The areas affected by a feedwater line break are in the areas affected by a main steam line break. The M&E release rates associated with a feedwater line break are bounded by the main steam line break. Farley UFSAR Appendix 3J discusses the main steam line break outside containment pressure and temperature analysis. One of the assumptions included in the AOR is a reactor power of (2785 MWt) plus a 2 percent calorimetric uncertainty. Therefore, the AOR bounds the MUR-PU proposed power level of 2821 MWt, and the NRC staff finds that the analysis is unaffected by the MUR-PU.

3.1.2.2.3 Fuel Handling Accident (FHA) in the Containment

With respect to FHA in the containment, the accident analysis is performed to demonstrate that the offsite and control room doses are within regulatory limits. The FHA analysis in Section 15.4.5, of the Farley UFSAR was performed using the NRC reviewed and approved Alternative Source Term methodology in RG 1.183 (Reference 23). The licensee described the FHA in Section II.1.D.iii, Item 25, Attachment 4 of the LAR. Since the source term was calculated at an initial power level of 102 percent RTP (2831 MWt), which bounds the MUR-PU proposed power level, the NRC staff finds that the analysis is unaffected by the MUR-PU.

3.1.2.2.4 Environmental Qualification (EQ)

With respect to EQ, the licensee states, in part, in Section II.1.D.iii, Item 29, Attachment 4 of the LAR, that the equipment and components of the equipment qualification program will continue to operate satisfactorily and perform their intended functions at the MUR-PU conditions, that the safety-related electrical equipment is qualified to survive the environment at its specific location during normal operation and during an accident. The LAR also indicates that the equipment qualification program equipment will accommodate MUR-PU conditions without exceeding electrical equipment qualification margins for the parameters of temperature, pressure, radiation, and similar parameters, as defined by Institute of Electrical and Electronics Engineers (IEEE) Standard 323-1974.

The NRC staff reviewed the EQ parameters of temperature, pressure, and radiation, with respect to any potential parameter changes due to the MUR-PU. Temperature, pressure, and radiation conditions in the containment following a large break LOCA, MSLB, or FHA accident were discussed above. As noted, the analyses for these events were performed at 102 percent RTP and bound the MUR-PU proposed power level. Therefore, the NRC staff finds that there is no EQ impact with respect to temperature, pressure, or radiation due to the MUR-PU. The NRC staff concludes that the EQ profile is conservative and acceptable with respect to operation at the proposed MUR-PU.

3.1.2.2.5 Containment Systems

The containment systems are provided to limit offsite releases following a design basis accident (DBA). These systems include the containment spray system (including recirculating water pH control system), the containment air cooling system, and containment isolation.

In Section VI.1.B, Attachment 4 of the LAR, the licensee evaluated the impact of the MUR-PU on these systems. The LAR stated that the existing containment analyses remain bounding for the MUR-PU conditions. The NRC staff reviewed the licensee's evaluation results discussed in the LAR and found that these systems are not impacted by implementation of the MUR-PU. Therefore, the NRC staff finds that they are acceptable with respect to operation at the proposed MUR-PU.

3.1.2.2.6 Containment Leakage Rate Testing Program

In Section VII.6.D "Containment Leakage Rate Testing Program," Attachment 4 of the LAR, the licensee stated that:

> The Appendix J (ILRT/LLRT) [integrated leakage rate text / local leak rate test] Containment Leakage Rate Testing Program is implemented in accordance with 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, Option B. The Farley program documents, [U]FSAR, and TSs all conform to the requirements of 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J, and are unaffected by the MUR-PU. Therefore, the Appendix J program will continue to comply with the licensing bases of 10 CFR 50.54(o) and 10 CFR Part 50, Appendix J.

The NRC staff reviewed the licensee's response and finds that Appendix J (ILRT/LLRT) Containment Leakage Rate Testing Program is unchanged with the proposed MUR-PU, and is, therefore, acceptable.

3.1.2.3 NRC Staff Conclusion Regarding Containment Systems

The NRC staff finds that the current containment analyses remain bounding for the MUR-PU. The NRC staff finds that the current peak containment pressure is less than the containment design pressure and the EQ envelope remain bounding. In addition, the previously approved analytical methods remain acceptable. Further, the NRC staff, finds that the criteria identified in GDC 4, GDC 16, GDC 38, and GDC 50 remain satisfied at MUR-PU conditions. Therefore, the NRC staff concludes that the LAR is acceptable regarding Containment Systems.

3.1.3 Engineered Safety Features Heating, Ventilation and Air Conditioning Systems

3.1.3.1 Regulatory Evaluation

For Engineered Safety Features (ESF) Heating, Ventilation and Air Conditioning (HVAC) Systems, the NRC regulations specify criteria for control room habitability and post-accident fission product control and removal.

GDC 4, "*Environmental and dynamic effects design bases*," is described in Section 3.1.2.1 of this SE. The effects of the release of post-accident fission products and toxic gases would be a consideration when evaluating Farley with respect to compliance with GDC 4.

GDC 19, "*Control room*," states, in part, that, “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 [roentgen equivalent man] whole body, or its equivalent to any part of the body, for the duration of the accident .... "

GDC 41, "*Containment atmosphere cleanup*," states, in part, that, "Systems to control fission products ... which may be released into the reactor containment shall be provided as necessary to reduce ... the concentration and quality of fission products released to the environment following postulated accidents .... "

GDC 60, "*Control of releases of radioactive materials to the environment*," states, in part, that, "The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences [AOOs] .... " (AOOs are defined in 10 CFR Part 50, Appendix A).

GDC 61, "*Fuel storage and handling and radioactivity control*," states, in part, that, "systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions .... "

GDC 64, "*Monitoring radioactivity releases*," states, in part, that, "Means shall be provided for monitoring ... effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents."

10 CFR 50.67, “Accident source term.”

In its review, the NRC staff used specific criteria relevant to the evaluation of ESF HVAC Systems found in the Standard Review Plan (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."

3.1.3.2 Technical Evaluation

The NRC staff reviewed the impact of implementation of the MUR-PU on the containment ventilation systems, penetration room filtration system, containment pre-access filtration and purge systems, control room ventilation systems, auxiliary building ventilation systems, radwaste area, and turbine building ventilation systems.

The containment ventilation system is designed to provide containment atmosphere mixing and cooling. The penetration room filtration system is designed to limit release to the environment of radioisotopes which may leak into the penetration room under accident conditions. The containment pre-access filtration system is designed to reduce the airborne fission product activity in the containment atmosphere prior to personnel access at normal power conditions or before a plant outage. These systems are described in UFSAR Section 6.2.3.

In Section VI.1.F “Engineered safety features (ESF) heating, ventilation, and air conditioning systems,” Attachment 4 of the LAR, the licensee described that these systems were evaluated at the expected MUR-PU conditions and stated that these systems remain capable of performing their intended functions without modifications, and all AOR remain bounding.

The control room ventilation system is described in UFSAR Section 9.4. The system design provides a controlled environment for the comfort and safety of control room personnel and assures the operability of control room components during normal operating, anticipated operational transient, and design-basis accident conditions. The licensee evaluated the MUR-PU impact on the system and stated that since the mass and energy release into the containment analyses are bounded by current analyses, and the radiological consequences for postulated accidents are within design limits, the system will continue to perform its design function following the MUR-PU.

The licensee stated that, “The source terms for the MUR-PU are bounded by the current analysis. Therefore, the capability of the ventilation system to protect personnel will not be challenged.” The NRC staff finds that since the source term remains bound by the current analysis, MUR-PU has no impact on the ventilation system (with relations to alternate source term) and continues to meet 10 CFR 50.67 (See SER Section 3.3.1, “Radiological Dose Assessment”).

The auxiliary building ventilation system is described in UFSAR Section 9.4. The system is designed to maintain ventilation, permit personnel access, and control the concentration of airborne radioactive material in the auxiliary building. The licensee evaluated the impact of the MUR-PU on the system and identified, that the ambient temperature could increase, due to an increase in the spent fuel pool (SFP) heat load. The NRC staff evaluation of SFP provided in Section 3.1.4.2.3 of this SE concludes that the impact of the additional heat load to the pool analysis is negligible. Therefore, current ventilation system will continue to perform its design function following the implementation of the MUR-PU.

The LAR states that the radwaste area heating, ventilating and filtration system is described in UFSAR Section 9.4.3. The system is designed to control and direct all potentially contaminated air to the vent stack via pre-filter, HEPA [High Efficiency Particulate Air], and charcoal filters, the system is also designed to maintain a suitable environment for equipment and personnel. The licensee evaluated the impact of the MUR-PU and determined that the increase in the area temperature will be insignificant with respect to the heat load to the rooms. The NRC staff evaluated the description of the system and its design requirements and finds that the system remains capable of performing its function following the MUR-PU because the system has no safety-related function and it is not credited to mitigate any accident scenario.

The turbine building heating and cooling system is described in Farley UFSAR Section 9.4.4. The system designed to maintain ventilation, permit personnel access, and control the concentration of airborne radioactive material in the turbine area. The licensee indicated that the system was evaluated for the expected MUR-PU conditions and the evaluation found that the increase in heat load will not be significant, the system remains capable of performing its function following the MUR-PU.

3.1.3.3 NRC Staff Conclusion Regarding ESF HVAC

The NRC staff finds that the increase in heat loads in the ESF HVAC Systems, are minimal and bounded by the current analyses. Therefore, the NRC staff finds that the criteria identified in GDC 4, GDC 19, GDC 41, GDC 60, GDC 61 and GDC 64 remain satisfied at MUR-PU conditions. Therefore, the NRC staff concludes that the LAR is acceptable regarding ESF HVAC.

3.1.4 Plant Systems

3.1.4.1 Regulatory Evaluation

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-PU conditions. The NRC staff evaluated the LAR with respect to guidance in RIS 2002-03.

The NRC staff’s review in the area of plant systems covers the impact of the proposed MUR-PU on the nuclear steam supply system (NSSS) interface systems, containment systems, safety-related cooling water systems, SFP storage and cooling, and radioactive waste systems. The NRC staff’s review is based on the guidance in the SRP Chapter 3, "Design of Structures, Components, Equipment, and Systems"; Chapter 6, "Engineered Safety Features"; Chapter 9, "Auxiliary Systems"; Chapter 10, "Steam and Power Conversion System"; and Chapter 11, "Radioactive Waste Management"; and RIS 2002-03, Attachment 1, Sections II, III, and VI. The licensee evaluated the effect of the proposed MUR-PU on the plant systems in Attachment 4 of the LAR. The NRC staff reviewed the impact of the MUR-PU on the following major plant systems and interactions:

- NSSS interface systems
- safety-related cooling water systems
- SFP cooling analyses and systems
- radioactive waste systems
- flooding analyses
- high energy line breaks

The NRC staff conducted its review to verify that the licensee's analyses bound the proposed plant operation at the proposed MUR power level of 2821 MWt, and that the results would continue to meet the applicable acceptance criteria following implementation of the proposed MUR-PU.

3.1.4.2 Technical Evaluation

3.1.4.2.1 NSSS Interface systems

The nuclear steam supply system (NSSS) interface systems include the main steam supply system (MSSS), the condensate and FW system, and the auxiliary feedwater (AFW) system.

3.1.4.2.1.1 Main Steam Supply System

The MSSS is described in UFSAR Section 10.3. Section VI.1.A.i, Attachment 4 of the LAR, describes the evaluation of the MUR-PU impact on the MSSS.

The MSSS includes piping from the three steam generators (SGs), the turbine generator, the moisture separator reheaters, the steam jet air ejector system, the turbine shaft gland seals, the SG feedwater pump turbines, the turbine-driven auxiliary feedwater pump, and the turbine bypass system. The design bases of the MSSS is to provide steam flow requirements at turbine inlet design conditions, dissipate heat from RCS following a turbine and/or reactor trip, provide main steam system overpressure protection, and provide steam to main FW and AFW pumps and other equipment. The licensee stated that an evaluation of the MSSS piping at MUR-power uprate conditions was performed and found that the expected conditions (pressure and temperatures) remain bounded by the current piping design parameters.

The licensee also stated that the main steam isolation valves, which are designed to close to prevent a steam generator from blowing down due to an MSLB or to preclude overpressurization of containment due to reverse flow from the other main steam lines (if there is a break inside containment), were evaluated under MUR-PU conditions. The licensee identified that even though there is an expected higher steam flow through the system, it has no impact in the valve capacity to close within the required time.

Based on the discussion above, the NRC staff finds that the MSSS system remains capable of performing its function following the MUR-PU.

3.1.4.2.1.2 Condensate and Feedwater System

The condensate and feedwater systems are described in UFSAR Section 10.4.7. Section VI.1.A.iii, Attachment 4 of the LAR, describes the evaluation of the MUR-PU impact on the condensate and feedwater system.

Condensate pumps deliver water to the feedwater pumps' suction header, the feedwater pumps deliver the water to the SGs. The licensee stated that the MUR-PU will result in an increased condensate flow of approximately 1.8 percent, and the feedwater flow of approximately 2.0 percent. The LAR indicates that the feedwater system temperature is expected to increase 2.6ºF, from 443.4ºF to 446ºF (less than a 1 percent). The licensee evaluated the expected conditions following the MUR-PU and identified no significant increase on piping pressures or temperatures.

The licensee's evaluation, included the systems controls, alarms, setpoints, and components. The evaluation found that with minor adjustments, the system is capable to perform its intended safety function without exceeding component design specifications or causing any adverse conditions that would challenge system operability.

The NRC staff evaluated the justification provided in the LAR and finds that the systems and components credited to provide containment isolation capability were evaluated (in the AOR) at 102 percent of the current licensed core power, which bounds the MUR-PU. The condensate and feedwater systems operating parameters will not significantly change at MUR-PU condition, therefore, the NRC staff finds that the condensate and feedwater systems remain capable of performing its function following the MUR-PU.

3.1.4.2.1.3 Auxiliary Feedwater System

The AFW system is described in the UFSAR Section 6.5. Section VI.1.A.iv, Attachment 4 of the LAR, describes the evaluation of the MUR-PU impact on the AFW system.

The AFW system provides FW to the SGs in the event of the loss of main FW. The AFW AOR is based on 102 percent RTP. The licensee stated that the analyzed core power level remains conservative and bounds the MUR-PU (2821 MWt). Further, the licensee stated that AFW system maximum operating temperature and pressure remain essentially unchanged. There are no changes in AFW system minimum flow requirements, and no proposed changes to AFW pump design or operation. There is no design change required for this system to operate at 2821 MWt. Therefore, the NRC staff finds that the AFW system remains capable of performing its safety function under the proposed MUR-PU.

3.1.4.2.1.4 Main Turbine-Generator

The turbine-generator is described in the UFSAR Sections 10.1 and 10.2. Section VI.1.A.ii, Attachment 4 of the LAR, describes the evaluation of the MUR-PU impact on the main turbine-generator.

A portion of NRC staff's evaluation can be found in Section 3.2.3.2.2, "Power Block Equipment," of this SE. The NRC staff evaluation below focuses on the turbine missile analysis.
To support the MUR-PU, the licensee stated that the high-pressure blading design would be replaced with modern designs and the existing Farley, Units 1 and 2, high pressure turbine internals would require modernization to increase valve wide-open steam flow capacity, and to recover turbine throttle flow margin. The licensee evaluated the effects of the proposed MUR-PU on the turbine missile analysis. The licensee's assessment of the high-pressure turbine replacement concluded that the MUR-PU would not require the turbine missile analysis to be updated because the turbine missile analysis is only for the low-pressure turbine, while the high-pressure turbine has large design margin and a postulated ruptured high-pressure rotor would be contained within the high strength, high pressure turbine casing.

However, since the components for the turbine are being modified (including modernization to increase valve wide-open steam flow capacity, and to recover throttle flow margin) the effects of these modifications can change the steam flow, pressure, temperature and moisture content of the steam seen by the low-pressure turbine. Any effects to the low-pressure turbine, such as increased moisture content, temperature or flow can affect the degradation mechanism and stresses on the low-pressure turbine rotor and impact the turbine missile analysis. Therefore, the NRC staff requested the licensee evaluated any potential impact to the low-pressure turbine missile analysis.

By letter dated April 22, 2020, the licensee submitted a response to an NRC staff request for additional information that clarified that during uprate conditions, the low-pressure turbine inlet temperatures increase only slightly from their baseline conditions (+1.5°F for Unit 1 and +0.3°F for Unit 2). The low-pressure turbine inlet steam temperatures for both Farley units after the MUR-PU were evaluated with respect to the current turbine missile analysis and concluded that the temperatures are bounded by the analysis. In addition, the moisture content values at the low-pressure turbine inlet on both Farley units are unchanged, while the steam conditions at the low-pressure turbines due to the high-pressure turbine upgrade and MUR-PU are small and remain within the bounding assumptions used in the missile analysis. Therefore, the licensee confirmed that the AOR regarding the missile analysis results are not affected by the MUR-PU. The NRC staff finds that the licensee has accounted for the moisture content, temperature and flow rate changes due to the MUR-PU implementation, and that the current turbine missile AOR remains valid for the MUR-PU to ensure adequate protection from turbine missiles.

3.1.4.2.1.5 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, finds that they are acceptable. The licensee indicated that there is no adverse impact on the NSSS interface systems from the proposed MUR-PU because there is sufficient operating margin to produce an additional 1.7 percent RTP. The NRC staff determines that an MUR-PU will not challenge the NSSS interface systems. Therefore, the NRC staff concludes that the LAR is acceptable regarding NSSS interface systems.

3.1.4.2.2 Safety-Related Cooling Water Systems

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

3.1.4.2.2.1 Component Cooling System

The component cooling system is described in Section 9.2.2 of the UFSAR. Section VI.1.C.i, Attachment 4 of the LAR, describes the evaluation of the MUR-PU impact performed on the safety-related component cooling water system (CCWS).

The CCWS transfers heat from reactor auxiliaries to SWS under normal and accident conditions. The licensee stated that, “The CCWS was evaluated to confirm that the heat removal capabilities are sufficient to satisfy the MUR-PU heat removal requirements during normal plant operation, plant cooldown, and accident cooldown conditions. The analysis confirms that, at MUR-PU conditions, normal plant operation and required cooldown time continue to be met.”

The NRC staff reviewed the information provided in the LAR regarding the CCWS and finds that it will continue to meet its safety functions upon implementation of the MUR-PU.

3.1.4.2.2.2 Nuclear Service Water System

The SWS is described in Section 9.2.1 of the UFSAR. Section VI.1.C.ii, Attachment 4 of the LAR, describes the evaluation of the MUR-PU impact on the SWS.

The SWS provides cooling water for heat removal from the reactor plant auxiliary systems during all phases of station operation. Each unit has two redundant trains of equipment. The licensee found that the MUR-PU has no impact on the system or any of its major components, and thus will have no impact on the system safety functions. The licensee stated that the existing AOR bounds operation under MUR-PU conditions.

The NRC staff reviewed the licensee's analysis of the impact of the MUR on the SWS and finds that the SWS will continue to meet its safety functions upon implementation of the MUR-PU.

3.1.4.2.2.3 Ultimate Heat Sink

The UHS is described in Section 9.2.5 of the Farley UFSAR. Section VI.1.C.iii, Attachment 4 of the LAR, describes the evaluation of the MUR-PU impact on the UHS.

The service water and circulating water systems discharge heat to the Chattahoochee River during normal operation, shutdown, and accident conditions. However, Farley credits the storage pond as the UHS, which is designed with adequate water such that the system is capable of providing sufficient cooling for at least 30 days to permit simultaneous safe shutdown and cooldown of both nuclear reactor units, without any makeup from the Chattahoochee River. The licensee evaluated the expected increase heat loads from MUR-PU and identified that the increased heat load remains bounded by heat loads used in the original UHS analysis, which contained additional margin.

The NRC staff reviewed the information in the LAR and finds that the UHS will continue to meet its safety function upon implementation of the MUR-PU.

3.1.4.2.2.4 Residual Heat Removal System (RHRS)

The RHRS is described in Section 5.5.7 of the Farley UFSAR. Section VI.1.C.iv, Attachment 4 of the LAR, describes the evaluation of the MUR-PU impact on the RHRS.

One train of equipment can cool the RCS from 350°F to 200°F on RHRS operation within the associated TS limit of 37 hours after shutdown by stopping the operating reactor coolant pump (RCP) when RCS temperature reaches 220°F. The calculated cooldown time for this alignment increased from 34.5 hours (current AOR) to 34.6 hours (MUR-PU). The acceptance criterion for single-train cooldown of 37 hours (TS 3.0.3) continues to be met at MUR-PU conditions. Therefore, the NRC staff finds that the RHRS will continue to meet its safety function upon implementation of the MUR-PU.

3.1.4.2.2.5 NRC Staff Conclusion Regarding Safety-Related Cooling Water Systems

The NRC staff reviewed the licensee's evaluation of safety-related cooling water systems. The licensee’s analyses showed that these systems have sufficient margin such that the AOR remains bounding. Therefore, the NRC staff concludes that there is reasonable assurance that the systems will continue to meet their safety functions after implementation of the MUR-PU.

3.1.4.2.3 SFP Storage and Cooling

The SFP cooling and cleanup system is in Section 9.1.3 of the UFSAR. Section VI.1.D, Attachment 4 of the LAR, describes the evaluation of the MUR-PU impact on the SFP storage and cooling systems.

The system is designed to remove decay heat SFP water and to maintain clarity (visual) and purity of the SFP water, the transfer canal water, and the refueling water. The primary impact of an MUR-PU would be an increase of the decay heat of the fuel recently discharged from the core. The licensee stated that the current analysis for SFP heat loads was performed at 102 percent rated power which bounds the MUR-PU.

The NRC staff finds that the proposed MUR-PU will not result in a significant change to the operation of the SFP storage and cooling system. Therefore, the NRC staff finds that the SFP storage and cooling system will not be impacted by implementation of the MUR-PU.

3.1.4.2.4 Radioactive Waste Management Systems

The Radioactive Waste Systems (Liquid Waste, Gaseous Waste, and Solid Waste) are described in UFSAR Chapter 11. Section VI.1.E, Attachment 4 of the LAR, describes the evaluation of the MUR-PU impact on the radioactive waste systems.

These systems are designed to provide means to sample, collect, process, store/hold, re-use or release low-level effluents generated during normal operation. The licensee stated that the MUR-PU could potentially increase the concentration of radioactive nuclides by 1.7 percent, but it is not expected to increase the volumes or frequency of effluents. The licensee stated that current systems and procedures are adequate to support the MUR-PU without requiring modifications.

The licensee's analysis showed that the design of the radioactive waste systems at the increased power level is bounded by existing AOR, therefore, the NRC staff finds that the radioactive waste systems would perform acceptably after implementation of the MUR-PU.

3.1.4.2.5 Flooding Analyses

Flooding Analysis is described in UFSAR Chapter 3.11. Section II.1.D.iii, Item 30, Attachment 4 of the LAR, describes the evaluation of the MUR-PU impact on flooding.

The licensee completed an engineering evaluation of the potential impact of the proposed MUR-PU on internal flooding in areas that include engineered safety features, reactor protection systems (RPSs), and other safety-related systems. The licensee indicates that the MUR-PU will not increase the volume of the RCS, accumulators, and the refueling water storage tanks (RWSTs). Also, the long-term LOCA M&E releases remain applicable for MUR-power uprate conditions. The licensee has identified no significant increases in fluid volumes in storage tanks or maximum flow rates through fluid system piping.

Outside containment, the licensee identified that there is a minor estimated increase in the feedwater temperature that results in a reduction in the volumetric break flow rate (due to an increase in the volume of water that turns to steam). Consequently, the results of the current flooding analysis documented in UFSAR remain bounding for the MUR-PU conditions. The licensee determined that the existing flood analyses remain valid and are not affected by operating at the increased power level described in the LAR.

The licensee’s analysis showed that the effects on internal flooding at the increased power level described in the LAR are bounded by existing plant analyses. Therefore, the NRC staff concludes that the internal flooding analyses remain acceptable following implementation of the MUR-PU.

3.1.4.2.6 High Energy Line Breaks (HELB) Inside and Outside Containment

The containment pressure and temperature analysis associated with a HELB inside containment is described in Farley UFSAR Chapter 6.2.

The licensee evaluated the consequences of a HELB inside the containment building and the turbine building with respect to impact on safety-related equipment. 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 degrees Fahrenheit (ºF). High-energy pipe cracks are postulated in piping for which either the operating pressure exceeds 275 psig or the operating temperature exceeds 200 ºF.

Section II.1.D.iii.28, Attachment 4 of the LAR, describes the evaluation of the MUR-PU impact on HELB inside containment building. The LAR states that the methodology for the most limiting mass and energy release evaluation AOR assumes a core thermal power of 2830.5 MWt, which is the licensed core power of 2775 MWt with an additional 2 percent calorimetric uncertainty applied. This bounds the MUR-PU conditions.

LAR Section II.1.D.iii, Item 33, “High Energy Line Breaks Outside Containment,” indicates that the AOR for the break outside containment assumes a 2 percent calorimetric uncertainty added to the 2785 MWt NSSS power, which calculates as 2841 MWt NSSS that bounds the MUR-PU conditions. In addition, the licensee evaluated the expected conditions following the MUR-PU and did not identify any new high energy pipping segment. The licensee stated that no new lines are added, no break locations have changed, and no change is made to the assumed blowdown from any postulated break. The licensee determined that there is, therefore, no impact on the HELB analysis that was performed for Farley.

The licensee analysis showed that the effects from a HELB at the increased power level described in the LAR are bounded by existing plant analyses. Therefore, the NRC staff concludes that the HELB analysis remains acceptable.

3.1.4.3 NRC Staff Conclusion Regarding Plant Systems Impacts

The NRC staff reviewed the licensee's safety analyses of the impact of implementation of the proposed MUR-PU on the major plant systems. The NRC staff finds 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-PU. Therefore, the NRC staff concludes that the LAR is acceptable regarding the impact of changes to plant systems.

3.1.5 Accident Analysis (MUR)

In addition to implementing the MUR-PU, the licensee is also implementing NRC-approved Westinghouse Report, WCAP-17642-P-A, Revision 1, “Westinghouse Performance and Design Model (PAD5)” for the non-LOCA transient and accidents (Reference 24). PAD5 is a code that is used to evaluate fuel rod performance and is used as input to many of the UFSAR Chapter 15 event evaluations.

3.1.5.1 Regulatory Evaluation

For the review of the proposed MUR-PU and PAD5 implementation for Farley, the NRC staff used the below:

GDC-10, “*Reactor design*,” states that the reactor core and associated coolant, control, and protection systems 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*,” states that the reactor coolant system and associated auxiliary, control, and protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

GDC-25, “*Protection system requirements for reactivity control malfunctions*,” which requires that the protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

GDC 26, “*Reactivity control system redundancy and capability*,” as it relates to the capability of control rods to reliably control reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences [AOOs], and with appropriate margin for malfunctions like stuck rods, specified acceptable fuel design limits are not exceeded.

GDC 27, “*Combined reactivity control systems capability*,” as it relates to the reactivity control systems which shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

GDC 28, “*Reactivity limits,*” as it relates to the reactivity control systems which shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core.

Requirements in 10 CFR 50.62, “Requirements for reduction of risk from anticipated transient without scram (ATWS) events for light-water-cooled nuclear power plants,” as it relates to the acceptable reduction of risk from ATWS events via inclusion of prescribed design features and demonstration of their adequacy.

Requirements in 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors.”

Requirements in 10 CFR 100.11 and 10 CFR 50.67 establish radiation dose limits for individuals at the boundary of the exclusion area and at the outer boundary of the low population zone. The fission product inventory released from all failed fuel rods is an input to the radiological evaluation.

The 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-PU LARs.

NUREG-0800, “Standard Review Plan (SRP),” Chapter 15, provides review guidance for transient and accident analysis.

3.1.5.2 Technical Evaluation

3.1.5.2.1 Evaluation of Accident and Transient Analysis

In its application, the licensee evaluated existing analyses for the proposed MUR-PU operating conditions with reduced uncertainty in one of three different ways:

- For analyses that assume steady-state plant operation with a core power of 2775 MWt, there is a 2 percent margin for power measurement uncertainty at the RTP (2831 MWt). These analyses also bound plant operation at the proposed MUR-PU power level of 2821 MWt, with an operating margin of 0.34.

- For analyses that are not bounded by increased core power level for the MUR, the licensee reanalyzed the event.

- Zero-power transients were not reanalyzed for the MUR-PU because the increased power would not have an impact on the results.

The NRC staff notes that some of the events in UFSAR Chapter 15 analysis would typically not require reanalysis due to the MUR-PU. However, for some events, the licensee reanalyzed for PAD5 implementation.

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 AOR. For these analyses, the NRC staff determined that existing analyses will continue to bound plant operation after implementation of the proposed MUR-PU. Thus, the NRC staff finds that these analyses are acceptable without a detailed review.

Table 3-1 below summarizes those areas of the accident and transient analyses that received a detailed NRC staff review, consistent with the guidance of RIS 2002-03. For these events, the NRC staff reviewed the submittals to ensure that the same analysis methods were used as in the UFSAR (except for PAD5) and the event would continue to meet the applicable acceptance criteria. The applicable acceptance criteria for the events is listed in Table 3-2 of this SE.

TABLE 3-1 - Evaluation of Accident and Transient Analysis for MUR-PU

Transient/Accident	Analytic Power Level Used (Percent RTP)	Review Comments
Uncontrolled RCCA [Rod Cluster Control Assembly] Bank Withdrawal at Power	102 (DNB)	Acceptable
Uncontrolled Boron Dilution	102	Acceptable
Partial Loss of Forced Reactor Coolant Flow	102	Acceptable
Loss of External Electrical Load and/or Turbine Trip	102	Acceptable
Loss of Normal Feedwater	102	Acceptable
Loss of All AC Power to the Station Auxiliaries	102	Acceptable
Excessive Heat Removal Due to Feedwater System Malfunctions	102	Acceptable
Accidental Depressurization of the RCS	102	Acceptable
Major Secondary System Pipe Rupture	102	Acceptable
Single Reactor Coolant Pump Locked Rotor	102	Acceptable
Rupture of a Control Rod Drive Mechanism (CRDM) Housing (RCCA Ejection)	102	Acceptable
Anticipated Transient without Scram (ATWS)	102	Acceptable

NRC Staff Conclusion Regarding Evaluation of Accident and Transient Analysis

The NRC staff reviewed the current accident and transient analyses discussed in the licensee's submittals. For events which the licensee stated were based on operation at 102 percent rated thermal power (2831 MWt), or initial power conditions were at 0 percent, the NRC staff confirmed the values in the Farley UFSAR. For events that were reanalyzed for the power uprate (listed in Table 3-1), the staff confirmed that the same analysis methods were used as in the UFSAR and that the licensee met the acceptance criteria listed in Table 3-2 of this SE. Therefore, the NRC staff finds that, for the MUR-PU, the Chapter 15 events meet all the applicable acceptance criteria.

3.1.5.2.2 Non-LOCA PAD5 Implementation Evaluation

In addition to implementing the MUR-PU, the licensee is also implementing NRC-approved Westinghouse Report, WCAP-17642-P-A, Revision 1, "Westinghouse Performance and Design Model (PAD5)" for the non-LOCA transient and accidents. For events that were re-analyzed as part of the MUR-PU, PAD5 was used to generate fuel rod data used in the transient and accident analysis methods. For transients and accidents that were not necessary to be reanalyzed as part of the MUR-PU, the licensee evaluated the AOR to determine if it was necessary to re-analyze the event with inputs from PAD5.

The licensee addressed the five limitations and conditions in WCAP-17642-P-A, Revision 1, in Attachment 4, Section III.1, of the LAR. The NRC staff finds the licensee has met the applicable limitations and conditions in WCAP-17642-P-A, Revision 1.

This section of the SE contains a summary of the NRC staff's review of PAD5 implementation. The acceptance criteria for all the non-LOCA accident analyses are described in each of their respective sections in Attachment 4, Sections II and III, of the LAR. The NRC staff reviewed the information provided in the application to determine if the non-LOCA analyses that are reanalyzed as part of this application meet the acceptance criteria listed in the UFSAR Chapter 15 and meet the guidance in NUREG-0800 related to the specific non-LOCA analyses.

The following table contains a list of all the non-LOCA transient events that were considered by the licensee for PAD5 implementation including cross References to the applicable UFSAR sections, the acceptance criteria used, and associated regulations. The table also indicates if the event was reanalyzed or evaluated for PAD5 implementation. If an event was reanalyzed, the computer codes used to simulate the event were rerun with input from the PAD5 code and the results were compared quantitatively to the acceptance criteria. If the event was evaluated, a qualitative assessment was performed to determine if using PAD5 input would have an impact on the AOR results.

TABLE 3-2 - Evaluation of Accident and Transient Analysis for PAD5 Implementation

Transient Event	UFSAR Section	Reanalyzed or Evaluated?	Acceptance Criteria Summarized from Attachment 4, LAR	Associated Regulation
Uncontrolled RCCA Bank Withdrawal from a Subcritical Condition	15.2.1	Evaluated	Calculated Departure from Nucleate Boiling Ratio (DNBR) Above Design Basis DNBR No fuel melt	GDC 10 GDC 25
Uncontrolled RCCA Bank Withdrawal at Power	15.2.2	Reanalyzed	Calculated DNBR Above Design Basis DNBR No fuel melt RCS Pressure Below 110% design value	GDC 10 GDC 15 GDC 25
RCCA Misalignment	15.2.3	Evaluated	Calculated DNBR Above Design Basis DNBR No fuel melt	GDC 10 GDC 25
Uncontrolled Boron Dilution	15.2.4	Reanalyzed	Ensure no loss of shutdown margin	GDC 10 GDC 26
Partial Loss of Forced Reactor Coolant Flow	15.2.5	Reanalyzed	Calculated DNBR Above Design Basis DNBR	GDC 10 GDC 26
Startup of an Inactive Reactor Coolant Loop	15.2.6	Evaluated	Calculated DNBR Above Design Basis DNBR	GDC 10 GDC 26
Loss of External Electrical Load and/or Turbine Trip	15.2.7	Reanalyzed	Calculated DNBR Above Design Basis DNBR RCS and Main Steam System (MSS) Pressure Below 110% design value	GDC 10 GDC 15 GDC 26
Loss of Normal Feedwater	15.2.8	Reanalyzed	Calculated DNBR Above Design Basis DNBR RCS and Main Steam System (MSS) Pressure Below 110% design value	GDC 10 GDC 15 GDC 26
Loss of All AC Power to the Station Auxiliaries	15.2.9	Evaluated	Calculated DNBR Above Design Basis DNBR RCS and Main Steam System (MSS) Pressure Below 110% design value	GDC 10 GDC 15 GDC 26

Transient Event	UFSAR Section	Reanalyzed or Evaluated?	Acceptance Criteria Summarized from Attachment 4, LAR	Associated Regulation
Excessive Heat Removal Due to Feedwater System Malfunctions	15.2.10	Reanalyzed	Calculated DNBR Above Design Basis DNBR No fuel melt	GDC 10 GDC 26
Excessive Load Increase Incident	15.2.11	Evaluated	Calculated DNBR Above Design Basis DNBR	GDC 10 GDC 26
Accidental Depressurization of the RCS	15.2.12	Reanalyzed	Calculated DNBR Above Design Basis DNBR	GDC 10 GDC 26
Accidental Depressurization of the Main Steam System	15.2.13	Evaluated	Calculated DNBR Above Design Basis DNBR No fuel melt	GDC 15 GDC 27 10 CFR 100.11
Inadvertent Operation of ECCS During Power Operation	15.2.14	Evaluated	Calculated DNBR Above Design Basis DNBR Pressurizer does not overfill	GDC 10 GDC 15 GDC 26
Complete Loss of Forced Reactor Coolant Flow	15.3.4	Reanalyzed	Calculated DNBR Above Design Basis DNBR	GDC 10 GDC 15 GDC 26
Single RCCA Withdrawal at Full Power	15.3.6	Evaluated	Calculated percent of rods undergoing DNB below limit	GDC 10 GDC 26
Major Secondary System Pipe Rupture (main steam line break)	15.4.2	Evaluated	Calculated DNBR Above Design Basis DNBR No fuel melt	GDC 15 GDC 27 10 CFR 100.11
Major Secondary System Pipe Rupture (feedwater line break)	15.4.4	Reanalyzed	Coolable core geometry RCS and Main Steam System (MSS) Pressure Below 110% design value	GDC 15 GDC 27 10 CFR 100.11

Transient Event	UFSAR Section	Reanalyzed or Evaluated?	Acceptance Criteria Summarized from Attachment 4, LAR	Associated Regulation
Single Reactor Coolant Pump Locked Rotor	15.4.4	Reanalyzed	Fuel cladding damage, including melting, must be prevented. RCS and Main Steam System (MSS) Pressure Below 110% design value The total number of rods-in-DNB calculated for the associated dose analysis is less than 20% of the core.	GDC 27 10 CFR 100.11
Rupture of a Control Rod Drive Mechanism (CRDM) Housing (RCCA Ejection)	15.4.6	Reanalyzed	Average fuel pellet enthalpy at hot spot requirements must be met RCS pressure below faulted condition stress limits Fuel melt less than innermost 10% of the fuel pellet volume at the hot spot	10 CFR 100.11 10 CFR 50.67 GDC 28
Anticipated Transients Without Scram (ATWS)	15.5	Reanalyzed	Calculated peak RCS pressure remains less than the ASME Service Level C pressure limit of 3200 psig	10 CFR 50.62

For the events that were reanalyzed, the NRC staff compared the licensee's analytical methods with the methods currently in the UFSAR, reviewed the analysis description for input changes, compared acceptance criteria from the UFSAR to the reanalysis acceptance criteria, and compared the reported results to the acceptance criteria. For all the events reanalyzed, the NRC staff determined the following:

1. The same methods were used in the UFSAR and reanalysis
2. No significant changes to the analysis inputs (besides changes for the MUR and PAD5) implementation were identified
3. The same acceptance criteria used in the UFSAR was used in the reanalysis except for the fuel melting limit (discussed below)
4. All the analysis results met the acceptance criteria

The only major change to the acceptance criteria used in the UFSAR and that in the application is the fuel melting limit. PAD5 implemented an updated fuel melting limit as part of the methodology. This fuel melting limit is an empirically based limit and has been approved by the

NRC staff with the approval of PAD5. Therefore, the NRC staff finds that the fuel melting limit used in the application is acceptable, and the NRC staff confirmed that, where there was a reanalysis that uses fuel melting as an acceptance criterion, the updated fuel melting limit was used.

For events that were evaluated, the NRC staff reviewed the information to determine if the licensee demonstrated sufficiently that the PAD5 implementation would have a negligible impact on the results. For these events, the staff used engineering judgement to determine that the qualitative analysis performed by the licensee provided reasonable assurance that PAD5 implementation would have a negligible impact on the AOR results. For example, for the hot zero power steam line break event, the licensee stated that this analysis uses the minimum fuel rod temperatures and is not adversely affected by PAD5. The NRC staff determined that, as a result of implementing fuel thermal conductivity degradation into the PAD5 methodology, predicted fuel temperatures are higher with increasing burnups and will not impact the AOR results.

NRC Staff Conclusion Regarding Non-LOCA PAD5 Implementation

The NRC staff reviewed the implementation of PAD5 for the non-LOCA analysis. The NRC staff finds that the analyses and evaluations performed by the licensee used appropriate methods and sufficiently demonstrated that the events would meet the applicable acceptance criteria listed in Table 3-2 of this SE.

3.1.5.2.3 Fuel Evaluations

This section covers the transient and accident analyses that are included in the plant's UFSAR and other analyses that are required to be performed by the licensee to support plant licensing. Section II.1.D.iii, Item 34 and Section III.1, Item 34, Attachment 4 of the LAR, describes the fuel evaluation to support the MUR-PU.

Fuel Rod Design

Fuel rod design analyses (FRD) were performed in support of MUR-PU for Farley units to a maximum core power level of 2831 MWt at uprated core conditions for the entire irradiation history of the fuel. The Farley, Units 1 and 2, currently are fueled with Westinghouse VANTAGE+ Optimized Fuel Assembly (OFA) design with both ZIRLO and Optimized ZIRLO High Performance fuel cladding material. The licensee stated that FRD analyses were performed using NRC-approved models, methods, and criteria to guarantee that all FRD criteria are satisfied for the representative uprated neutronic models. Fuel parameters, such as, fuel temperatures, rod internal pressure (RIP), power-to-melt, and core stored energy were calculated to support the MUR-PU conditions to bound cycle-specific operation, using the NRC approved PAD5 fuel performance model. FRD analyses were performed on a cycle-specific basis using the cycle specific plant conditions as well as the fuel duty for each fuel regions in the core for specific cycle. The results of these analyses have shown that all FRD criteria were met at the MUR-PU power level.

Core Thermal Hydraulics

There is no fuel design change associated with the Farley MUR-PU. Core thermal-hydraulic (T/H) analyses were performed to support the Farley MUR-PU on a full core of 17x17

VANTAGE+ fuel assemblies for the uprated Farley core designs. The T/H analysis shows prevention of departure from nucleate boiling (DNB) on the limiting fuel rod with 95 percent probability at a 95 percent confidence level (95/95). The T/H analysis assumed a nominal power level of 2823 MWt that represents a 1.7 percent increase to the current Farley core power. DNB calculations are performed on a statistical basis as well as using analytical method. Appropriate plant parameter values for coolant flow, RCS temperature, and bypass flow have been used in the DNB analyses calculations. As stated in Section III.1.34, Attachment 4 of the LAR, for the T/H analyses design methods, the following deviation from UFSAR have been implemented for MUR-PU:

- NRC-approved VIPRE-W subchannel analysis code, WCAP-14565-P-A, Revision 0, (Reference 25) is used instead of THINC-IV subchannel analysis code and the FCTRAN code for DNB ratio (DNBR) calculations.

- The NRC-approved W-3 alternative correlation (Reference 26) have been used in place of the W-3 correlation as the secondary DNB correlations in cases where the DNB primary correlations are not applicable.

Implementation of the VIPRE code and the W-3 alternative DNB correlations have been accomplished by satisfying the SER conditions for the respective code and correlations. The licensee addressed the NRC SER Limitations and Conditions for VIPRE and W-3 Alternative DNB Correlations in Section III.1.34, Attachment 4 of the LAR, summarized below:

WCAP-14565-P-A

For the VIPRE code, there are four conditions in its SER which were considered in the licensee's safety analysis for Farley MUR-PU. Condition 1 requires the selection of appropriate critical heat flux (CHF) correlations, DNBR limit, engineered hot channel factors for enthalpy rise, and other fuel dependent parameters for a specific plant application must be justified with each submittal. Since there is no fuel change associated with Farley MUR-PU, the plant-specific hot channel factors and other fuel-dependent parameters in the DNBR analysis for the VANTGAE+ fuel for the Farley MUR-PU conditions are unchanged from currently approved values.

The VIPRE SER Condition 2 requires that inputs in to VIPRE, such as, core inlet coolant flow and enthalpy, core average power, power shape, and nuclear peaking factors should be justified as conservative for each use of the VIPRE code. The licensee has used the VIPRE code for VANTAGE+ fuel at MUR conditions with inputs and boundary conditions generated from NRC-approved codes and analyses. Reactor power used in the analyses reflects the 1.7 percent increase in nominal core power to reflect the MUR-PU. All other boundary conditions and inputs are unchanged from the conservative values that were previously justified for the current operating license.

The VIPRE SER Condition 3 requires that all requirements be met for CHF correlations used in the DNBR calculations for the VANTAGE+ fuel design. Licensee has shown that all the correlations including the ABB-NV DNBR limit and WLOP DNBR limit were previously approved for use with VIPRE code.

Condition 4 requires that appropriate justification be submitted with each usage of VIPRE in the post-CHF region to guarantee that conservative results are obtained. For Farley MUR PU safety analysis, the use of VIPRE in the post-critical heat flux region is limited to peak clad temperature (PCT) for the locked rotor transient. Licensee has implemented conservative assumptions, such as, DNB assumed to occur at the beginning of the transient, film boiling calculations were done using the Bishop-Sandberg-Tong correlation, and the Baker-Just correlation used to account for heat generation in cladding due to Zr-water reaction. Conservative results were assured with inputs such as:

- Fuel rod input was based on the maximum fuel temperature at the given power.
- The hot spot power factor was equal to or greater than the design linear heat rate.
- Uncertainties were applied to the initial operating conditions in the limiting direction.

WCAP-11397-P-A

The seven conditions identified in the safety evaluation for WCAP-11397-P-A, "Revised Thermal Design Procedure" (RTDP) (Reference 27), were addressed by the licensee. The NRC staff reviewed the licensee's conformance with the conditions with respect to their application for MUR-PU and finds that the licensee has satisfied the conditions and limitations for the RTDP.

WCAP-14565-P-A, Addendum 2-P-A

Condition 1 for this code (WCAP-14565-P-A, Addendum 2-P-A) (Reference 25) requires the range of the ABB-NV and WLOP correlations as presented in the approved technical report (TR). Licensee has performed DNB analysis based on conditions specified in the approved TR.

Condition 2 requires that the ABB-NV correlation and the WLOP correlation must use the same Fe factor for power shape correction as used in the primary DNB correlation for a specific fuel design. Licensee performed the DNB analyses at MUR-PU conditions based on ABB-NV and WLOP correlations using the Fc factor for power shape that was applied for WRB-2 correlation for the VANTAGE+ fuel in Farley cores.

Condition 3 requires that the selection of appropriate DNB correlation, DNBR limit, engineering hot channel factors for enthalpy rise, and other fuel-dependent parameters shall be justified for each application of each correlation on a plant specific basis. There is no fuel design change associated with the Farley MUR-PU. The plant-specific hot channel factors and other fuel-dependent parameters in the DNBR calculations for the VANTAGE+ fuel in Farley at MUR-PU conditions are unchanged from the currently approved values.

Condition 4 requires that the ABB-NV correlation for Westinghouse pressurized water reactor (PWR) applications and the WLOP correlation must be used in conjunction with the Westinghouse version of the VI PRE-01 (VIPRE) code subject to VIPRE modeling specifications. Westinghouse version of VIPRE code was qualified and approved with ABB-NV and WLOP correlations for DNB analyses of the VANTAGE+ used in Farley at MUR-PU conditions.

Subchannel Analysis Code

The DNBR calculations were performed with the VIPRE code for the DNB-limited Chapter 15 events described above that are currently analyzed with the THINC subchannel analysis code. The DNBR calculations performed with the VIPRE code address the increased nominal power and the change in power measurement uncertainty associated with the MUR-PU.

DNB Methodology

The primary DNB correlation for VANTAGE+ fuel at MUR-PU conditions is WRB-2. For current DNB analysis, W-3 correlation is used to supplement the primary DNB correlation where the primary correlation is not applicable. With the RTDP, the uncertainties in plant operating parameters are considered statistically to obtain overall DNB uncertainty factors with 95 percent probability with 95 percent confidence that DNB will not occur on the most limiting fuel rod during normal operation, operational transients, or transient conditions arising from faults of moderate frequency.

Based on current RTDP design limit, the DNBR values for the VANTAGE+ fuel in the Farley units were based on the THINC-IV code and the WRB-2 correlation. The reduced power measurement uncertainty provides a benefit in the calculation of the RTDP design limit DNBR values. DNBR sensitivity factors required to develop the WRB-2 DNB uncertainty factors for the MUR-PU were based on the VIPRE code.

The ABB-NV correlation is applicable for the DNB analysis of the portion of the fuel rod below the first mixing vane grid. The design limit DNBR was developed based on RTDP using the ABB-NV correlation at the MUR-PU conditions. The DNBR sensitivity factors required to develop the ABB-NV DNB uncertainty factors for the MUR-PU were based on the VIPRE code. Sufficient DNBR margin is conservatively maintained in the safety analysis DNBR limits to offset the rod bow DNBR penalty and to provide flexibility in design and operation of the plant.

For cases where RTDP is not applicable, statistical methods using the standard thermal design procedure (STDP) continue to be used for DNB analyses. For the STDP, the initial condition uncertainties are accounted for deterministically by applying the uncertainties to the nominal conditions. The DNBR limit for STDP is the appropriate DNB correlation limit with consideration for applicable DNBR penalties.

Rod bow that occurs in the spans between the assembly grids reducing the spacing between adjacent fuel rods and reduces the margin to DNB. Rod bow must be accounted for in the DNB safety analysis of Condition I and Condition II events. For MUR-PU conditions at Farley, licensee has implemented rod bow DNBR penalty continues to be applied using the NRC-approved methodology in WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation," July 1979 (Reference 28) and (Reference 29).

Appropriate rod bow DNBR penalty for VANTAGE+ fuel design for all regions of the fuel assembly, grid span, mixing grid region, region below the first mixing vane and the grid spans that contained intermediate flow mixer (IFM) grid have been applied.

NRC Staff Conclusion on Fuel Evaluations

The NRC staff reviewed the licensee’s use of approved computer codes for fuel rod design, core thermal hydraulic analysis, sub-channel T/H analysis codes, DNB correlations and their limits, and DNB methodology. The NRC staff verified whether the licensee has met the limitations and conditions for each of the codes used in the above analyses and finds that the licensee has used the appropriate codes for the analyses and satisfied all limitations and conditions for the codes.

3.2 Engineering and Materials

3.2.1 Reactor Vessel Integrity and Reactor Vessel Internal and Core Support Structures

The NRC staff's review of reactor vessel (RV) integrity focuses on the potential impact of the MUR-PU 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-PU, 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."

The NRC staff also reviewed the potential effects of the MUR-PU on the service conditions for RV internal and core support structure components to verify that the MUR-PU will not have an adverse impact on the suitability of plant inspection programs for ensuring component functionality during the periods of extended operation.

3.2.1.1 Pressurized Thermal Shock

3.2.1.1.1 Regulatory Evaluation

Regulations in 10 CFR 50.61 requires PTS evaluations to ensure that adequate fracture toughness exists for RV beltline materials in PWRs to protect against RV failure during a PTS event. Fracture resistance of RV beltline materials during PTS events is evaluated by calculating the nil-ductility temperature (RT_{NDT}) for PTS, identified as RT_{PTS}. 10 CFR 50.61(a) defines RT_{PTS} as the RT_{NDT} evaluated for the "EOL [end-of-life] Fluence" for each of the RV beltline materials 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 RT_{PTS} accepted by the NRC for each RV beltline material. Regulation 10 CFR 50.61(c)(2) requires that RT_{PTS} 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 RT_{PTS} 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). Regulations in 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 demonstrating that their RT_{PTS} values are less than the PTS screening criteria at the expiration of the operating license.

The definition of the "RV beltline" is applicable to all ferritic RV materials with projected neutron fluence values greater than 1×10^{17} n/cm^2 (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 RIS 2014-11 (Reference 30).

The regulations in 10 CFR 50.61 require that licensees update their PTS evaluations whenever there is a significant change in operating conditions that affect the projected values of RT_{PTS}. Therefore, for power uprate amendment applications, the NRC staff's review must address the impact of the increase in RTP on the licensee's calculation of the RT_{PTS} values for all of the RV beltline materials.

3.2.1.1.2 Technical Evaluation

Licensee Evaluation

The licensee discussed the impact of the MUR-PU PTS evaluation in Section IV.1.C.i of Attachment 4 of the LAR. The licensee's AOR RT_{PTS} calculations were performed using the procedures required by 10 CFR 50.61(c) for 54 effective full power years (EFPY). To evaluate the effects of the MUR-PU on the RT_{PTS} evaluation, the licensee recalculated the RT_{PTS} values for the Farley Units 1 and Unit 2 RV beltline materials using 54 EFPY neutron fluence values that were generated by applying the new 3-D neutron fluence model, Westinghouse Electric Company Topical Report WCAP-18124-NP-A, Revision 0, "Fluence Determination with RAPTOR-M3G and FERRET" for MUR-PU conditions. The licensee provided a summary of the RT_{PTS} calculations associated with the MUR-PU for all Farley, Unit 1 and Unit 2, RV beltline materials in Tables IV.1.C-1 and IV.1.C-2, Attachment 4 of the LAR. The licensee identified that the RT_{PTS} values for all RV beltline materials will remain below the 10 CFR 50.61 PTS screening criteria using the projected neutron fluence values for MUR-PU conditions at 54 EFPY.

NRC Staff Evaluation

The AOR RT_{PTS} values for the Farley, Unit 1 and Unit 2, RV beltline materials are provided in the current P-T limits report (PTLR) for 54 EFPY (Reference 31). All AOR RT_{PTS} values are less than the applicable 10 CFR 50.61 PTS screening criteria (270 °F for plates, forgings, and axial weld materials and 300 °F for circumferential weld materials). For MUR-PU conditions, the NRC staff verified that the licensee correctly recalculated the RT_{PTS} values for all Farley, Unit 1 and Unit 2, RV beltline materials using the procedures required by 10 CFR 50.61(c) and incorporating 54 EFPY neutron fluence values that were generated using the new 3-D neutron fluence model described in WCAP-18124-NP-A, Revision 0. The NRC staff's technical review of the licensee's implementation of the 3-D neutron fluence model for generating the new neutron fluence values for MUR-PU conditions is discussed in Section 3.2.1.6 of this SE. The NRC staff verified that the RT_{PTS} values for all Farley, Unit 1 and Unit 2, RV beltline materials will continue to remain less than the applicable PTS screening criteria. Therefore, the NRC staff finds that the licensee's PTS evaluation for the Farley, Unit 1 and Unit 2, RV beltline materials will continue to meet the requirements of 10 CFR 50.61 for MUR-PU conditions.

3.2.1.2 Pressure-Temperature Limits

3.2.1.2.1 Regulatory Evaluation

Regulations in 10 CFR Part 50, Appendix G, include requirements for establishing reactor coolant system (RCS) P-T limits for protection of the reactor coolant pressure boundary (RCPB) against brittle fracture during normal operation, anticipated operational occurrences, and hydrostatic tests. P-T limits are not required for accident conditions. 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 (Reference 32). 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 (K_{IC}). As specified in Paragraph G-2210 of the ASME Code, Section XI, K_{IC} is an exponential function of the difference in metal temperature at the postulated crack tip and the Reference nil-ductility temperature (RT_{NDT}) for the ferritic RV material. Section IV.A of 10 CFR Part 50, Appendix G requires that the values of RT_{NDT} 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 RT_{NDT} values, thereby causing a rightward shift in the K_{IC} curve and a corresponding rightward shift in the P-T limit curve. The P-T limit curve shifts due to the effects neutron irradiation requires operation at lower pressures and/or higher temperatures (i.e., below and/or to the right of the curve) to maintain the required safety margins for protection of the RV against brittle fracture per 10 CFR Part 50, Appendix G. The NRC staff's recommended guidelines for calculating the effects of neutron radiation on the adjusted RT_{NDT} for RV beltline materials are specified in RG 1.99, Revision 2 (Reference 33). RIS 2014-11 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, in the development of P-T limits. The definition of the "RV beltline" is applicable to all ferritic RV materials with projected neutron fluence values greater than 1 x 10^{17} n/cm^2 (E > 1.0 MeV). The NRC staff's basis for this fluence threshold is provided in RIS 2014-11.

For power uprate applications, the NRC staff's review must address the impact of the increase in RTP on the adjusted RT_{NDT} values used for generating P-T limits for the RV beltline materials.

3.2.1.2.2 Technical Evaluation

Licensee Evaluation

The licensee discussed the impact of the MUR-PU on the Farley, Unit 1 and Unit 2, P-T limits in Section IV.1.C.iii of LAR Attachment 4. The current Farley, Unit 1 and Unit 2 PTLR, implements the AOR P-T limit curves for 54 EFPY. The AOR P-T limits were developed in accordance with 10 CFR Part 50, Appendix G; RG 1.99, Rev. 2; and the NRC-approved P-T limits methodology specified in WCAP-14040-A, Revision 4 (Reference 34). The 54 EFPY P-T limits specified in the PTLR were calculated using the adjusted RT_{NDT} values for the limiting RV beltline shell materials. The licensee evaluated the impact of the MUR-PU on the AOR P-T limits by recalculating the limiting adjusted RT_{NDT} values per RG 1.99, Rev. 2, using 54 EFPY neutron fluence values that were generated by applying the new 3-D neutron fluence model. Table IV.1.C-3, Attachment 4 of the LAR, shows that the adjusted RT_{NDT} values for the limiting RV beltline shell materials under MUR-PU conditions will exceed those used in the AOR P-T limits at 54 EFPY. To address MUR-PU conditions, the licensee proposed to maintain AOR P-T limits by reducing the EFPY term corresponding to the neutron fluence and adjusted RT_{NDT} values used in the existing analysis. The licensee determined that the MUR-PU will result in the neutron fluence used for the AOR adjusted RT_{NDT} and P-T limit calculations being reached at

51.9 EFPY and 52.1 EFPY for Farley, Unit 1 and Unit 2, respectively. Therefore, the licensee proposed to decrease the EFPY applicability term for the AOR P-T limits in the PTLR to these values.

The licensee also discussed the impact of the MUR-PU on its AOR evaluation of the effect of the RV nozzles on the bounding P-T limits for the Farley, Unit 1 and Unit 2, RVs. The licensee reported that its evaluation of the effects of the MUR-PU on the RV nozzle adjusted RT_{NDT} values demonstrates that the nozzle P-T limits will remain bounded by the P-T limits for the limiting RV beltline shell materials for Farley, Unit 1 and Unit 2.

In addition to its P-T limits evaluation, the licensee also discussed the impact of the MUR-PU on its low-temperature overpressure protection (LTOP) system requirements in Section IV.1.C.iv, Attachment 4 of the LAR. The LTOP system is required to automatically protect the RV against overpressurization transients during low temperature operations (i.e., when the limiting RV materials are with or below the ductile-to-brittle transition region). Per TS LCO 3.4.12, the LTOP system shall be operable at temperatures below the "LTOP system applicability temperature," which is specified in the PTLR. The licensee determined that the implementation of the MUR-PU will not affect the design basis transient characteristics and pressure relief capacity for the LTOP system during low temperature operations. Since MUR-PU conditions are addressed through a reduction to the EFPY applicability term for the AOR P-T limits in the PTLR, the licensee identified that the EFPY term for the AOR LTOP system applicability temperature should also be reduced consistent with the P-T limits.

NRC Staff Evaluation

The AOR P-T limits curves are established in the Farley, Unit 1 and Unit 2, PTLR for 54 EFPY. The contents of the PTLR are controlled administratively in accordance with TS 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)." As part of this MUR-PU amendment, TS 5.6.6 will be changed to incorporate a new Reference for the 3-D neutron fluence model, as discussed in the LAR and accompanying TS changes. The NRC staff's technical review of the licensee's implementation of the 3-D neutron fluence model for generating the new neutron fluence values for MUR-PU conditions is discussed in Section 3.2.1.6 of this SE. The NRC staff finds that the addition of WCAP-18124-NP-A, Revision 0, for the new 3-D fluence model to TS 5.6.6 is consistent with TS administrative control recommendations of Generic Letter 96-03 (Reference 35).

For the proposed MUR-PU conditions, the NRC staff verified that the licensee correctly recalculated 54 EFPY adjusted RT_{NDT} values, consistent with the methods in RG 1.99, Revision 2, for the limiting RV beltline shell materials and the RV inlet and outlet nozzles. The recalculated adjusted RT_{NDT} values incorporate new neutron fluence values generated using the 3-D neutron fluence model in WCAP-18124-NP-A, Revision 0. The NRC staff verified that the licensee's adjusted RT_{NDT} calculations for the MUR-PU demonstrate that the RV nozzle P-T limits will remain bounded by the P-T limits for the limiting RV beltline shell materials.

For a given EFPY operating term, an increase in the rate of neutron fluence accumulation (i.e., neutron flux) for power uprate conditions will result in an increase in the limiting adjusted RT_{NDT} values used for calculating P-T limits, thereby requiring a rightward shift in the P-T limits for that EFPY term. For power uprate applications, licensees frequently choose to maintain their AOR P-T limits for uprated conditions by recalculating the EFPY term corresponding to the neutron fluence and adjusted RT_{NDT} values used for calculating the P-T limits. By accurately reducing the EFPY term to account for the increase in neutron flux associated with the increase in RTP, the existing neutron fluence values, limiting adjusted RT_{NDT} values, and P-T limits can remain the same. The NRC staff verified that the licensee correctly addressed the impact of the increase in neutron flux for MUR-PU conditions by accurately calculating a lower EPPY applicability term for the AOR P-T limits. Therefore, the NRC staff finds that the licensee's AOR P-T limits for Farley, Unit 1 and Unit 2, will remain in compliance with 10 CFR Part 50, Appendix G for MUR-PU conditions through 51.9 EFPY and 52.1 EFPY for Farley, Unit 1 and Unit 2, respectively.

The TS requirements ensure LTOP system pressure relief capacity is not affected by the MUR-PU. However, the LTOP system applicability temperature is determined based on the adjusted RT_{NDT} 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 RT_{NDT} value. The NRC staff finds that the licensee's reduction to the EFPY term for the LTOP system applicability temperature, consistent with that used for the limiting adjusted RT_{NDT} values and P-T limits, ensures that the applicability temperature is valid for MUR-PU conditions.

3.2.1.3 Upper Shelf Energy (USE)

3.2.1.3.1 Regulatory Evaluation

Section IV.A.1 of 10 CFR Part 50, Appendix G, provides requirements for maintaining acceptable levels of Charpy USE for RV beltline materials throughout the licensed operating terms of nuclear power reactors. The rule 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" (EMA).

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 staff's recommended guidelines for calculating the effects of neutron radiation on the USE values for the RV beltline materials are provided in RG 1.99, Rev. 2. For power uprate applications, the NRC staff's review must address the impact of the increase in RTP on the licensee's calculation of projected USE values and, if applicable, the EMA results for the RV beltline materials.

3.2.1.3.2 Technical Evaluation

Licensee Evaluation

The licensee discussed the impact of the MUR-PU on the Farley, Unit 1 and Unit 2 USE evaluation in Section IV.1.C.v of Attachment 4 of its LAR. The licensee's AOR USE calculations for 54 EFPY showed that all RV beltline materials will maintain Charpy USE values greater than 50 ft-lbs throughout the operating life of the RVs. To evaluate the effects of the MUR-PU, the licensee recalculated its projected USE values for the Farley, Unit 1 and Unit 2, RV beltline materials using 54 EFPY neutron fluence values that were generated by applying the new 3-D neutron fluence model. The licensee provided a summary of its 54 EFPY USE calculations associated with the MUR-PU for all Farley, Unit 1 and Unit 2, RV beltline materials in Tables IV.1.C-5 and IV.1.C-6 of LAR Attachment 4. The licensee identified that its 54 EFPY USE calculations for MUR-PU conditions show that all RV beltline materials will maintain USE values greater than 50 ft-lbs throughout the operating periods of the RVs, as required by Section IV.A.1 of 10 CFR Part 50, Appendix G.

NRC Staff Evaluation

The NRC staff verified that the licensee correctly recalculated the projected USE values for all RV beltline materials to address MUR-PU conditions by incorporating 54 EFPY neutron fluence values that were generated using the new 3-D neutron fluence mode. The NRC staff's technical review of the licensee's implementation of the 3-D neutron fluence model for generating the new neutron fluence values for MUR-PU conditions is discussed in Section 3.2.1.6 of this SE. The NRC staff verified that the licensee's 54 EFPY USE calculations for MUR-PU conditions demonstrate that all RV beltline materials will maintain USE values greater than 50 ft-lbs throughout the operating life of the RVs; therefore, an EMA is not required. Therefore, the NRC staff finds that the licensee's USE evaluation for the Farley, Unit 1 and Unit 2, RV beltline materials will continue to meet the requirements of Section IV.A.1 of 10 CFR Part 50, Appendix G for MUR-PU conditions.

3.2.1.4 RV Material Surveillance Program

3.2.1.4.1 Regulatory Evaluation

Regulations in 10 CFR Part 50, Appendix H, provides requirements for RV material surveillance programs to monitor changes in RV beltline material fracture resistance due to exposure to neutron radiation. Fracture toughness test data are obtained from testing of material specimens exposed to neutron radiation in surveillance capsules, which are withdrawn periodically from the RV. RV material surveillance data are used in RV beltline fracture toughness evaluations to demonstrate compliance with 10 CFR 50.61 and 10 CFR Part 50, Appendix G.

Regulations in 10 CFR Part 50, Appendix H, requires implementation of a 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" (Reference 36). 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 60-year license terms in Section XI.M31 of NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report" (GALL Revision 2) (Reference 37). GALL Revision 2 recommends that the 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.

For power uprate applications, the NRC staff's review must address the impact of the increase in RTP on the licensee's RV surveillance capsule withdrawal schedule.

3.2.1.4.2 Technical Evaluation

Licensee Evaluation

The licensee discussed the impact of the MUR-PU on the Farley, Unit 1 and Unit 2, surveillance capsule withdrawal schedules in Section IV.1.C.vi, Attachment 4 of the LAR. All six surveillance capsules have been withdrawn from each of the RVs, and their materials have been tested in accordance with the requirements of 10 CFR Part 50, Appendix H and ASTM E185-82. Considering the increase in neutron fluence and RT_{NDT} shift for the limiting RV beltline materials associated with the MUR-PU, the licensee determined that the withdrawal schedule for the Farley, Unit 1 and Unit 2, surveillance capsules continues to meet ASTM E185-82, as required by 10 CFR Part 50, Appendix H, and the recommendations in GALL Revision 2. Therefore, the licensee determined that the previous surveillance capsule withdrawal times remain valid, and no change to the RV material surveillance program is required to support the MUR-PU.

NRC Staff Evaluation

The NRC staff verified that all surveillance capsules have been withdrawn from the Farley, Unit 1 and Unit 2, RVs and the surveillance materials have been tested in accordance with ASTM E 185-82. The NRC staff verified that the previous surveillance capsule withdrawal times remain acceptable for meeting the mandated criteria Appendix H and the recommended criteria in GALL Revision 2 considering the increase in the neutron fluence and RT_{NDT} shift for the limiting RV beltline materials for the MUR-PU. Therefore, the NRC staff finds that the license's RV material surveillance program will continue to meet the requirements of 10 CFR Part 50, Appendix H for MUR-PU conditions.

3.2.1.5 RV Internals and Core Support Structures

3.2.1.5.1 Regulatory Evaluation

The safety functions of the RV internal and core support structure (CSS) components (collectively "RV internals") include structural support and alignment functions to ensure control of reactivity, core cooling, and fission product confinement. Inservice inspection (ISI) and aging management of RV internals is performed to provide timely and reliable detection, evaluation, and corrective actions for addressing service-induced degradation in the RV internal components. Regulatory requirements for "baseline" ISI of RV internals are established in 10 CFR 50.55a, "Codes and standards" and the ASME Code, Section XI.
For renewed license holders, "augmented" aging management program (AMP) criteria for PWR internals are defined in Section XI.M16A of GALL Revision 2. These activities are necessary to ensure that the effects of aging will be adequately managed so that the intended design functions will be maintained consistent with the current licensing basis for the period of extended operation, per 10 CFR 54.21(a)(3). The latest NRC-approved inspection and evaluation guidelines for PWR internals AMPs under the GALL report are established in the Electric Power Research Institute (EPRI) Materials Reliability Program Topical Report MRP-227, Revision 1-A (Reference 38).

3.2.1.5.2 Technical Evaluation

Licensee Evaluation

The licensee discussed the impact of the MUR-PU LAR on the RV internals in Section IV.1.A.ii, Attachment 4 of the LAR. The licensee evaluated the effects of MUR-PU on reactor coolant thermal hydraulic behavior and the resulting impact on the integrity and functionality of the RV internal components. The licensee identified that for power uprates, changes in the reactor coolant vessel outlet temperature (T-hot) and the reactor coolant vessel/core inlet temperature (T-cold) associated with increases in RTP require evaluation for their potential impact on the internals. The maximum T-hot increases by 0.7°F and the minimum T-cold decreases by 0.7°F due to the increase in RTP for MUR-PU conditions. The licensee reported that these temperature changes will have an insignificant impact on the design parameters affecting the functional performance of the internals (i.e., core bypass flow, reactor internals pressure drop, hydraulic lift forces, baffle joint momentum flux, and flow-induced vibration).

The licensee also addressed the potential impact of a slight increase in radiation-induced heat generation in the internal component materials and the resulting impact on their structural integrity for MUR-PU conditions. The internal components that undergo significant radiation-induced heat generation effects are the upper core plate, lower core plate, core baffle plates, former plates, core barrel, baffle-former bolts, and barrel-former bolts. The licensee's evaluation determined that the AOR heat generation rates, thermal stresses, and fatigue cumulative usage factors for these components remain bounding, accounting for the small change in fluid temperatures and the small increase radiation-induced heat generation for the MUR-PU. The licensee concluded that its evaluations demonstrate that the structural integrity and functionality of the internal components are not adversely affected by the MUR-PU conditions.

NRC Staff Evaluation

The NRC staff reviewed the licensee's evaluation to assess the impact of the MUR-PU on the effectiveness of ISI and AMP activities for ensuring RV internal component functionality during periods of extended operation. The NRC staff noted that the small changes to RCS fluid temperatures and heat generation rates for the MUR-PU will not invalidate the existing analyses of the structural integrity and functional performance of the RV internals components. The NRC staff finds that the licensee's evaluations demonstrate that service conditions (e.g., neutron fluence, temperatures, stresses, fatigue cycles, etc.) related to internal component aging degradation will remain bounded by the service conditions used for generic aging degradation analyses supporting development of the MRP-227, Revision 1-A guidelines. Therefore, the NRC staff finds that the MUR-PU will not have an adverse impact on the effectiveness of ISI and AMP activities for ensuring RV internal component functionality during periods of extended operation.

NRC Staff Conclusion Regarding RV Integrity and RV Internal and Core Support Structures

The NRC staff reviewed the licensee's evaluation of the effects of the MUR-PU on the RV integrity analyses required by 10 CFR 50.61 and 10 CFR Part 50, Appendix G, and the RV material surveillance or program criteria required by 10 CFR Part 50, Appendix H. The NRC staff finds that the licensee's evaluation demonstrates that these requirements will continue to be satisfied following implementation of the MUR-PU at Farley, Unit 1 and Unit 2. The NRC staff reviewed the licensee's evaluation of the potential impact of the MUR-PU on the effectiveness of ISI and aging management activities for the RV internal and CSS components. The NRC staff finds the MUR-PU will not have an adverse impact on the suitability of these programs for ensuring component functionality during the periods of extended operation at Farley, Unit 1 and Unit 2. Therefore, the NRC staff concludes that implementation of the MUR-PU is acceptable with respect to RV integrity analyses and plant programs for ISI and aging management of RV internal and CSS components.

3.2.1.6 Neutron Fluence Evaluation

In its LAR, the licensee stated that RAPTOR-M3G and FERRET were used for the neutron fluence calculations for Farley, Units 1 and 2, MUR-PU in the reactor vessel beltline and extended beltline regions.[2] This section addresses item IV.1.C.ii of RIS 2002-03 and includes the review of Attachment 3 and section IV.1.C.ii of Attachment 4 to the LAR.

3.2.1.6.1 Regulatory Evaluation

The NRC staff review was performed in consideration of the requirements contained in GDCs 14, *"Reactor Coolant Pressure Boundary*," GDC 30, "*Quality of Reactor Coolant Pressure Boundary*," and GDC 31, "*Fracture Prevention of Reactor Coolant Pressure Boundary*." These GDCs 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, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," (Reference 39) provides guidance on methods for determining reactor pressure vessel fluence that are acceptable to the NRC staff.

The guidance in RG 1.190 states that an acceptable neutron fluence calculation has the following attributes:

- Fluence estimation using an appropriate calculational methodology
- Analytic uncertainty analysis identifying possible sources of uncertainty
- Comparisons with benchmark measurements and calculations from applicable test facilities including:
    - Plant-specific operating reactor measurements
    - Pressure vessel simulator measurements
    - Calculational benchmarks

The NRC staff notes that RG 1.190 is specific to neutron fluence calculations in the beltline region with close proximity to the active fuel region of the core. The licensee has performed neutron fluence calculations for both beltline and extended beltline regions and the justification for use of those calculations is in Section 3.2.1.6 of this SE, consistent with the limitations and conditions in the NRC staff SE for WCAP-18124-NP-A, "Fluence Determination with RAPTOR-M3G and FERRET" (Reference 40).

[2] As noted in NRC Regulatory Issue Summary (RIS) 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components" (ADAMS Accession No. ML14149A165), the term "beltline" is applicable to all reactor vessel ferritic materials with projected neutron fluence values greater than 1 x 10^{17} neutrons per square centimeter (n/cm^2). In this SE, the phrase "extended beltline" is intended to refer to those beltline regions that are further away from the active fuel region of the core.

3.2.1.6.2 Technical Evaluation

The licensee stated in Attachment 3 to the application dated October 30, 2019, that the fluence calculations were performed in accordance with the Westinghouse Licensing Topical Report WCAP-18124-NP-A.

The licensee provided a description of the neutron fluence methodology in section IV.1.C.ii of Attachment 4 to the LAR. The licensee stated that the neutron calculations were performed in a manner consistent with the guidance described in RG 1.190. The licensee approximated a solution to the Boltzmann transport equation using the three-dimensional discrete ordinates radiation transport code RAPTOR-M3G and least squares adjustment with FERRET (Reference 41). The licensee uses the BUGLE-96 cross-section library, which utilizes 47-neutron and 20-gamma-ray groups specifically for LWR applications (Reference 42). The BUGLE-96 cross-section library is derived from the Brookhaven National Laboratory Evaluated Nuclear Data File, 6th Release (ENDF/B-VI) cross-section library. The guidance in RG 1.190 specifies that ENDF/B-VI-based nuclear data are acceptable. Anisotropic scattering was treated with a P_3 Legendre expansion and the angular discretization was modeled with an S_{12} order of angular quadrature, which are both consistent with recommendations in RG 1.190. RG 1.190 states that when off-midplane locations are analyzed, the adequacy of an angular discretization greater than S_8 order of angular quadrature must be demonstrated. The licensee demonstrated this adequacy of discretization by increasing the degree of Legendre expansion and angular discretization until further parameter refinement resulted in a less than 2-percent change in the core-adjacent beltline region results. The uncertainty of RAPTOR-M3G in the core-adjacent beltline region is within the 20 percent recommended by RG 1.190 for determination of vessel fluence. As described above, the NRC staff reviewed the modeling approach described by the licensee and agrees that the neutron fluence calculations are consistent with RG 1.190 guidance. In addition, RAPTOR-M3G and FERRET for determination of RPV fluence in the core-adjacent beltline region have been approved for use as discussed in the safety evaluation for WCAP-18124-NP-A. Based on these considerations, the NRC staff determined that the fluence methodology for core-adjacent beltline materials acceptable.

The NRC staff SE for WCAP-18124-NP-A (Reference 43) includes two limitations and conditions:

- Applicability of WCAP-18124-NP, Revision 0, is limited to the RPV region near the active height of the core based on the uncertainty analysis performed and the measurement data provided. Additional justification should be provided via additional benchmarking, fluence sensitivity analysis to response parameters of interest (e.g., pressure-temperature limits, material stress/strain), margin assessment, or a combination thereof, for applications of the method to components including, but not limited to, the RPV upper circumferential weld and reactor coolant system inlet and outlet nozzles and reactor vessel internal components.

- Least squares adjustment is acceptable if the adjustments to the M/C ratios and to the calculated spectra values are within the assigned uncertainties of the calculated spectra, the dosimetry measured reaction rates, and the dosimetry reaction cross sections. Should this not be the case, the user should re-examine both measured and calculated values for possible errors. If errors cannot be found, the particular values causing the inconsistency should be disqualified.

The licensee stated that the second limitation and condition does not apply as the least-squares procedures were not used to adjust the calculated fast neutron fluence values for RPV materials evaluated in the reactor vessel integrity analysis.

The licensee's reactor vessel integrity analysis includes materials and components in an extended beltline region of the RPV. The licensee has provided justification for use of RAPTOR-M3G and FERRET in Attachment 3, "Technical Information to Address the Application of WCAP-18124-NP-A in a non-Beltline Region" to the LAR.

The licensee has collected measurement benchmark data through use of ex-vessel neutron dosimetry (EVND). The EVND capsules were installed at the elevation of the reactor vessel support at a 4-loop Westinghouse plant, roughly the same axial height from the core midplane as some of the major extended beltline materials analyzed in the Farley neutron fluence evaluation. The EVND capsules contain a variety of radiometric monitor foils appropriate for use in benchmarking fluence calculations. Some of the most common reactions that occur in RPV materials include those with iron, copper, and nickel. All of these materials are included in the EVND capsules and account for the lowest reaction rate uncertainties of all reactions at a 1σ uncertainty of 5 percent. The NRC staff considers the use of EVND capsules for additional benchmarking acceptable, because RG 1.190 indicates that EVND is an acceptable means of qualifying fluence estimates, and because the EVND under consideration had been installed in an upper elevation where the RG indicates that cavity streaming effects may have a more dominant influence on the total fluence, in comparison to a core mid-plane location.

The licensee provided measured-to-calculated (M/C) ratios of the calculated EVND capsule reaction rates and the measured data from counting laboratories. This comparison established that the average M/C ratio is 0.78 with a standard deviation of 25.5 percent. It is noted that the iron, copper, and nickel capsules have M/C ratios consistently below average with small standard deviations. Additionally, the licensee provided best-estimate-to-calculated (BE/C) ratios for the fluence rate and iron atom displacement rate. This comparison determined that the BE/C ratio for the fluence rate is 0.84 with a standard deviation of 8.9 percent and a BE/C ratio for iron atom displacement rate of 0.93 with a standard deviation of 11 percent. These three comparisons demonstrate that the calculations consistently over predict the neutron exposure. Additionally, both of the BE/C ratio uncertainties are within the ±20 percent uncertainty described in RG 1.190. RG 1.190 also describes an acceptable ±30 percent uncertainty for cavity dosimetry, which the NRC staff considered in the justification for the 25.5 percent uncertainty of the M/C reaction rate ratio. All of the neutron transport calculations were carried out using RAPTOR-M3G and BUGLE-96 cross section library. The anisotropic scattering was treated with a P_3 Legendre expansion and the angular discretization modeled in an S_{20} order of angular quadrature. RG 1.190 states that a minimum of an S_8 order of angular quadrature is acceptable.

The NRC staff noted a discrepancy in the neutron fluence evaluation with respect to the angular discretization. In section IV.1.C.ii of Attachment 4 to the LAR, the licensee stated that an S_{12} order of angular quadrature was used in the neutron fluence evaluations. In Attachment 3 to the LAR, the licensee stated that an S_{20} order of angular quadrature was used in the additional benchmarking analysis for neutron fluence evaluation in extended beltline materials. The ±30 percent uncertainty proposed by the licensee is based on the benchmarking results using an S_{20} order of angular quadrature. In its response to a RAI dated April 22, 2020, the licensee stated that an additional 15.2 percent uncertainty is introduced by reducing the order of angular quadrature. By combining the square root of the sum of the squares, the total uncertainty equates to 33.6 percent, an increase of 3.6 percent in total uncertainty for extended beltline materials. The NRC staff does not consider this increase in uncertainty to be significant as extended beltline materials are not limiting by a significant margin, as discussed in the remainder of this SE. The NRC staff concludes that the treatment of the order of angular quadrature for calculation of vessel fluence for beltline and extended beltline materials is acceptable.

Guidance in RG 1.190 describes an acceptable uncertainty in fast neutron fluence calculations of within ±20 percent. The analytic uncertainty described in WCAP-18124-NP-A is about 19-20 percent at the top and bottom of the active fuel. This is consistent with the guidance in RG 1.190. The licensee estimates the uncertainty of RAPTOR-M3G to be on the order of ±30 percent at the RPV supports. This estimation is derived from the M/C and BE/C ratios described above and the analytic uncertainty associated with the cavity locations and the top and bottom of the core. The 20 percent uncertainty allowances recommended by RG 1.190 are based on associated margin terms provided in the calculation of the Reference temperature for nil-ductility transition. However, RG 1.190 suggests that more approximate methods for determining the fluence may be appropriate when there is a large margin to the RT_{NDT} limits. The licensee indicated that RAPTOR-M3G consistently over predicts the fast neutron fluence and that none of the extended beltline materials are limiting and have significant margin to become limiting. The NRC staff reviewed the extended beltline materials, and their associated RT_{PTS}, and determined that the most limiting extended beltline material for both Units 1 and 2 is the upper shell forging (nozzle shell). The NRC staff performed a sensitivity study on the uncertainty required for the nozzle shell to become the most limiting RPV material. The results of that study show that a departure from the best estimate fluence of 110 percent and 190 percent for Units 1 and 2, respectively, would be required for the nozzle shell to be the most limiting material. Based on a review of the benchmarking data, a departure of 110 percent is equivalent to 3.67σ, and is therefore a significant difference from the uncertainty estimated by the licensee. Therefore, the NRC staff determined that the justification for application of the RAPTOR-M3G fluence methodology in extended beltline regions is acceptable, and that the first limitation and condition in the NRC staff SE for WCAP-18124-NP-A is satisfied.

3.2.1.6.3 NRC Staff Conclusion Regarding Neutron Fluence

The NRC staff finds that the neutron fluence calculation provided by the licensee in support of the MUR-PU is consistent with the guidance in RG 1.190, and adequately addresses the limitations and conditions in the NRC staff SE of WCAP-18124-NP-A. Therefore, the NRC staff concludes that the licensee’s analysis provides reasonable assurance is provided, with respect to the use of RAPTOR-M3G and FERRET for neutron fluence calculations in beltline and extended beltline regions of Farley, Units 1 and 2, and is, therefore, acceptable.

3.2.2 Mechanical and Inservice Testing

3.2.2.1 Regulatory Evaluation

The Farley Units consist of three loop Westinghouse pressurized-water reactors. Each unit is designed to generate 2775 MWt, or approximately 910 Megawatt electric (MWe).

The UFSAR, Section 3.1 addresses conformance with NRC GDC published as Appendix A to 10 CFR 50 in July 1971, including Criterion 4 as amended on October 27, 1987. The design features and procedures of Farley conform to the GDC.

The NRC staff's review of the LAR in the areas of mechanical and inservice testing focused on verifying that the licensee has provided reasonable assurance that the structural and pressure boundary integrity of Structures, Systems and Components (SSCs) at Farley will continue to be maintained adequately following the implementation of the proposed MUR-PU under normal, upset, emergency and faulted loading conditions, as applicable. Reasonable assurance is provided by demonstrating compliance with the NRC regulations listed below, which address the mechanical and inservice testing scope of the NRC staff's review.

The NRC staff's assessment of the LAR in the areas of mechanical and inservice testing considered the following regulations: 10 CFR 50.55a, "Codes and standards"; 10 CFR 50, Appendix A, GDC 1, "*Quality standards and records*"; GDC 2, "*Design bases for protection against natural phenomena*"; GDC 4, "*Environmental and dynamic effects design bases*"; GDC 14, "*Reactor coolant pressure boundary*"; and GDC 15, "*Reactor coolant system design*."

The acceptance criteria are based on continued conformance with the requirements of the following regulations: (1) 10 CFR 50.55a, and GDC 1 as they relate to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed; (2) GDC 2 as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions; (3) GDC 4 as it relates to structures and components important to safety being designed to accommodate the effects of, and to be compatible with, the environmental conditions of normal and accident conditions and these structures and components being appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids; (4) GDC 14 as it relates to the reactor coolant pressure boundary being designed, fabricated, erected, and tested to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture; and (5) GDC 15 as it relates to the RCS being designed with sufficient margin to ensure that the design conditions are not exceeded.

The NRC staff's review focused primarily 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-PU conditions. This, in turn, provides reasonable assurance that compliance with the applicable regulations will be maintained upon implementation of the proposed MUR-PU.

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

3.2.2.2 Technical Evaluation

The NRC staff's review in the areas of mechanical and inservice testing covers the structural and pressure boundary integrity of the piping, components and supports, which make up the NSSS and the balance-of-plant (BOP) systems. The mechanical and inservice testing review scope also includes an evaluation of other new or existing SSCs, which are affected by the implementation of the proposed MUR-PU. Specifically, this review focuses on the impact of the proposed MUR-PU on the structural integrity of the Farley pressure-retaining components and their supports and the RV Internals. The NRC staff's review also considered the impact of the proposed MUR-PU on postulated HELBs locations and corresponding dynamic effects resulting from the postulated HELBs, including pipe whipping and jet impingement. A review of the impact of the MUR-PU on moderate energy pipe rupture locations 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 piping systems, components, component internals and their supports under normal and transient loadings, including those due postulated accidents and natural phenomena, such as earthquakes.

The proposed MUR-PU will increase the rated thermal power level from 2775 MWt to 2821 MWt. In accordance with the 10 CFR 50, Appendix K requirements discussed above, the licensee notes in Section IV of Enclosure 2 of the LAR that maximum analyzed thermal power of 2831 MWt corresponding to 102 percent of 2775 MWt remains unchanged. As noted in Section 3.2 and Table 3.2-1 of the LAR, the licensee has previously performed NSSS and some BOP evaluations assuming a power level of 2831 MWt and the implementation of the proposed MUR-PU would revise the RTP to a level lower than that for which the licensee has already analyzed.

3.2.2.2.1 Power Uprate Evaluation Parameters and Design Bases

In Table IV-1 in Section IV.1 of Attachment 4 to the LAR, the licensee provided the pertinent temperatures, pressures, and flow rates for the current and uprated conditions. The licensee evaluated the effects of the proposed MUR-PU at a bounding power level of 102 percent RTP (2831 MWt). This power level corresponds to the proposed level following the implementation of the MUR-PU (i.e 2821 MWt) plus the revised uncertainty of 0.3 percent.

Section II.1.D.ii, Attachment 4, of the LAR discusses UFSAR Chapter 15 analyses. The licensee performed a review of UFSAR Chapter 15 to support the Farley MUR-PU with respect to the accident analyses. The existing AOR, as currently presented in the UFSAR, were either performed conservatively and remain valid and bounding for the proposed power uprate or were explicitly reanalyzed.

As shown in Table IV-1 of the LAR, there is no change in the RCS operating pressure 2250 pounds per square inch absolute (psia)) as a result of the MUR-PU. The RCS mechanical design flow of 101,800 gpm also remains unchanged after implementation of the MUR-PU. At full power, the implementation of the MUR-PU would yield a hot leg temperature (T-hot) of 614 F (from the current temperature of 613.3°F) and a cold leg temperature (T-cold) of 540.5°F (from the current temperature of 541.1°F), resulting in no change to the average RCS temperature. The main steam (MS) pressure decreases by 8 psia at the MUR-PU conditions and the MS steam flow increases from 12.24 Mlbm/hr to 12.54 Mlbm/hr at the MUR-PU conditions. The FW temperature would increase by 2.6°F to 446°F as a result of MUR-PU implementation.

The information related to the structural qualification of SSCs at Farley is contained in Chapter 3 of the UFSAR. The UFSAR describes the design criteria applicable to the Farley SSCs, including loads, load combinations, and acceptance criteria stipulated by the applicable codes of record for these SSCs. In section IV.1.A, Attachment 4, of the LAR, the licensee notes that implementation of the LAR does not change current operating transients, nor does it introduce additional transients. As such, loads resulting from these transients that are used in the structural evaluations of SSCs are not affected. Similarly, the proposed MUR-PU has no effect on the deadweight and seismic loads of existing SSCs. Therefore, the NRC staff finds that the loads used in the existing AOR for these SSCs remain valid.

The functional description of the RCS, including the RV, RCPs, RCS piping and SGs is discussed in Chapter 5 of the Farley UFSAR. Chapter 10 of the Farley UFSAR provides the design basis information for the secondary side systems, including the MS and the FW and condensate system.

Section IV.1.D, Attachment 4 of the LAR, notes that the applicable code of record for the Farley RCS components (Piping and Supports, Steam Generator tube side and shell side, Reactor Vessel, Reactor Vessel Closure Heads, Reactor Coolant Pump Casing, CRDMs, and Pressurizer) is ASME Boiler and Pressure Vessel Code, Section III, and there were no changes to the ASME Codes of Record. Component supports are addressed in UFSAR Section 5.5.14.

As discussed in UFSAR Section 4.2.2.5, the allowable stress limits during the DBA used for the core support structures are based on the January 1971 draft of the ASME Code for Core Support Structures, Subsection NG, and the Criteria for Faulted Conditions. Design of the replaced baffle-former bolts in Farley reactor vessels is based on the 1989 ASME Code, Subsection NG, as a guideline for allowable stress limits during the design basis accident.

A listing of Class 1 Code Cases used for the Farley RCS is provided in the UFSAR Section 5.2.1.4. A listing of ASME Code Cases for Class 1 components is provided in the UFSAR Table 3.2-5. No stress/fatigue analyses were revised, and, therefore no Codes of Record changed.

The licensee confirmed that MUR-PU evaluations did not include any changes to the tabulated design codes of record. The pressure-retaining components and component supports, including piping and pipe supports, which must be evaluated in support of an MUR-PU include the following: the RPV, including the RPV shell, RPV nozzles and 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, 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 evaluation of those SSCs that AOR are affected by implementation of an MUR-PU should identify and evaluate any changes related to the power uprate in the following areas:

- 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

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-PU. Affected components and supports refer to those for which their AOR is not bounded at MUR-PU conditions. Pressure-retaining components and their supports generally remain unaffected by the implementation of an MUR-PU, based on the fact that they have been analyzed at conditions which are more limiting than those which will be present at MUR-PU conditions (i.e., bounded). The licensee was able to disposition a number of components and their associated supports as unaffected by the proposed MUR-PU, based on whether the plant parameter changes resulting from implementation of the MUR-PU, identified above, affect the loads included in the AOR for the component and its supports. Based on its evaluations of the impact of MUR-PU 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-PU at Farley: the RPV, RPV nozzles and RPV supports; the pressure-retaining portions of the CRDMs; RCS piping and supports and loop branch nozzles; pressurizer shell, nozzles and surge line; the replacement SGs, including the shells, nozzles and secondary side internal support structures; and the pressure-retaining portions of the RCPs.

The licensee evaluated CRDMs as shown in Section IV.1.A.iii, Attachment 4 of the LAR. There are only small changes to the plant operating parameters, namely T-hot and T-cold. It is also concluded that the design transient definitions and frequency of occurrences applicable to current power conditions remain applicable for MUR-PU conditions. Therefore, the seismic response of the NSSS is not significantly impacted by the MUR-PU.

The impact of the MUR-PU on LOCA hydraulic loading was also evaluated by the licensee and determined to be insignificant. Therefore, the response of the reactor assembly to this LOCA hydraulic loading would not be affected, and the LOCA stresses calculated in the AOR for CRDM remain valid for MUR-PU conditions.

All relevant plant parameters pertaining to the structural evaluation of the CRDMs are not significantly impacted by MUR-PU conditions, and, therefore, the CRDM design report remains applicable without modification.

Section IV.1.A.viii, Attachment 4 of the LAR, addresses evaluation for pressurizer shell, nozzles, and surge line. The licensee reviewed MUR-PU operating conditions for impact on the existing pressurizer design basis analysis for Farley. The limiting T-hot and T-cold conditions for MUR-PU did not change from the current operation. Since the changes in T-hot and T-cold are enveloped by the AOR parameters, the pressurizer shell, nozzles, and surge line are acceptable for MUR-PU conditions.

The NRC staff reviewed balance-of-plant (BOP) piping as discussed in Section IV.1.A.v, Attachment 4 of the LAR. The licensee's evaluation of the structural integrity of those BOP piping systems also demonstrated that the BOP piping systems will continue to meet their design basis under MUR-PU conditions and remain bounded by the current AOR at MUR-PU conditions. The licensee accounted for FW temperature increase by 2.6°F, main steam flow increase by 0.3E6 lbm/hr, main steam temperature decrease by 1.2°F, and main steam pressure decrease by 8 psi by using appropriate thermal, pressure, and flow change factors. The BOP piping systems remain acceptable for MUR-PU conditions. Similarly, the licensee confirmed that the MUR-PU has no effect on the structural integrity of safety-related valves at Farley and these also remain bounded by their current AOR. Based on these considerations, the NRC staff concludes that all pressure-retaining components and supports, including piping and pipe supports remain bounded at MUR-PU conditions.

The NRC staff considered the licensee's assessments of the pressure-retaining components and component supports acceptable based on the following considerations: (1) the licensee's approach to disposition SSCs as unaffected by the proposed power uprate is consistent with RIS 2002-03; (2) the licensee confirmed that the existing AORs for all of the aforementioned SSCs remain bounding when considering the plant parameter changes at the MUR-PU level, ensuring that there will be no impact on the structural and pressure boundary integrity of these SSCs at the MUR-PU level; and (3) the magnitudes of plant parameter changes, as documented in Table IV-1 of Attachment 4 to the LAR are generally minor and support the licensee's assessment, which concludes that all pressure-retaining components remain bounded.

Based on these considerations, the NRC staff concludes that there is reasonable assurance that the structural and pressure boundary integrity of SSCs will be adequately maintained following the implementation of the MUR-PU.

3.2.2.2.3 RV Internals

In accordance with Section IV.1.A.ii of RIS 2002-03, the licensee evaluated the effects of the proposed MUR-PU on the Farley RV internals. As discussed above, Section IV.1.B of RIS 2002-03 indicates that for those SSCs, including RV internals, whose AORs are affected by implementation of an MUR-PU, the licensee should address the following, as they relate to the impact of the uprate on the AOR: stresses, cumulative usage factors (i.e., fatigue), flow-induced vibration (FIV), and changes in temperature, pressure and flow rates resulting from the MUR-PU. The licensee summarized its evaluation of the effects of the proposed MUR-PU on the structural integrity of the RV internals in Section IV.1.A. ii, Attachment 4 of the LAR.

Mechanical and structural evaluations were performed by the licensee to determine any effects on the RV internals due to the conditions, which would be present following the implementation of the proposed MUR-PU. The mechanical evaluations of FIV performed by the licensee are summarized in Section IV.1.B.iii, Attachment 4 of the LAR. These evaluations focused on the potential for an increase in the vibratory response of the RV internals resulting from changes in the flow field at the MUR power level. An increase in vibratory response can introduce increased alternating stress intensities and subsequently higher cyclic fatigue of the RV internals. Per the values in Table IV-1, the volumetric mechanical design flow remains unchanged for the MUR-PU. Hence the vortex shedding frequencies remain unchanged. Also, the temperature changes (maximum T-hot increase is 0.7 deg. F, T-cold decrease is 0.7 deg. F) due to the MUR-PU are less than 0.1 percent, which causes a negligible change in the frequencies of the internals. Thus, the stresses imparted on the RPV internals due to flow induced vibrations remain unchanged as a result of the MUR-PU conditions, and the existing AOR remain bounding. Based on these considerations, the licensee confirmed that the FIV characteristics of the RV internals are bounded by the current AOR. The changes in operating temperatures are provided in Table IV-1. The average temperature is unchanged, and the cold leg decreases 0.7°F, while the hot leg temperature increases 0.7°F. These changes have minimal impact on the MUR-PU. Based on this assessment, the licensee noted that the RV internals remain bounded at MUR-PU conditions and no revision to the AOR is required to support MUR-PU implementation.

The NRC staff reviewed the licensee's assessment of the RV internals and considers the licensee's evaluation acceptable. Concerning the effects of the MUR-PU on the FIV of the RV internals, the NRC staff finds that the licensee's assessment is acceptable given that it is shown in the licensee's submittal that the RCS operating parameters (flow, temperature, and pressure), which directly affect FIV either do not change or do not change enough to affect the FIV of the RV internals. For the structural evaluations, the NRC staff finds that the licensee's conclusion that the RV internals are bounded by the current AOR at the MUR-PU conditions is acceptable, based on the RV internals have been previously evaluated at a power level, which is greater than the proposed MUR-PU power level. Additionally, a comparison between the RCS operating parameters before and after MUR-PU implementation suggests that there would be a minimal impact on the loads used in the evaluation of the RV internals for structural integrity.

Further, no abnormal loads (i.e., transient and seismic) are changing as a result of the MUR-PU. Therefore, the NRC staff concludes that the design basis analyses of the RV internals remain unaffected and bounding following implementation of the MUR-PU.

3.2.2.2.4 Postulated Pipe Ruptures and Associated Dynamic Effects

The licensee evaluated the effects of the proposed MUR-PU on systems classified as high energy to determine whether any changes to the HELB AOR will result from the implementation of the MUR-PU. This assessment is summarized in section IV.1.B.vii, Attachment 4 to the LAR. The licensee stated in a summary to its assessment that the current AORs were reviewed to determine whether the MUR-PU would have any impact on the current HELB AOR. The licensee concluded that because the temperature and pressure changes in high energy systems are considered nominal, no new HELB locations are required to be postulated as a result of MUR-PU implementation.

For the Moderate Energy Line Breaks (MELBs), the licensee also confirmed that the MUR-PU has no effect on moderate energy piping systems and, as such, no new moderate energy pipe cracks are required to be postulated.

The licensee summarized its assessment of the impact of MUR-PU implementation on jet impingement and thrust forces (dynamic effects) in section IV.1.B.vii.b, Attachment 4 of its LAR. Because there was no adverse impact as a result of the MUR-PU to high/moderate energy piping systems in areas with safety related components, there will also be no impact to applicable pipe break evaluations. Therefore, the NRC staff concludes that the design basis for pipe break, jet impingement, and pipe whip considerations remains valid for MUR-PU conditions.

The licensee stated that it had justified the elimination of large primary loop pipe rupture and pressurizer surge line pipe rupture from the design basis for Farley by using Leak-Before-Break (LBB) concepts. The licensee also confirmed that piping loads used in the LBB evaluation are not affected by the power uprate and concluded that the LBB evaluation remains acceptable and is bounded by existing AOR. The licensee concluded that these are not affected by the implementation of the MUR-PU because the changes in the temperatures and pressures of these systems resulting from MUR-PU implementation were within the bounds of the temperatures and pressures, which have been previously evaluated.

The NRC staff reviewed the licensee's evaluations related to determinations of pipe rupture locations and their corresponding dynamic effects and concludes that the licensee's assessments performed for these areas is acceptable. The AOR related to HELBs, MELBs and dynamic effects resulting from postulated pipe ruptures remain bounding under the proposed MUR-PU. Given the small magnitude in temperature and pressure increases, these small changes generally have no substantive impact on pressure-retaining components such as piping.

The licensee summarized its assessment of the impact of MUR-PU on the LBB analysis of the primary reactor coolant loop (RCL) and the pressurizer surge line piping associated with Farley, Units 1 and 2. Since there were no physical plant modifications to the RCL and surge line piping systems, only operational changes associated with temperature, pressure and flow of the reactor coolant would affect the LBB analysis. The evaluation of the LBB analysis for the MUR-PU included the LBB inputs that potentially could be affected by operational changes, such as NSSS design parameters, NSSS design transients, and RCS piping loads. The licensee confirmed that piping loads used in the LBB evaluation are not affected by the power uprate since the changes in the temperatures and pressures of these systems resulting from MUR-PU implementation were within the bounds of the temperatures and pressures, which have been previously evaluated. Therefore, the licensee concluded that the LBB evaluation remains acceptable and is bounded by existing AOR.

Given the small magnitude in temperature and pressure increases, which accompany MUR-PU implementation, the NRC finds there is reasonable assurance that these small changes generally have no impact on pressure-retaining components such as piping.

3.2.2.3 NRC Staff Conclusion Regarding Mechanical and Civil Engineering

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

3.2.3 Electrical Engineering

3.2.3.1 Regulatory Evaluation

The licensee developed the LAR consistent with the guidelines in RIS 2002-03. The regulatory requirements include the following:

Regulations in 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants," requires that licensees establish programs to qualify electric equipment important to safety.

Regulations in 10 CFR 50.63, "Loss of all alternating current power," 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.

Regulations in 10 CFR 50, Appendix A, GDC 17, "*Electric power systems*," require, in part, that an onsite power system and an offsite electrical power system provide enough capacity and capability to permit functioning of SSCs important to safety. Conformance to GDC 17 is discussed in Section 3.1 of the UFSAR.

3.2.3.2 Technical Evaluation

The electrical equipment design information is provided in Section V, Attachment 4 of the LAR. The NRC staff reviewed the licensee's evaluation of the impact of the MUR-PU on the following electrical systems/components:

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

3.2.3.2.1 AC Distribution System

The AC Distribution System is the source of power for the non-safety-related buses, the safety-related emergency buses, and the loads supplied by them. According to Section 8.2 of the Farley UFSAR, Unit 1 and Unit 2, are connected by separate generator step-up transformers to the 230 kilovolt (kV) and 500 kV switchyards, respectively. Section V of Attachment 4 of the LAR indicates that the 22 kV output of each main generator connects to unit auxiliary transformers (UATs) which normally supply the 4.16 kV normal and essential buses. Section 8.3.1 of the UFSAR states that the AC auxiliary system for each unit consists of the 4.16 kV, 600 VAC, 480 VAC, 208 VAC, and 120 VAC subsystems, and each is designed to provide reliable electrical power during all modes of plant operation and shutdown conditions. The AC auxiliary system for each unit is designed with enough power sources and redundant buses to accomplish this.

The licensee stated in its LAR that each Farley unit will see load increases only at the 4.16 kV and 120 VAC vital power buses due to the proposed MUR-PU. This load increase at the 4.16 kV buses is discussed in more detail in Section 3.2.3.2.2 of this SE for the unit and startup auxiliary transformers. The NRC staff verified that this modest increased loading at the 4.16 kV buses are within the ratings of those buses based on their 1200-, 2000-, or 3000-ampere ratings as stated in Section 8.3.1.1.3 of the UFSAR. The load increase at the 120 VAC vital power buses is discussed below for the low voltage AC (LVAC) power system.

The licensee did not identify any load changes at the 600 VAC and 480 VAC voltage levels. Therefore, the NRC staff concludes that the AC power system at those voltage levels has adequate capacity to operate the plant equipment within its design to support implementation of the MUR-PU.

The LVAC power system consists of the 120 VAC vital power, 120 VAC regulated power, and the 208/120 VAC power systems for Farley. The 120 VAC vital power source supplies the leading-edge flowmeters (LEFMs) from non-safety-related buses. Each set of Caldon LEFMs per unit draws about 5 amps. The licensee stated this increase in loading is bounded within the existing AOR for ratings of 120 VAC vital power. The licensee's evaluation of the LVAC power system determined that, the system shall be capable of performing its assigned functions without exceeding equipment ratings or industry guidelines. Based on the low current drawn by each set of Caldon LEFMS per unit and that the existing analysis is still bounding, the NRC staff finds LVAC power system will perform its intended function at MUR-PU plant conditions.

3.2.3.2.2 Power Block Equipment

The main power system for each Farley unit consists of the main generator, main transformer, unit auxiliary transformer(s), and startup auxiliary transformers as indicated in Section V of Attachment 4 to the LAR and Section 8.1.2 of the UFSAR.

Because of the proposed MUR-PU, the rated thermal power will increase from the previously analyzed core power level of 2775 MWt to 2821 MWt. The turbine-generator for each Farley unit converts the thermal energy of steam produced in the steam generators into mechanical shaft power and then into electrical energy. The generator is rated at 1045 mega-volt amperes (MVA) with 75 psig hydrogen pressure, 0.85 power factor, 22 kV three-phase, and 60 Hz at 1800 rpm. This increase in thermal power results in an increase in each unit's main generator electrical power output as follows, based on Section V, Attachment 4 of the LAR:

Unit 1:

- 944.7 megawatts-electrical (MWe), 1045 MVA, 447 MVAR, at 0.90 lagging power factor
- 944.7 MWe, 971MVA, 225 MVAR, at 0.97 leading power factor

Unit 2:

- 953.3 MWe, 1045 MVA, 428 MVAR, at 0.91 lagging power factor
- 953.3 MWe, 978 MVA, 220 MVAR, at 0.97 leading power factor

The NRC does not approve the revised power output in MWe. The NRC review was limited to confirming the increase did not present a design or operating challenge to equipment capacity. The licensee made a number of changes (e.g., power factor) to derive added output efficiency that do not require prior NRC approval.

By NRC letter dated April 29, 1998 (Reference 44), Amendment Nos, 137 and 129, the licensee undertook a program to uprate the Farley units to a maximum reactor core power level of 2775 MWt. Each main generator's output increased approximately 25 MWe from 885 to 910 MWe for that 1998 power uprate. At uprated conditions for the MUR-PU, the generator output for each Farley unit will remain at 1045 MVA with the power factor for each main generator, at both lagging and leading conditions, increasing slightly for both units which means increased MWe output for each unit. Since the main generator for each Farley unit is the same type and rating, the scenario for Farley Unit 2 with the generator output of 1045 MVA and either 0.91

lagging or 0.97 leading power factor provides the bounding cases for both Farley units regarding the overall increase in MWe output based on the loading conditions analyzed by the licensee for the MUR-PU. The electrical output for the Unit 1 main generator will increase to 944.7 MWe, and the electrical output for the Unit 2 main generator will increase to 953.3 MWe after changes for the MUR-PU are adopted. The licensee has stated that, with the increase in electrical output, the main generators for Units 1 and 2 shall be capable of performing their assigned functions without exceeding equipment ratings. Since each main generator will be operating within its design ratings just at different locations on its capability curve and that the licensee performed a similar evaluation of each main generator's capability for the successful power uprate in 1998, the NRC staff finds that that the main generator for each unit will operate within its capability for the conditions at the MUR-PU.

Each unit has a main transformer (MT) that connects the main generator output of 22 kV with the respective switchyard. Unit 1 MT connects to the 230 kV switchyard, and Unit 2 MT connects to the 500 kV switchyard. The MT for each Farley unit has a maximum design rating of 1200 MVA, which is greater than the main generator output capability of 1045 MVA. The NRC staff finds that the MT for each Farley unit is adequately sized for MUR-PU.
As indicated in Section V, Attachment 4 of the LAR and Section 8.1.2 of the UFSAR, each unit is provided with at least one UAT. The power required for Units 1 and 2 station auxiliary loads during normal operation is usually supplied from each main generator through a UAT. The licensee stated that the only 4.16kV loads affected by the uprate are three condensate pumps for each unit with each pump having an increase of 25 horsepower (HP). The NRC staff determined that the 75 HP increase for the three condensate pumps combined equates to about 0.05595 MVA. At the lowest rating of 12.5 MVA, per section 8.3.1.1.3 of UFSAR, of a UAT at 4.16kV, the loading on a UAT would only increase about 0.45 percent for that increase of .05595 MVA. The NRC staff concludes that modest load increase would be within the capability of a UAT. The licensee also stated in its April 22, 2020, response to an request for additional information (RAI) that the brake horsepower (BHP) of each condensate pump, at MUR-PU conditions, was determined to be 2909 HP (2169 KW) for Unit 1 and 2937 HP (2190 KW) for Unit 2. However, the Electrical Transient Analyzer Program (ETAP), which is the load analyzing model used by the licensee, assumes each condensate pump operates at 3000 HP. Therefore, the MUR-PU increased loads for each condensate pump at MUR-PU conditions are bounded by the 3000 HP assumed for each condensate pump in ETAP. As a result, the increased BHP of 25 HP per condensate pump will not adversely affect the UATs and startup auxiliary transformers (SATs) analyzed power capability. The licensee stated that load changes at MUR-PU conditions based on evaluation are within each UAT's design ratings. The licensee also stated that there would be no changes to the protective device settings for the UATs, nor to the sizes of the cables for the UATs, which the NRC staff interprets as the supply and output cables. Based on the above, the NRC staff agrees with the licensee that any increased loading of the unit auxiliary transformers for MUR-PU will be within their nominal capability.

Section 8.2.1.3 of the Farley UFSAR and Section V, Attachment 4 of the LAR states that the four SATs, two for each unit, are connected to the Alabama Power Company's transmission system through four separate 230-kV oil-static cables. These transformers provide a source of power for startup, shutdown, and after-shutdown requirements for both units. The NRC staff determined that at the lowest rating of 13 MVA, per section 8.3.1.1.3 of UFSAR, of a SAT at 4.16 kV that the loading on a SAT would increase about 0.43 percent for the increase of .05595 MVA if it supplied three condensate pumps at MUR-PU conditions. That modest load increase would be within the capability of a SAT. As was stated above for the UATs, the licensee used the ETAP program to analyze the loading of the 4.16 kV buses and the UATs and

SATs. The licensee stated, based on evaluation, that load changes at MUR-PU conditions are within each SAT ratings and are bounded by existing calculations. The licensee also stated that there would be no changes to the protective device settings for the SATs, nor to the sizes of the cables for the SATs, which the NRC staff interprets as the supply and output cables. The NRC staff finds that each SAT will not have its ratings exceeded for the MUR-PU including its cabling and will thus be acceptable for its intended service at MUR-PU conditions.

3.2.3.2.3 DC Power System

The direct current (DC) power systems described in UFSAR section 8.3.2 and Section V, Attachment 4 of the LAR provide a reliable source of continuous power for control, instrumentation, and emergency lighting. The safety-related DC systems are in the auxiliary building and service water area. The 125 VDC system in the auxiliary building for each unit consists of two 125 VDC switchgear assemblies, three 125 VDC battery chargers, two 125 VDC batteries, and six DC distribution cabinets. The 125 VDC system in the service water area consists of two independent and redundant subsystems. Each subsystem consists of two battery/chargers sets and two DC distribution panels. The DC power system for the non-safety-related loads is independent of and separated from the safety-related DC system. Two 125 VDC battery chargers and two independent 60-cell batteries, located in the high voltage switchyard, supply separate DC distribution cabinets to provide power for tripping, through primary and secondary relay systems, for protection and control of 230 kV and 500 kV circuits.

The licensee's evaluation of the DC power system determined that there will be no change due to operation under the MUR-PU conditions, and it shall be capable of performing its assigned functions without exceeding equipment ratings or industry guidelines. Based on the above, NRC staff finds that the DC power system will operate within its capability after the MUR-PU is incorporated.

3.2.3.2.4 Emergency Diesel Generators

The onsite emergency AC power supply, per Section V, Attachment 4 of the LAR consists of five emergency diesel generators (EDGs) that supply standby power for the 4.16 kV emergency buses of each unit when offsite power is unavailable. The loading on the EDGs was evaluated to determine potential changes at the MUR-PU levels. The evaluation determined that there were no changes to the loading or the load sequencing of the EDGs. Therefore, there is no impact to the existing EDG loading analysis, which bounds the MUR-PU conditions. The emergency onsite power system has adequate capacity and capability to provide onsite standby power for safety-related loads following a loss of offsite power. Therefore, the existing emergency onsite power system bounds the design requirements at the MUR-PU levels. The NRC staff finds for the onsite emergency AC power supply system that there are no load additions, modifications, or changes in load sequences or durations for the existing EDGs.

3.2.3.2.5 Switchyard

The Farley switchyards, per Section V, Attachment 4 of the LAR, are owned and maintained by Alabama Power Company. The 230 kV and 500 kV switchyards both employ a breaker-and-a-half arrangement-to provide the necessary operating flexibility and, consequently, reliability. The licensee stated that the connection between the Unit 1 MT 230 kV high voltage side and the first circuit breaker in the switchyard has been designed to accommodate the maximum output current of the MT at its 1200 MVA rating. The licensee further stated that the connection between the Unit 2 MT 500 kV high voltage side and the first circuit breaker in the switchyard has been designed to accommodate a maximum output current of the MT at its 1200 MVA rating. The SATs for each unit are connected to the 230 kV switchyard. The licensee stated that the connections between the SATs and the switchyard are not impacted, since the loads on the SATs are not altered except for the modest change due to the three condensate pumps having an increase of 25 HP each. This increase is addressed in Section 3.2.3.2.2 above. Based on the licensee's evaluation of the switchyard which indicated that each unit's MT and SATs would function within their design capability, the NRC staff finds there is reasonable assurance that the switchyard will be able to perform it's intended function.

3.2.3.2.6 Grid Stability

The licensee performed studies for grid stability/transmission planning, as described in Section V, Attachment 4 of the LAR. Those studies for (1) local area system impacts, (2) stability impacts, (3) nuclear plant offsite power Impacts, (4) bus ampacity impacts, and (5) interface transfer capability impacts were conducted starting with load flow base cases developed for use as one possible scenario of the future of what could occur. Information from other transmission owners within the Southern Balancing Authority Area (SBAA) was obtained and incorporated in these models. The systems external to the SBAA were obtained either from the given balancing authority or from the most recent North American Electric Reliability Corporation Multiregional Modeling Working Group cases. The Farley facility requested a 48 MWe incremental designation to serve the native load of Southern Companies. The study for the local area system impacts reviewed two scenarios with one considering the 48 MWe incremental designation. For that scenario, a steady state analysis did not identify any transmission constraints that require any transmission projects. The studies for the stability, nuclear plant offsite power, and bus ampacity impacts did not identify any stability-related transmission constraints nor transmission projects attributable to the 48 MWe incremental designation. The study for interface transfer capability impacts performed an assessment of the impact of the incremental designation on import and export transfer capability across the SBAA interlaces and did not identify any transmission constraints that require transmission projects.

The NRC staff finds reasonable assurance is provided by the studies of grid stability/transmission planning for the incremental designation of 48 MWe requested by the Farley plant to address future needs including its MUR-PU, and finds there is neither a transmission constraints nor the need for additional transmission projects for Farley for the purpose of MUR-PU.

3.2.3.2.7 Station Blackout

The licensee reviewed the potential impact from the MUR-PU on the licensing basis for station blackout (SBO) as described in Section V, Attachment 4 of the LAR. The licensing basis include the capability to maintain (1) the coping duration, (2) core cooling, (3) power to Class 1E battery chargers, (4) a source of compressed air, (5) required ventilation, (6) containment integrity, and (7) adequate RCS inventory.

The minimum acceptable SBO coping duration of 4 hours was based on offsite power design characteristics, emergency AC power configuration group, and targeted emergency diesel generator reliability. Those three factors are not affected by the implementation of the MUR-PU, therefore, the coping duration is also unaffected.

For the SBO duration, the Farley plant is required by 10 CFR 50.63 to maintain core cooling. The licensee stated that the condensate storage tank (CST) volume requirements in TS 3.7.6 envelope the required storage capacity for dedicated safety grade water for an SBO at MUR-PU conditions, therefore decay heat removal (core cooling) during an SBO under MUR-PU conditions will continue to be accomplished.

The licensee selected the alternate AC (AAC) approach for coping with an SBO event and dedicated Class 1E EDG 2C as the AAC power source which will be available in ten minutes after initiation of an SBO. Station battery chargers aligned to the station's Class 1E batteries are powered by that AAC source. The licensee also stated there are no required changes to the station's DC or AAC power systems for operation at MUR-PU conditions. Therefore, there will be no effect on Class 1E battery availability and the ability to power the station’s battery chargers during an SBO.

According to Section 8.3.1.1.7.3 of the Farley UFSAR, the initiating event for an SBO is assumed to be a loss-of-site power (LOSP) at a plant site. At a multiunit site such as Farley, the LOSP is assumed to affect all units, while the SBO is assumed to occur in only one unit. Farley has five EDGs - two in Unit 1 and two in Unit 2 with EDG 2C being the AAC power source. The two EDGs in the unit experiencing the SBO and one of the two EDGs in the non-SBO unit are assumed to fail. The remaining operable EDG in the non-SBO unit is used to assist in the shutdown of that unit, and EDG 2C (AAC power source) is used solely for the shutdown of the SBO unit. In the following discussion for an SBO at MUR-PU conditions, AC power needs for the SBO unit are addressed by the AAC power source.

The licensee stated that an air compressor can be manually aligned to an engineered safety feature bus powered by the AAC source to supply air-operated valves necessary for safe shutdown of a unit experiencing an SBO event. Section V, Attachment 4 of the LAR also states there are no required changes to the compressed air or AAC systems for operation at MUR-PU conditions. Therefore, the licensee concluded that the MUR-PU will have no effect on the ability to provide a source of compressed air during an SBO.

Heating, ventilation, and air conditioning (HVAC) systems relied upon to provide ventilation to a unit experiencing an SBO are powered by the AAC source. The licensee stated there are no required changes to the station's HVAC or AAC systems for operation at MUR-PU conditions. Thus, the licensee concluded the MUR-PU has no impact on either its ability to cope with the effects nor the licensing basis for coping with a loss of ventilation during an SBO.

The AAC source powers at least one division of containment isolation valves in order to maintain containment integrity in the unit experiencing an SBO. The licensee concluded there are no required changes to the designated containment isolation valves, AC power system, or AAC power source for operation at MUR-PU conditions. Section V, Attachment 4 of the LAR concludes the MUR-PU has no impact on both the ability to maintain containment isolation and the licensing basis for maintaining containment integrity during an SBO.

Pumps from the chemical and volume control system (CVCS) are powered by the AAC source for the unit experiencing SBO. Section V of Attachment 4 of the LAR also states there are no required changes to the station's CVCS or AAC power source for operation at MUR-PU conditions. Therefore, the licensee concluded the MUR-PU has no impact on the requirements of maintaining adequate RCS inventory during an SBO.

The NRC staff finds the licensee's assessment of the effects of MUR-PU on SBO acceptable in that there are no changes in any of the seven areas evaluated by the licensee that warrant further evaluation. The seven areas reviewed by the licensee determine the impact of MUR-PU on station blackout, and in all seven areas reviewed, the current licensing basis is maintained and remains bounding.

3.2.3.2.8 Environmental Qualification Program

3.2.3.2.8.1 Regulatory Evaluation

The regulation in 10 CFR 50.49(e)(1) 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 and following the accident, of which this equipment is required to remain functional.

The regulation in 10 CFR 50.49(e)(4) 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 the accident, of which the equipment is required to remain functional.

The regulation in 10 CFR 50.49(b)(2) requires qualification of nonsafety-related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions specified in subparagraphs (b)(1) (i) (A) through (C) of paragraph (b)(1) of 10 CFR 50 49 by the safety-related equipment.

RIS 2002-03, Section V.1.C, provides guidance to address the scope and detail of the information that should be provided for environmental qualification of electrical equipment.

3.2.3.2.8.2 NRC Staff Evaluation

In accordance with 10 CFR 50.49, the NRC staff reviewed the LAR to ensure that the EQ of electrical equipment remained bounded as a result of the proposed change.

In LAR, Attachment 4, Section II.1.D.iii, "Discussion of RIS 2002-03 Section II.1 Events," Section 29, "Equipment Qualification – FSAR Section 3.11", the licensee noted that its review of the effect on the Farley nuclear equipment qualification analysis as a result of implementing the MUR-PU on the electrical equipment identified in the equipment qualification program for EQ included:

- Containment Pressure and Temperature Analyses
- Containment Flooding
- MSLB in the Main Steam Valve Room (MSVR)
- Other HELBs Pressure/Temperature Outside Containment
- Post- LOCA Sump Water pH
- Radiation Environments to Support Equipment Qualification

The licensee concluded that, the existing AOR remain bounding, and the MUR-PU will not affect equipment in the equipment qualification program for EQ and determined that:

- The equipment and components of the equipment qualification program will continue to operate satisfactorily and perform their intended functions at the uprated conditions to satisfy the requirements outlined in 10 CFR 50.49, and the safety-related electrical equipment is qualified to survive the environment at its specific location during normal operation and during an accident.

- The equipment qualification program equipment will accommodate MUR-PU conditions without exceeding electrical equipment qualification margins for the parameters of temperature, pressure, radiation, and similar parameters, as defined by IEEE Standard 323-1974.

Section V.1.C, Attachment 4 of the LAR, “Environmental qualification of electrical equipment,” similarly re-stated the above description of the evaluation and conclusions. This section further stated that “[t]he evaluations determine that there is no impact on the existing analyses or changes to equipment qualification areas and, therefore, the existing analyses remain bounding and the MUR-PU will not affect equipment in the equipment qualification program for EQ.”

According to LAR, Attachment 4, Table 11.1-1, “FSAR Accidents, Transients, and Other Analyses,” the NRC staff notes that the following accidents/transients analyzed previously in FSAR remain bounding for the MUR-PU: major reactor coolant system pipe rupture (LOCA), containment analyses, flooding, high energy line break outside containment, and major secondary system pipe rupture.

Furthermore, according to the evaluation performed in Section II.1, for flooding (item 30), (NRC staff evaluation found in SER section 3.1.4.2.5) and main steam line break in the Main Steam Valve Room (MSVR) (item 33), (NRC staff evaluation found in SER section 3.1.4.2.6) the NRC staff notes that the current analyses remain valid and unaffected by MUR-PU. According to Section VI.1.B.iii, Attachment 4 of the LAR, “ECCS Recirculation Sump pH Control System,” the NRC staff finds that the current sump pH calculations remain valid for the MUR-PU. Therefore, the NRC staff finds that the existing AORs remain bounding, and the MUR-PU will not affect the qualification of electrical equipment in Farley’s EQ program.

Regarding temperature, pressure, radiation, and humidity, the NRC staff reviewed the licensee’s LAR to determine whether the licensee adequately addressed the impact of the proposed change on the EQ of electrical equipment inside and outside of containment (e.g., Auxiliary Building, MSVR, etc.). For the containment and the MSVR, the NRC staff confirmed that the current design basis analyses were performed at 102 percent of 2775 MWt (i.e., 2831 MWt), which bounds the MUR-PU, and that there is no impact on the EQ of electrical equipment with respect to temperature or pressure due to the MUR-PU in these areas. The NRC requested additional information to confirm whether the existing accident analyses for all areas of the plant were performed at 102 percent RTP versus being limited to inside containment and the MSVR (e.g., temperature/pressure profiles, radiation dose calculations, etc.). The NRC staff also requested additional information to determine that electric equipment in other areas of Farley will remain qualified due to the proposed change during normal operation and accident conditions. In its letter dated April 22, 2020, the licensee clarified that the conditions used in Farley EQ Program are based on accident analyses (FSAR Appendix 3K), bound the MUR-PU thermal power level and that these conditions are applied to other areas of the plant (MSVR as well as feedwater pipe chase) and are not limited to inside containment. The NRC staff reviewed the additional information provided by the licensee and determined that there is no impact on the EQ of electrical equipment with respect to temperature, pressure, radiation, and humidity due to the MUR-PU in all other areas of the plant.

According to 10 CFR 50.49(b)(2), certain non-safety-related electric equipment is required to be considered for EQ. The NRC staff requested the licensee to provide additional information to determine that non-safety-related electric equipment subject to 10 CFR 50.49(b)(2) will remain qualified due to the proposed change. In its letter dated April 22, 2020, the licensee stated that the MUR-PU conditions are bounded by existing evaluations (UFSAR Appendix 3K) and therefore, there is no impact on qualified non-safety-related equipment subject to 10 CFR 50.49(b)(2). The NRC staff reviewed the additional information and finds that there is no EQ impact with respect to non-safety-related electric equipment due to the MUR-PU. Based on its review of the information provided by the licensee, the NRC staff also confirmed that no areas transition from mild to harsh due to the proposed change.

3.2.3.2.8.3 NRC Staff Conclusion Regarding Environmental Qualification of Electrical Equipment

The NRC staff concludes that the MUR-PU will not adversely impact the EQ of electrical equipment at Farley since the existing qualification for the normal and accident conditions for electrical equipment inside and outside containment remain adequate and bounded by the MUR-PU conditions. Therefore, the NRC staff concludes there is reasonable assurance that the proposed MUR-PU will have no adverse impact on the Farley EQ Program as it continues to comply with the requirements of 10 CFR 50.49.

3.2.3.3 NRC Staff Conclusion Regarding Electrical Engineering

The NRC staff reviewed the licensee's technical evaluations described above and, based on that information and the licensee’s April 22, 2020, response to the staff’s RAI, the NRC staff finds that Farley will continue to meet the requirements of 10 CFR 50.49, 10 CFR 50.63, and GDC 17. Therefore, the NRC staff concludes that the LAR is acceptable with respect to electrical engineering evaluations.

3.2.4 Chemical Engineering and Steam Generator Integrity

In the LAR, the licensee proposed revisions to several TSs, as well as the current licensed thermal power (CLTP) to increase it from 2775 MWt to 2821 MWt. The change in CLTP was evaluated for potential impacts on flow accelerated corrosion (FAC), protective coatings in containment SG inspection frequency and wear, SG fatigue, and SG blowdown and secondary chemistry.

3.2.4.1 <u>Chemical Volume and Control System</u>

3.2.4.1.1 <u>Regulatory Evaluation</u>

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 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 SRP, Section 9.3.4, "Chemical and Volume Control System (PWR) (Including Boron Recovery System)" (Reference 45).

3.2.4.1.2 Licensee Description

In the supplement dated April 22, 2020, the licensee stated that no significant changes in bulk chemistry of the primary side are expected due to the MUR-PU. The licensee also stated that primary chemistry will be controlled consistent with Electric Power Research Institute (EPRI) document 300200505, "Pressurized Water Reactor Primary Water Chemistry Guidelines," Revision 7 (Reference 46), and the site strategic water chemistry plan. The strategic water chemistry plan includes the EPRI guidance as well as EPRI Action Level 1, 2, and 3 descriptions and limits for chlorides, fluoride, and oxygen.

3.2.4.1.3 Staff Evaluation and Conclusion

The NRC staff finds the CVCS design at the proposed MUR-PU conditions acceptable because the CVCS is designed with a temperature control valve that can divert bypass flow around the demineralizer if the temperature of the water exceeds the demineralizer temperature limit. The power uprate conditions are not expected to have a significant impact on primary water quality, and, therefore, the NRC staff has reasonable assurance ability of the CVCS to purify the primary water will not be impacted. The licensee also stated that the Farley strategic water chemistry program is consistent with the EPRI PWR Water Chemistry Guidelines, including Action Levels and limits on chlorides, fluorides, and oxygen. Additionally, as per Section 18.2.3, "Water Chemistry Control Program," of the UFSAR, the licensee's water chemistry program follows EPRI PWR Water Chemistry Guidelines. Therefore, the NRC staff has reasonable assurance that GDC 14 will continue to be met at the proposed MUR-PU conditions and that the LAR is acceptable with respect to its review of the CVCS.

3.2.4.2 Steam Generator Blowdown System

3.2.4.2.1 Regulatory Evaluation

Control of secondary-side water chemistry is important for preventing degradation of SG tubes. The SG blowdown system (SGBS) provides a means for removing SG secondary-side impurities design basis of the SGBS that 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. Additionally, the SGBS is described in Section 10.4.8, "Steam Generator

Blowdown Processing System," and secondary water chemistry controls are described in Section 10.3.5, "Water Chemistry," of the Farley UFSAR. Section 18.2.3, "Water Chemistry Control Program," of the UFSAR states that, during the period of extended operation the primary and secondary water chemistry will be based on the EPRI guidelines.

3.2.4.2.2 Technical Evaluation

Licensee Description

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

NRC Staff Evaluation

The NRC staff reviewed the impacts from the proposed MUR-PU on the SGBS and the effects on secondary water chemistry control. The NRC staff also reviewed UFSAR Sections 10.3.5, 10.4.8, and 18.2.3. The NRC staff finds the impacts from the proposed MUR-PU 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 UFSAR Section 18.2.3. 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 reasonable assurance that the SGBS will be able to achieve the necessary flow rates to maintain secondary water chemistry.

3.2.4.2.3 Conclusion

The NRC staff reviewed the licensee's evaluation of the MUR-PU implementation on the SGBS and finds that the licensee has adequately addressed changes in system flow and impurity levels and their effects on the SGBS. The NRC staff finds that the licensee has demonstrated reasonable assurance that the SGBS will continue to meet the requirements of 10 CFR 50 Appendix A, GDC 14, following implementation of the MUR-PU and that the LAR is acceptable with respect to the SGBS.

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

3.2.4.3.1 Regulatory Evaluation

The SG tubes constitute a large part of the RCPB. As a result, their integrity is important to the safe operation of a reactor. The NRC staff's review in this area covered the effects of changes in operating conditions resulting from the proposed MUR-PU on SG materials and the SG program. The NRC staff's review criteria for the SG Program are based on the Farley TSs. Farley TS 3.4.17, "Steam Generator (SG) Tube Integrity," 5.5.9, "Steam Generator (SG) Program," and TS 5.6.10, "Steam Generator (SG) Tube Inspection Report," govern the SG inspections for Farley. Details of the Farley SGs can be found in UFSAR Section 5.5.2, "Steam

Generator." Specific review criteria for this topic are contained in the SRP, Section 5.4.2.1, "Steam Generator Materials," (Reference 47) for the SG materials, and Section 5.4.2.2, "Steam Generator Program," (Reference 48) for the SG program. Additionally, RIS 2002-03 recommends the licensee provide a discussion regarding the impacts of the MUR-PU on the structural integrity of the SG tubes, secondary side internal support structures, shell, and nozzles.

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

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

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

> ... 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.

3.2.4.3.2 Technical Evaluation

Licensee Description

The Farley replacement SGs are Westinghouse Model 54F SGs. Each SG contains 3592 thermally treated SB163 Ni-Cr-Fe (Alloy 690) tubes. In its LAR the licensee stated that it evaluated the change in operating conditions due to the MUR-PU with respect to the structural limits of various SG components. The licensee described the operating parameters that will change at the MUR-PU conditions and stated that the design conditions of the SGs bound the MUR-PU conditions.

NRC Staff Evaluation

The NRC staff evaluated the material provided by the licensee and determined that the changes in operating conditions at the proposed MUR-PU conditions would be relatively small. The changes in operating conditions for the proposed MUR-PU are described in Table 3.2-1 of the LAR and assume 0 percent, 15 percent, and 20 percent tube plugging. Further, the new operating temperatures and pressures are typical of those used by other plants. Similar SGs have operated successfully under these conditions.

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

With respect to the SG program, the NRC staff finds that the changes in operating conditions have no effect on the ability to implement the SG program. As a result, the NRC staff finds 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 finds that the SG program remains acceptable since it requires the licensee to continue to ensure tube integrity for the operating interval between inspections.

In addition, as evaluated in Section 3.2.4.4 of this SE, the NRC staff reviewed the licensee's evaluation of the impact of the power uprate on SG tube vibration and fatigue and finds that it remains within acceptable limits for safe operation.

With respect to the SG program, the NRC staff finds that the changes in operating conditions have no effect on the ability to implement the SG program. As a result, the NRC staff finds 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 finds that the SG program remains acceptable since it requires the licensee to continue to ensure tube integrity for the operating interval between inspections.

With respect to the tube repair criteria included in the TSs for the SG program, the small changes in operating conditions are expected to have a relatively small effect on the structural limits for the tubes. Since the tube repair criterion is determined from the structural limit, it may also be slightly affected by the MUR-PU conditions. Although this analysis was not reviewed by the NRC staff in detail, the NRC staff finds that the tube repair criteria remain valid under the MUR-PU conditions. This determination is based on NRC staff's approval of repair criteria at other similarly designed and operated units and the performance-based requirement to ensure tube integrity for the operating interval between inspections. As a result of the above, the NRC staff finds that the SG program remains acceptable for MUR-PU conditions.

3.2.4.3.3 Conclusion

The NRC staff reviewed the licensee's evaluation of the effect of implementation of the MUR-PU on SG tube integrity and finds 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 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-PU. Therefore, the NRC staff concludes that the LAR is acceptable with respect to the SG tube material and program.

3.2.4.4 Rapidly Propagating Fatigue Cracks in Steam Generator Tubes

3.2.4.4.1 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 SG tube high cycle fatigue is consistent with NRC Bulletin 88-02, “Rapidly Propagating Fatigue Cracks in Steam Generator Tubes,” dated February 5, 1988 (Reference 49), which was addressed to all licensees of Westinghouse designed reactors with steam generators that utilize carbon steel support plates. The Bulletin described a SG tube rupture (SGTR) event at North Anna Unit 1 that was caused by rapidly propagating fatigue cracks due to high cycle fatigue. The Bulletin described actions for the addressees to take in order to minimize the potential for a SGTR such as the one that occurred at North Anna Unit 1.

3.2.4.4.2 Licensee Description

In its LAR, the licensee stated that Farley Units 1 and 2 use Model 54F recirculating steam generators which use stainless steel support plates. The licensee stated that use of stainless steel support plates precludes the tube denting mechanism from occurring. Additionally, the licensee stated that high cycle fatigue is not predicted for tubing with design basis support plates and no tube denting and therefore the fatigue issue raised in NRC Bulletin 88-02 is addressed and the MUR-PU conditions will not cause adverse effects with regards to this issue.

The licensee References Section IV.1.A.vi.c, “SG Tube Wear and FIV Evaluation,” Attachment 4 of its LAR for further discussion of high cycle fatigue for SG tubes. The licensee stated that the impacts of the MUR-PU on the SG tubes due to FIV and tube wear were evaluated. The licensee concluded that the proposed MUR-PU conditions will not result in high wear rate or tube vibration for the general population of the tubes. Additionally, in its supplement dated April 22, 2020, the licensee stated that the stress levels for the SG tubes at the vibration levels calculated remain well below the limit provided in the ASME Code version to which the Farley SGs were designed.

3.2.4.4.3 Staff Evaluation and Conclusion

The NRC staff reviewed the licensee's evaluation of the effect of implementation of the MUR-PU on SG tube integrity with respect to SG tube vibration-induced fatigue. The NRC staff finds that the licensee has adequately addressed impacts to FIV and SG tube wear resulting from the operating condition changes at the MUR-PU condition. The licensee has addressed these mechanisms at up to 20 percent tube plugging and has stated that stress levels for tubes calculated at the vibration levels are well below the limit in the ASME Code version to which the SGs were designed. Therefore, in conjunction with the staff evaluation in Section 3.2.4.3 of this SE, the staff has reasonable assurance that SG tube integrity will be maintained with respect to SG tube high cycle fatigue concerns due to the impacts of the MUR-PU.

3.2.4.5 Flow-Accelerated Corrosion Program

3.2.4.5.1 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 Farley, Units 1 and 2, was developed to be in compliance with NRC Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," dated May 2, 1989 (Reference 50). Additionally, the licensee stated that the program is consistent with NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," Section XI.M17, "Flow-Accelerated Corrosion," (Reference 51) At the time of the submittal of the Farley, Units 1 and 2, license renewal application the initial revision of NUREG-1801 was available for use. Section 18.2.9, "Flow Accelerated Corrosion Program," of the UFSAR, states that the FAC program includes activities to determine susceptible locations, baseline and follow-up inspections of wall thickness and use of predictive modeling techniques.

3.2.4.5.2 Licensee Description

In Section IV.1.E.iii, "Flow Accelerated Corrosion Program," of Attachment 4 to the LAR, the licensee stated that there is an established FAC program at Farley 1 and 2 based on the compliance with GL 89-08 as well as NUREG-1801, Revision 0, Section XI.M17. The licensee also stated that the EPRI CHECWORKS™ predictive software is used to provide a calculated estimate of component wear due to FAC. Changes in operating conditions due to the MUR-PU have been incorporated into the plant FAC model. Additionally, as stated in the letter dated April 22, 2020, the licensee provided wear rate changes for certain systems and components resulting from the proposed MUR-PU. The licensee stated that the increase in wear rates due to the MUR-PU may alter the selection of locations for examination but will not impact the FAC program.

3.2.4.5.3 NRC Staff Evaluation and Conclusion

The NRC staff reviewed the effects of the proposed MUR-PU on FAC and the adequacy of the licensee's FAC program to predict the rate of material loss so that repair or replacement of affected components can be made before reaching a critical thickness. The NRC staff finds the basis for the FAC program acceptable because it was designed for compliance with GL 89-08 and is also consistent with the initial revision of the GALL Report, neither of which are impacted by the MUR-PU. The GALL Report recommends use of EPRI NSAC-202L, "Recommendations for an Effective Flow-Accelerated Corrosion Program," (Reference 52) which the NRC staff found acceptable as a basis for a FAC program. Even though some of the average changes in wear rates are large in percent measure, the actual value of these changes (mils/year) are not relatively large. The licensee also stated that these wear rates will be used to determine remaining service life of susceptible piping and components and determine if these components should be repaired or replaced. Therefore, the NRC staff has reasonable assurance the licensee will be able to monitor and manage wear, repair or replace piping and components prior to reaching the minimum wall thickness.

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

3.2.4.6 Containment Coatings Program

3.2.4.6.1 Regulatory Evaluation

Protective coating systems (paints) provide a means for protecting the surfaces of facilities and equipment from corrosion and contamination from radionuclides and also provide wear protection during plant operation and maintenance activities. The NRC staff reviewed the protective coating systems used inside containment for their suitability for and stability under design basis 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 (SSCs).

RG 1.54, Revision 0, “Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants,” (Reference 53) is discussed in Updated Final Safety Analysis Report Section 6.1.4, “Degree of Compliance with Regulatory Guide 1.54 for Paints and Coatings Inside Containment.” 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.”

As stated in the LAR, RG 1.54 post-dates the Farley, Unit 1 and 2, construction permits. The licensee stated that the containment coatings program meets the intent of American National Standards Institute (ANSI) Standard N101.2-1971, “Protective Coatings (Paints) for Light-Water Nuclear Reactor Containment Facilities,” (Reference 54). The Farley UFSAR Section 6A.8, “Compatibility of Protective Coatings with Post Accident Environment,” discusses the impact of coatings in containment on emergency core cooling system sump strainer performance following a postulated design basis accident. The Farley UFSAR Chapter 3 Appendix A discusses the conformance to RG 1.54 and N101.2-1971.

3.2.4.6.2 Licensee Description

Attachment 4 to the LAR, Section VII.6.B, “Containment Coatings Program,” states that the licensee evaluated the impacts of the MUR-PU on containment coatings program documents and the FSAR. The licensee found the MUR-PU had no effect on these documents, and, therefore, the containment coatings program for Service Level I coatings continues to comply with plant licensing bases and meet the intent of ANSI N101.2.

The licensee concluded that because the DBA pressure, temperature, and dose analyses are bounding for the MUR-PU conditions, Service Level I coatings that are currently DBA-qualified remain qualified and are bound by the power uprate conditions. The licensee, therefore, concludes that no changes are needed to its coatings program.

3.2.4.6.3 NRC Staff Evaluation and Conclusion

The NRC staff reviewed the information provided by the licensee and the UFSAR with regards to MUR-PU impacts to the containment coatings program. Because the proposed conditions in containment after a DBA LOCA due to the MUR-PU are unchanged and therefore bounded by the current analyses and the proposed post-DBA LOCA conditions in containment are bounded by current analyses, the coating qualifications continue to bound the predicted conditions in containment after a DBA LOCA at the proposed MUR-PU conditions. Therefore, the NRC staff concludes there is reasonable assurance that the coatings in containment will not be adversely impacted by the power uprate conditions and finds the MUR-PU acceptable with respect to protective coatings. The NRC staff concludes that the protective coatings continue to meet the requirements of 10 CFR 50, Appendix B and ANSI N101.2.

3.2.5 Inservice Inspection Program

The NRC staff reviewed the effects of the implementation of the proposed MUR-PU on the licensee's ISI program.

3.2.5.1 Regulatory Evaluation

The NRC staff's criteria for reviewing the licensee's ISI program are based on the requirements in 10 CFR 50.55a.

3.2.5.2 Technical Evaluation

In Section IV.1.E.i, Attachment 4 of the LAR, the licensee described its evaluation of the impact of the MUR-PU on the ISI program for ASME Class 1, 2, and 3 components at Farley, stating, in part, that:

> The ISI Program is discussed in FSAR Subsection 5.2.8.6.2 and is in compliance with the requirements of 10 CFR 50.55a(g), including that ASME Class 1, 2 and 3 components are examined in accordance with the provisions of the ASME Boiler and Pressure Vessel Code Section XI. The MUR does not affect the classifications or boundaries of these components and therefore does not affect the inspection requirements as described in the ISI plan. The MUR-PU conditions were reviewed for impacts on the ISI Program. Therefore, the ISI Program will continue to comply with the licensing bases of 10 CFR 50.55a(g).

3.2.5.3 NRC Staff Conclusion Regarding the ISI Program

Based on its review of the licensee's evaluations, the NRC staff finds that the ISI program will continue to meet the regulatory requirements of 10 CFR 50.55a upon implementation of the

MUR-PU. Therefore, the NRC staff concludes that the LAR is acceptable with respect to the ISI program.

3.2.6 Inservice Testing Program

3.2.6.1 Regulatory Evaluation

In its submittal, the licensee described the review of the Inservice Testing (IST) Program for safety-related pumps and valves at Farley during MUR-PU operation. The Code of Record for Farley is the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code) 2004 Edition through the 2006 Addenda in compliance with the requirements of the 10 CFR 50.55a(f).

3.2.6.2 Safety-Related Valves

The NRC staff reviewed the licensee's safety-related valves analyses for Farley. The NRC staff examined the overall design change and included plant specific evaluations of Generic Letter(s) (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," (Reference 55) GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves," (Reference 56)95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," (Reference 57) and GL 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions," (Reference 58) The NRC's acceptance criteria for reviewing the safety-related valve analysis are based on *Title 10 of the Code of Federal Regulations* (10 CFR) 50.55a, "Codes and Standards."

In the submittal dated October 30, 2019, the licensee reviewed the impact of the proposed MUR-PU on the existing safety-related valves design basis analysis. No changes in RCS flow, design, or operating pressure were made as part of the power uprate. The licensee's evaluations concluded 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-PU. The analyses also confirmed that the existing main steam safety valves capacity is adequate for overpressure protection at MUR-PU conditions and that the existing lift setpoints are unchanged. The NRC staff reviewed the licensee's analysis and determined that none of the safety-related valves required a change to their design or operation as a result of the MUR-PU.

The licensee also evaluated the impact of the proposed MUR-PU on the current air-operated valve (AOV) program, GL 89-10 and GL 96-05 motor-operated valve (MOV) program, and GL 95-07 pressure locking/thermal binding (PLTB) program. The overall system evaluations concluded that valve function, valve design, operational conditions, thrust, and torque requirements are unaffected by the MUR-PU and all valves remain capable of performing their design basis functions. Therefore, no changes are required to the existing AOV, MOV, and PLTB programs. Based on the licensee's evaluations, the NRC staff concludes there is reasonable assurance that the performance of existing safety-related valves is acceptable with respect to the MUR-PU.

3.2.6.3 Reactor Coolant Pumps

The NRC staff reviewed the effects of the implementation of the proposed MUR-PU on the licensee's RCP analysis.

3.2.6.3.1 Regulatory Evaluation

The NRC staff's criteria for reviewing the RCP analysis is based on the requirements in 10 CFR 50.55a.

3.2.6.3.2 Technical Evaluation

In Section IV.1.A.vii of its LAR, the licensee evaluated the impact of the proposed MUR-PU conditions on the existing design basis analyses for the RCPs. There is no change to the current reactor coolant pressure operating pressure of 2250 psia. The maximum MUR RCS cold leg temperature is less than the temperature evaluated in the AOR. NSSS primary side design transients were determined to be unaffected by the MUR-PU. Further, the loads applied to the RCP casing nozzles and support feet have not changed due to MUR-PU conditions.

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 RCP performance. The current plant design is considered bounding and requires no modifications to the RCPs.

3.2.6.3.3 NRC Staff Conclusion Regarding the RCP

Based on its review of the licensee's evaluations, the NRC staff finds reasonable assurance that the performance of the RCPs will continue to meet the regulatory requirements of 10 CFR 50.55a upon implementation of the MUR-PU. Therefore, the NRC staff concludes that the LAR is acceptable with respect to the RCP analysis.

3.2.6.4 NRC Staff Conclusion

The IST program assesses the operational readiness of pumps and valves within the scope of the ASME OM Code. There were no significant changes to operating conditions or the design basis requirements that would affect component performance, test acceptance criteria, or Reference values. Therefore, the existing IST program will not be impacted by the MUR-PU. Based on the licensee’s evaluation, the NRC staff concludes that the IST program provides reasonable assurance of acceptable SSC performance under MUR-PU conditions.

3.2.7 Inservice Examination and Testing Program for Snubbers (Dynamic Restraints)

In Farley submittals dated April 22, 2020, and May 8, 2020, the licensee described the review of the Inservice Examination and Testing Program for safety-related snubbers at Farley during MUR-PU operation. The Code of Record for Farley is the American Society of Mechanical Engineers (ASME) Code for Operation and Maintenance of Nuclear Power Plants (OM Code) 2004 Edition through the 2006 Addenda in compliance with the requirements of the regulation 10 CFR 50.55a. The inservice examination and testing snubber program assesses the operational readiness of snubbers within the scope of the ASME OM Code. There were no significant changes to operating conditions or the design basis requirements that would affect component performance, test acceptance criteria, or Reference values. Therefore, the existing snubber program will not be impacted by the MUR-PU.

The NRC staff reviewed the licensee's safety-related snubbers impacted by load increases due to MUR-PU. The NRC's acceptance criteria for reviewing the safety-related snubbers is comparing the existing design load and the actual load due to MUR-PU.

The NRC staff reviewed the impact of the proposed MUR-PU conditions on the existing design of the safety-related snubbers. In letter dated April 22, 2020, the license stated that the only safety-related snubbers impacted by the MUR are in the main steam system for Farley. The snubbers have load increases associated with increased turbine trip fluid transient loads related to the increased flow rates in the main steam piping due to MUR. There are five safety-related snubbers installed on main steam piping on Farley Unit 1 and three safety-related snubbers installed on main steam piping on Farley Unit 2. All eight safety-related snubbers are evaluated and determined to have adequate margin and are acceptable for post-MUR conditions.

3.2.7.1 NRC Staff Conclusion

Based on the licensee's evaluation, the NRC staff concludes that the existing snubber program will not be impacted by the MUR-PU. The NRC staff concludes that the increased load for the main steam safety-related snubbers is still within the design load and remains acceptable for post-MUR conditions.

3.3 Safety Programs

3.3.1 Radiological Dose Assessment

3.3.1.1 Regulatory Evaluation

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

For the radiation protection-related sections, the NRC staff conducted an evaluation to verify that annual doses are within the applicable 10 CFR 20 annual limit of 100 mrem (millirem), 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). For the DBA-related section, the NRC staff conducted an evaluation to verify that the results of the licensee's DBA radiological consequence analyses continue to meet the dose acceptance criteria given in 10 CFR 100.11; 10 CFR 50.67; SRP 15.0.1; and RG 1.183 at the proposed uprated power level.

3.3.1.1.1 Radiation Protection Section

The NRC's acceptance criteria for normal occupational and public doses are based on: (1) 10 CFR Part 20, *Standards for Protection Against Radiation,* insofar as it establishes requirements for radioactivity in liquid and gaseous effluents released to unrestricted areas; (2) 10 CFR Part 50, Appendix I, *Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low as is Reasonably Achievable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents*, insofar as it establishes numerical guides for design objectives and limiting conditions for operation to meet the "as low as is reasonably achievable" (ALARA) criterion; (3) final GDC-60, "*Control of* *releases of radioactive materials to the environment,"* insofar as it requires that the plant design include means to control the release of radioactive effluents; and, (4) Item II.B.2 of NUREG 0737, *Clarification of TMI Action Plan Requirements,* insofar as it relates to plant shielding for spaces/systems which may be used in post-accident operations. Specific review criteria are contained in Section 11.1, *Coolant Source Terms,* of NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (SRP)*.

3.3.1.1.2 Design Basis Accident

The regulatory requirements and guidance which the NRC staff considered in its review are as follows:

Title 10 of the *Code of Federal Regulations* (10 CFR) 100.11, *Determination of exclusion area, low population zone, and population center distances*, states:

> (a) As an aid in evaluating a proposed site, an licensee should assume a fission produce release[1][3] 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 licensee 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[2][4] or a total radiation dose in excess of 300 rem[2][4] 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.

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[3] 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.

[4] 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 recommendations may be disregarded in the determination of their radiation exposure status (see NBS 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.

10 CFR 50.67, *Accident source term,* states that:

> (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 (25 rem) total effective dose equivalent (TEDE).
>
> (ii) An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) 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) TEDE for the duration of the accident.

Appendix A to 10 CFR Part 50, GDC 19, *Control room*, 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 [(licensees)] under this part who apply on or after January 10, 1997, applicants [licensees] 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 [(licensees)] 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 § 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 § 50.2 for the duration of the accident.

3.3.1.1.3 Other Regulatory Guidance

Regulatory Guide 1.183, *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*, Rev. 0, July 2000 (Reference 23) provides the methodology for analyzing the radiological consequences of several DBAs to show compliance with 10 CFR 50.67. Regulatory Guide 1.183 provides guidance to licensees on acceptable application of alternate source term submittals, including acceptable radiological analysis assumptions for use in conjunction with the accepted alternate source term.

RIS 2002-03 recommends that to improve efficiency of the NRC staff's review, licensees requesting a MUR-PU should identify existing DBA AOR, which bound plant operation at the proposed uprated power level. For any existing DBA AOR that do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the re-analysis. This safety evaluation documents the NRC staff's review of the impact of the proposed changes on analyzed DBA radiological consequences.

License Amendment Numbers 216 and 213, dated December 20, 2017, Joseph M. Farley Nuclear Plant, Units 1 and 2 – Issuances of Amendments Adopting Alternative Source Term, TSTF-488, Revision 3, and TSTF-312, Revision 1 (CAC Nos. MF8861, MF8862, MF8916, MF8917, MF8918, AND MF8919; EPID NOS. L-2016-LLA-0017, L-2016-LLA-0018, AND L-2016-LLA-0019), (Reference 59) approved a full-scope implementation of the Alternative Source Term 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.

3.3.1.2 Technical Evaluation

3.3.1.2.1 Radiation Protection Section

3.3.1.2.1.1 Section VI.1.E, Attachment 4 of the LAR, "Radioactive Waste Systems"

The NRC staff reviewed the radioactive waste systems (Liquid Waste, Gaseous Waste, and Solid Waste) are described in FSAR Chapter 11, Sections 11.2, 11.3, and 11.5, respectively. These systems provide the means to sample, collect, process, store/hold, re-use or release low-level effluents generated during normal operation.

3.3.1.2.1.1.1 Liquid Waste Processing System

The liquid waste processing system is designed to receive, segregate, process, recycle, and discharge liquid wastes. The system design considers potential personnel exposure and ensures that quantities of radioactive releases to the environment are ALARA.

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. A permanent record of liquid releases is provided by analyses of known volumes of waste.

As discussed by the licensee, the existing capacities of various holding, processing, and storage tanks are sufficient at MUR-PU conditions because system flow rates and liquid inventories are not affected by the uprate. The volume of liquid waste primarily depends on reactor coolant bleed, steam generator blowdown, and leakage from various components. The volume generated during normal operation will not change because of the uprate. Implementing the MUR-PU 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 1.7 percent. This increase in nuclide concentration does not significantly impact liquid waste processing system operation.

3.3.1.2.1.1.2 Gaseous Waste Processing System

The gaseous waste processing system is designed to remove fission product gases from the reactor coolant and has the capacity to contain these throughout the 40-year plant life. This is based on continuous operation with reactor coolant system activities associated with operation, with cladding defects in the fuel rods generating 1 percent of the rated core thermal power. The system is also designed to collect and store expected fission gases from the boron recycle evaporator and reactor coolant drain tank throughout the plant life.

The gaseous waste processing system consists mainly of a closed loop comprised of two waste gas compressors, two catalytic hydrogen recombiners, and gas decay tanks to accumulate the fission product gases. 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-PU 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-PU does not materially affect the system flow rates or gas volumes. The MUR-PU may increase radioactivity of gaseous waste a maximum of 1.7 percent. However, as discussed by the licensee, system operating procedures can support the potential increase in radioactivity.

Although the gaseous waste processing system was designed for continuous purge of the volume control system and 40-year holdup of fission gases, operating experience at Farley has shown that the gaseous waste processing system can be operated without a continuous purge while maintaining personnel exposure within limits and maintaining releases within concentration and offsite dose limits.

3.3.1.2.1.1.3 Solid Waste Processing System

The solid waste processing ystem is designed to transfer spent resins, evaporator concentrates, and chemical tank effluents. This system is installed in Unit 1 and has adequate capacity to serve both units. To provide more efficient solidification and to ensure compliance with current burial ground license requirements (including volume restrictions), provision has been made for the use of a portable cement solidification system. The portable system is operated in the solidification / dewatering facility outside the Unit 1 and Unit 2 auxiliary building and is capable of solidifying resins, evaporator concentrates, and chemical drains from both units. The system also serves as a solidification system for the disposable demineralizer system, should solidification be required prior to shipment. A separate system is available to compact dry active wastes such as paper, disposable clothing, rags, towels, floor coverings, shoe covers, plastics, cloth smears, and respirator filters.

Bulk waste may be shipped to a licensed waste processor or to a disposal facility without encapsulation or solidification in accordance with regulations and per applicable license and regulations for the receiver of the waste. During normal work activities, tools, scrap, and other miscellaneous equipment and materials may become radioactively contaminated. The solid waste system can be used as a decontamination area when needed.

Solidification via the portable system is accomplished with the liner inside a shipping cask or a shielded enclosure in the solid waste system, which provides the necessary personnel shielding. The MUR-PU does not affect the generation of solid waste volumes. Therefore, the quantities of low-level, compressible, radioactive wastes (e.g., paper, rags, plastics, clothing, respiratory filters) will not increase because of the uprate. The same is true for high-level wastes such as spent resins and filters. Procedures are in place to segregate, store, classify, package, and track low-level and high-level solid wastes; and because there is no increase in solid waste generation due to uprate, there is not an increase in solid waste storage requirements or shipments from the plant.

3.3.1.2.1.2 Section VII.5.A.ii, Attachment 4 of the LAR, “Radiological Effluents”

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

The core isotopic inventory is a function of the core power level. The reactor coolant isotopic activity concentration is a function of the core power level, the migration of radionuclides from the fuel, radioactive decay and the removal of radioactive material by coolant purification systems. Radiation sources in the reactor coolant include activation products, activated corrosion products and fission products. During reactor operation, some stable isotopes in the coolant passing through the core become radioactive (activated) as a result of nuclear reactions. For example, the non-radioactive isotope oxygen-16 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.

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

As discussed by the licensee, during normal operations, the controls for the release rates of radwaste systems do not change with operating power. Thus, no impact on routine releases is anticipated due to the MUR-PU. 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 are ALARA. The MUR-PU 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-PU. Therefore, the NRC staff concludes that the licensee's analysis provides reasonable assurance that the proposed license amendment is acceptable with respect to the radiological effluents from radwaste systems.

3.3.1.2.1.3 VII.5.A.iii A, Attachment 4 of the LAR, "A discussion of the effect of the power uprate on individual or cumulative occupational radiation exposure"

Farley Nuclear Plant was designed with sufficient margin for higher-than-expected radiation sources. During normal and post-accident conditions, radiation levels in most areas of the plant increase by no more than the percentage increase in power level. The licensee reviewed the radiological effects for the uprate, using current licensing basis methodologies, to verify that expected coolant concentrations at the MUR power levels will be bounded by the existing analyses of record at 102 percent of the currently licensed thermal power. Due to the design of the shielding and containment surrounding the reactor vessel, and since the reactor vessel is inaccessible to plant personnel during operation, an increase in the radiation sources in the reactor core over the original licensed thermal power level, will have no effect on occupational worker personnel doses during power operations.

Similarly, the radiation shielding provided in the balance of plant is conservatively sized such that the increased source terms discussed above are not expected to significantly increase the dose rates in the normally occupied areas of the plant. In addition, occupational exposure is controlled by the plant radiation protection program and is maintained within limits required by regulations. Thus, the increase in radiation levels does not affect radiation zoning or shielding in the various areas of the plant because it is offset by conservatism in the design, source terms, and analytical techniques. Therefore, no change is required in the design basis radiation protection design features for the MUR uprate.

The current ALARA program practices at FNP, such as work planning and source term minimization, coupled with existing radiation exposure procedural controls, will be able to compensate for the anticipated increases in dose rates associated with this MUR. Therefore, the increased radiation sources resulting from this proposed MUR, as discussed above, will not adversely impact the licensee's ability to maintain doses resulting from plant operation with the applicable limits in 10 CFR 20 and ALARA, and is therefore considered acceptable. Therefore, the NRC staff concludes that the licensee's analysis provides reasonable assurance that the proposed license amendment is acceptable with respect to the effect of the power uprate on individual or cumulative occupational radiation exposure.

3.3.1.2.2 Design Basis Accident Dose Analyses

The NRC staff reviewed the regulatory and technical analyses performed by the licensee in support of its proposed MUR-PU 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 a MUR-PU should first identify existing DBA AOR, which bound plant operation at the proposed uprated power level. Secondly, for any existing DBA AOR that do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the re-analysis. Information regarding these analyses was provided by the licensee in the Section II and Section III, Attachment 4 of the LAR, consistent with Regulatory Issue Summary 2002-03. The findings of this safety evaluation are based on the descriptions and results of the licensee's analyses and other supporting information docketed by the licensee.

The Farley accident and safety analyses including loss of coolant accident events are addressed in Chapter 15 of the Farley UFSAR.

In Section II and Section III, Attachment 4 of the 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-PU. As previously discussed, the MUR LAR would increase the Farley authorized core power level from 2,775 MWt to 2,821 MWt which is an increase of approximately 1.7 percent rated thermal power, based on the use of the CheckPlus™ LEFM system. In accordance with current licensing basis which incorporates the full implementation of AST, and as documented in UFSAR 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. 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 2,831 MWt (102 percent of the current RTP of 2,775 MWt), which bounds operation at MUR-PU operating conditions.

The NRC staff reviewed the impact of the proposed 1.7 percent MUR-PU on DBA radiological consequence analyses, as documented in Chapter 15 of the UFSAR. The NRC staff confirmed that the current licensing basis dose consequence analyses remain bounding at the proposed MUR uprated power level of 2,821 MWt with a margin that is within the assumed uncertainty associated with advanced flow measurement techniques, including use of the CheckPlus™ LEFM system credited by the licensee. Specific areas of review include:

- Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accidents) - FSAR Section 15.4.1
- Fuel Handling Accident - FSAR Section 15.4.5
- Major Secondary System Pipe Rupture - FSAR Section 15.4.2
- Steam Generator Tube Rupture - FSAR Section 15.4.3
- Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) - FSAR Section 15.4.6
- Single Reactor Coolant Pump Locked Rotor - FSAR Section 15.4.4
- Waste Gas Decay Tank Rupture - FSAR Section 15.3.5

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.

The LEFM 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 to switch from the normal operation to the maintenance mode or to the fail mode. Normal operation for the LEFM system is the check plus mode. The control room operators are provided a visual alarm on the plant processing computer when the LEFM system shifts from check plus mode (normal mode) to the check mode (maintenance mode). The visual alarm is displayed on the operator overview display screen.

The NRC staff confirmed that the applicable current licensing basis dose consequence analyses remain bounding at the proposed MUR uprated power level of 2,821 MWt with a margin that is within the assumed uncertainty associated with advanced flow measurement techniques, including use of the CheckPlus™ LEFM 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 Farley UFSAR, in addition to information in the LAR, the NRC staff verified that the existing radiological analyses and release assumptions bound the conditions for the proposed MUR-PU, considering the higher accuracy of the proposed feed water flow measurement instrumentation.

3.3.1.3 NRC Staff Conclusion Regarding Radiological Dose Assessment

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 Farley at the proposed uprated power level will continue to meet the applicable dose limits following implementation of the proposed 1.7 percent MUR-PU. The NRC staff further finds reasonable assurance that Farley, 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. Therefore, the NRC staff concludes that the licensee's analysis provides reasonable assurance that the proposed license amendment is acceptable with respect to the source terms for radwaste systems and the radiological dose consequences of DBAs.

3.3.2 Fire Protection

The purpose of the fire protection program established by NFPA 805 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 during any operational mode or plant configuration. The NRC staff's review focused on the effects of the increased decay heat due to implementation of the proposed MUR-PU on the plant's nuclear safety capability assessment[5] (NSCA) on the plant's safe shutdown analysis to ensure that the 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.

3.3.2.1 Regulatory Evaluation

The NRC's review criteria for the fire protection program are based on 10 CFR 50.48, "Fire protection"; 10 CFR 50 Appendix A, 10 CFR Part 50, Appendix K, effective July 31, 2000, GDC 3, "Fire protection"; and GDC 5, "Sharing of structures, systems, and components."

Regulations in 10 CFR 50.48 requires the development of a fire protection program to ensure, among other things, the capability to safely shutdown the plant.

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[5] For plants that have transitioned to NFPA 805, the safe-shutdown analysis is often referred to as the NSCA.

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

GDC 5, states: “Structures, systems and components [SSCs] 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.”

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

3.3.2.2 Technical Evaluation

SNC developed the LAR consistent with the guidelines in RIS 2002-03. In the LAR, the licensee re-evaluated the applicable SSCs and safety analyses at the proposed MUR core power level of 2821 MWt against the previously analyzed core power level of 2775 MWt.

The NRC staff reviewed Attachment 4: Summary of RIS 2002-03 Requested Information for Farley Nuclear Plant License Amendment Request, Section VII.6.A, “Fire Protection Program,” of the FNP specific evaluations of each item outlined in RIS 2002-03 in the LAR. The NRC staff also reviewed the licensee’s commitment to 10 CFR 50.48, “Fire protection,” i.e., the approved fire protection program. The review covered the impact of the proposed MUR-PU on the results of the plant’s ability to achieve and maintain the Nuclear Safety Performance Criteria (NSPC) or safe-shutdown capability as noted in RIS 2002-03, Attachment 1, Sections II and III. The review focused on the effects of the MUR-PU on the NSPC and increase in decay heat generation following plant trips.

In Attachment 4, Section VII.6.A, “Fire Protection Program,” of the LAR, the licensee stated that the fire protection program is based on the NRC's requirements and guidelines, NFPA 805, Nuclear Electric Insurance Limited property loss prevention standards, and related industry standards. The fire protection program design basis contains various References to the calculations that demonstrate compliance to these requirements. The licensee stated that these calculations were reviewed for any impact due to the MUR-PU (i.e., change in temperature, pressure, flow, or reactor power level) and that the reviewed calculations do not consider reactor power level or the operating conditions of supporting systems. The licensee stated that the change in reactor power level and operating conditions of systems do not affect fire protection program compliance with required regulations and guidelines. Therefore, the fire protection program is not impacted by the MUR-PU.

Further, the licensee stated that the there are no fire protection program recommendations, and there are no recommended plant modifications to any fire protection systems required to support the MUR-PU. Any changes to combustible loadings resulting from the installation of equipment and components installed in support of the MUR-PU will be evaluated and controlled under the FNP design change process.
Further, in LAR Attachment 4, Section VII.6.A, “Fire Protection Program,” the licensee stated that in the event of a fire, the program and system designs ensure the capability to shut down the reactor and maintain it in a safe and stable (achieve and maintain safe shutdown) shutdown condition. This includes the following:

- Classical fire protection elements such as fire detection and suppression systems and equipment (including active as well as passive design features) and programmatic / organizational elements for minimizing the chance of fire occurring and ensuring minimal impact should one occur.
- NSCA, safe shutdown capability, and Farley's design capability to achieve nuclear safety capability safe and stable conditions in the event of a single damaging fire.
- A supporting fire probabilistic risk assessment that provides both overall and detailed risk insights.
- Radioactive release.
- Non-power operations.

Based on the licensee’s fire-related safe-shutdown assessment, the NRC staff finds that the licensee has adequately accounted for the effects of the implementation of the MUR-PU on the ability of the required fire protection systems to achieve and maintain safe shutdown conditions. The NRC staff finds that this aspect of the capability of the associated SSCs to perform their design-basis functions after implementation of the MUR-PU is acceptable with respect to fire protection. The information provided in the LAR demonstrates that compliance with the fire protection program and NSPC of NFPA 805 will not be affected because the MUR-PU evaluation did not identify changes to design or operating conditions that will adversely impact the NSPC capability. Further, the NRC staff reviewed the information provided by the licensee concerning the fire protection program elements listed above and finds that the licensee’s analysis provides reasonable assurance that they are not impacted by the MUR-PU.

3.3.2.3 NRC Staff Conclusion Regarding Fire Protection

Based on its review, the NRC staff concludes that the proposed implementation of the MUR-PU will not have a significant impact on the fire protection program, NSPC of NFPA 805, or post-fire safe shutdown capability at Farley, Units 1 and 2. Therefore, the NRC staff concludes that the LAR is acceptable with respect to fire protection.

3.3.3 Human Factors

The NRC staff’s human factors review addresses programs, procedures, training, and plant design features related to operator performance during normal and accident conditions. The NRC staff’s human factors evaluation was conducted to confirm that operator performance would not be adversely affected as a result of system and procedure changes made to implement the proposed MUR-PU.

3.3.3.1 Regulatory Evaluation

Appendix A to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities,” GDC 19, “Control room,” states that a control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Equipment at appropriate locations outside the control room shall be provided: (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

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

NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: [Light-Water Reactor] LWR Edition,” Chapter 13.5.2.1, Revision 2, “Operating and Emergency Operating Procedures,” provides the methodology for the NRC staff’s review of operating procedures that will be used by the operating organization to ensure that routine operating, off-normal, and emergency activities are conducted in a safe manner.

The scope of NRC staff review included licensee-identified changes to operator actions, human-system interfaces, procedures, and training needed for the proposed MUR-PU.

Guidance for NRC staff’s review of the licensee’s human factors evaluation is available in RIS 2002-03, Attachment 1, Section VII, Items 1 through 4.

3.3.3.2 Technical Evaluation

The NRC staff developed a standard set of questions for review of human factors issues associated with the review of MUR-PUs in RIS 2002-03, Attachment 1, Section VII, Items 1 through 4. The licensee’s response to the questions are described in Sections VII.1 through VII.4 of Attachment 4 to the LAR. The following sections are NRC staff’s evaluation of the licensee’s response to the questions.

3.3.3.2.1 Operator Actions

In RIS 2002-03, Attachment 1, Section VII.1: asks, Has the licensee made a statement confirming that operator actions that are sensitive to the power uprate, including any effects on the time available for operator actions, have been identified and evaluated?

NRC Staff Evaluation

SNC stated in its October 30, 2019, submittal, that the power uprate will be implemented under the administrative controls of the SNC design change process. This process ensures that any impacted normal, abnormal and emergency operating procedures having operator actions are revised prior to implementation of the MUR-PU, if required. An evaluation of the operator actions was performed, and no impacts were identified. The LAR states that Time Critical Operator Actions (TCOA) are associated with the mitigation of postulated events. These actions must be performed in a specified time order to assure plant compliance with assumptions made during analysis of the design basis events, regulatory commitments, and events with high Probabilistic Risk Assessment values. These TCOA were evaluated individually in system evaluations against the Farley licensing analyses presented in Section II of the LAR to ensure they remain bounded. SNC states that all TCOAs remain unchanged following the MUR-PU.

The NRC staff reviewed the licensee's statements relating to any impacts of the MUR-PU on operator actions. The NRC staff finds that the proposed MUR-PU will not adversely impact operator actions or their response times because there are no changes required. Therefore, the NRC staff finds that the statements provided by the licensee conform to Section VII.1 of Attachment 1 to RIS 2002-03 and concludes that the licensee's analysis provides reasonable assurance that the proposed MUR power uprate will not adversely impact the licensee-identified operator actions, including the time available for operator actions.

3.3.3.2.2 Emergency and Abnormal Operating Procedures

In RIS 2002-03, Attachment 1, Section VII.2.A: Asks, has the licensee made a statement confirming that it has identified all required changes to the current emergency operating procedures (EOPs) and abnormal operating procedures (AOPs) to ensure that changes to the EOPs and/or AOPs do not adversely affect defense-in-depth or safety margins?

NRC Staff Evaluation

SNC stated in its submittals that the EOPs and AOPs have been reviewed for power uprate impacts, and no impacted procedures have been identified. Additionally, the uprate is being implemented under the administrative controls of the design change process. The design change process ensures any impacted procedures will be revised prior to implementation of the proposed power uprate.

Based on its review, the NRC staff finds that the statements provided by SNC are consistent with Section VII.2.A of Attachment 1 to RIS 2002-03 and concludes that the proposed changes to the emergency and abnormal operating procedures will not adversely affect defense in depth or safety margins.

3.3.3.2.3 Changes to Control Room Controls, Displays, and Alarms

In RIS 2002-03, Attachment 1, Section VII.2.B: asks, Has the licensee made a statement confirming that it has identified all required changes to the control room controls, displays (including the safety parameter display system), and alarms to ensure that any required changes do not adversely affect defense-in-depth or safety margins?

NRC Staff Evaluation

SNC states that the physical modifications of the plant required to support the Farley MUR-PU include the installation of the Cameron "Leading Edge Flow Meter" (LEFM) Check-Plus feedwater ultrasonic measurement system to the new high pressure turbine. Additionally, a review of the plant systems has indicated that only minor modifications are necessary (e.g., software modification that redefines the new 100-percent RTP). Farley follows the established design change process and procedures to ensure that any necessary minor modifications are installed prior to implementing the proposed power uprate. A "LEFM System Trouble" alarm window will be added to the control room alarm panel to alert the operator when there's a problem with the LEFM. The new high pressure turbine modifications require no changes to the control room controls, displays or alarms.

The NRC staff reviewed the licensee's evaluation of the proposed changes to the control room. The NRC staff finds that the proposed changes are minimal and do not present any adverse effects to the operators' functions in the control room. SNC committed to making all modifications to the control room and simulators and providing training on these changes prior to MUR-PU implementation. The NRC staff finds that the statements provided by SNC are consistent with Section VII.2.B of Attachment 1 to RIS 2002-03 and concludes that the proposed changes will not adversely affect defense in depth or safety margins.

3.3.3.2.4 Control Room Plant Simulator

In RIS 2002-03, Attachment 1, Section VII.2.B: Asks, has the licensee made a statement confirming that it has identified all required changes to the control room plant simulator to ensure that any required changes do not adversely affect defense-in-depth or safety margins?

NRC Staff Evaluation

As part of the Farley design change process and procedures, a review of the plant simulator is conducted, and necessary changes resulting from the MUR-PU (including changes for the new calorimetric and displays for the LEFM interface) to the Farley simulator are identified. The design change process ensures that the simulator modifications are made prior to the implementation of the uprate.

The NRC staff reviewed SNC's evaluation of proposed changes to the plant simulator related to the MUR-PU and the changes. SNC has committed to making all modifications to the plant simulator and incorporating these changes into the operator training program prior to MUR-PU implementation. The NRC staff finds that the statements provided by SNC are consistent with Section VII.2.C of Attachment 1 to RIS 2002-03 and concludes that the proposed changes to the control room plant simulator will not adversely affect defense in depth or safety margins.

3.3.3.2.5 Operator Training

In RIS 2002-03, Attachment 1, Section VII.2.C: asks, Has the licensee made a statement confirming that it has identified all required changes to the operator training program to ensure that any required changes do not adversely affect defense-in-depth or safety margins?

NRC Staff Evaluation

SNC states that the Operations Training department has been involved in the design review process for the modifications required to support the MUR-PU. In addition, Operations Staff are trained on the modifications, Technical Specification changes, and procedure changes prior to implementation of the MUR-PU. Training on the operation of the Cameron LEFM Check-Plus system and calorimetric impacts are also developed and completed prior to implementation of the MUR-PU.

The NRC staff reviewed the licensee's evaluation of the proposed changes to the operator training program. SNC will provide training on these changes prior to MUR-PU implementation.

The NRC staff finds that the statements provided by SNC are consistent with Section VII.2.C of Attachment 1 to RIS 2002-03.

3.3.3.2.6 Modifications

In RIS 2002-03, Attachment 1, Section VII.3: Asks, has the licensee made a statement confirming its intent to complete the modifications identified in Section VII.2 of Attachment 1 to RIS 2002-03 (including the training of operators), prior to implementation of the power uprate?
NRC Staff Evaluation

The licensee stated in its submittal that all changes and modifications discussed above (including changes to the simulator and the associated manuals and instructional materials) are implemented in accordance with the Farley engineering design change process. SNC will complete all modifications related to the MUR-PU and complete the training of operators prior to the implementation of the power uprate. Plant modifications are evaluated to ensure that changes in the operator actions do not adversely affect defense-in-depth or safety margins. The NRC staff finds that the statements provided by SNC are in conformance with Section VII.3 of Attachment 1 to RIS 2002-03.

3.3.3.2.7 Temporary Operation above Licensed Full Power Level

In RIS 2002-03, Attachment 1, Section VII.4: asks, Has the licensee made a statement confirming its intent to revise existing plant operating procedures related to temporary operation above “full steady-state licensed power levels” to reduce the magnitude of the allowed deviation from the licensed power level? The magnitude should be reduced from the pre-power uprate value of 2 percent to a lower value corresponding to the uncertainty in power level credited by the proposed power uprate application.

NRC Staff Evaluation

SNC states that the unit operating procedure for Mode 1 includes precautions for temporary operation above the licensed power level for certain periods of time. These precautions will be revised to account for the MUR-PU power level.

The NRC staff finds that the statements provided by SNC are consistent with Section VII.4 of Attachment 1 to RIS 2002-03.

3.3.3.3 NRC Staff Conclusion Regarding Human Factors

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

3.4 Spent Fuel Pool Criticality Analysis

Section II.1.D.iii, Item 31, Attachment 4 of the LAR, states that the spent fuel pool criticality AOR was updated to support the MUR-PU and submitted for NRC approval by letter dated September 30, 2019, (Reference 60). By NRC letter dated October 6, 2020 (Reference 61), the NRC issued Amendment Nos. 229 and 226, for Farley, Units 1 and 2, respectively, approving a new SFP criticality analysis. The licensee intends to use this new SFP criticality AOR in its use of the MUR-PU.

3.5 Changes to the RFOL and Technical Specifications

The licensee requested changes to the TSs for Joseph M. Farley Nuclear Plant Units 1 and 2. Per the LAR, the amendment would revise Farley's TS to allow for a MUR-PU. The licensee states, "This MUR-PU license amendment request (LAR) would increase FNP's authorized core power from 2775 megawatts thermal (MWt) to 2821MWt."

The proposed changes contained herein would revise Operating License (OL) 2.C(1) "Maximum Power Level" and TS Sections 1.1 Definition "Rated Thermal Power (RTP)," 2.1.1 "Reactor Core Safety Limits," 3.4.1 "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling Limits," and 5.6.6 "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)."

3.4.1 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 TSs. Specifically, 10 CFR 50.36(c) requires that TSs 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.

3.4.2 Technical Evaluation

In its LAR, the licensee proposed changes to the RFOL and TS, specifically stating, in part, that:

> <u>Operating License – Maximum Power Level</u>
>
> For the Farley, Units 1 and 2 operating license, the steady state licensed power level will change from 2775 MWt to 2821MWt.
>
> <u>TS 1.1, Definition of Rated Thermal Power</u>
>
> RATED THERMAL POWER will change from 2775 MWt to 2821 MWt.

The following provides the NRC staff's technical evaluation for each of the licensee's proposed TS changes as listed in Section 2.1 of this SE:

As discussed throughout this SE, the NRC staff finds that the licensee's proposal to increase the RTP from 2775 MWt to 2821 MWt as part of an MUR-PU is acceptable. Therefore, the proposed change to TS 1.1, "Definitions," is appropriate. The SE also confirms that the proposed change to TS 2.1.1, “Safety Limits,” regarding peak fuel centerline temperature is acceptable.

The licensee’s proposed changes to LCO 3.4.1, “RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits,” are shown in bold text below:

> LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified in the COLR. The minimum RCS total flow rate shall be ≥ **258,000** GPM, and ≥ the limit specified in the COLR.

The NRC staff’s email dated March 23, 2020 (Reference 62), requested additional information from the licensee regarding proposed changes to LCO 3.4.1. The licensee’s response, dated, April 22, 2020, stated, “The proposed deletion of the two flow measurement methods specified in the LCO while replacing them with the minimum limit for RCS flow is contained within WCAP-14483- A (Reference 63) and the associated NRC Safety Evaluation Report (SER). This change is also consistent with LCO 3.4.1 in NUREG-1431,” (Reference 64).

The licensee stated that the statistical DNBR calculations will use an updated minimum measured flow, i.e., 273,900 gallons per minute (gpm), relative to the previously analyzed value. This value is an input to NRC-approved methods in use at Farley and listed in the TS COLR References section. Because the licensee uses this value in accordance with NRC-approved methods and includes it in the cycle-specific core operating limits report, it is acceptable to relocate the value to the COLR, provided the licensee retains the minimum analyzed limit in the TS, consistent with WCAP-14483-A. Because the licensee will retain the minimum RCS flow rate of 258,000 gpm in the TS LCO, the NRC staff finds that the revised TS is consistent with NRC-approved WCAP-14483-A and is, therefore, acceptable.
Based on the above, the NRC staff finds that the proposed changes to LCO 3.4.1 are acceptable.

With this license amendment, the licensee is adopting WCAP-18124-NP-A, “Fluence Determination with RAPTOR-M3G and FERRET.” The licensee originally proposed to amend TS 5.6.6.b, “Reactor Coolant Systems (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)”, which describes the methodologies used to determine the RCS pressure and temperature limits, to state, with the proposed revision in bold:

> The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in WCAP-14040-A, Revision 4, “Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves,” May 2004 and **WCAP-18124-NP-A, Revision 0, “Fluence Determination with RAPTOR-M3G and FERRET,” July 2018.**

The NRC staff noted that the scope of WCAP-14040-A includes an entire methodology to determine reactor pressure vessel pressure-temperature limits, whereas WCAP-18124-NP-A," merely describes transport methods used to estimate neutron fluence. Therefore, the methods described in Section 2.2, “Neutron Fluence Methodology,” of WCAP-14040-A may be replaced with WCAP-18124-NP-A. On May 21, 2020, (Reference 65) the NRC staff issued a RAI requesting further clarification in TS 5.6.6.b to explain this interrelationship. In its letter dated May 29, 2020, the licensee revised the above to state, with the proposed revision in bold:

> The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," May 2004. **WCAP-18124-NP-A, Revision 0, "Fluence Determination with RAPTOR-M3G and FERRET," July 2018, may be used as an alternative to Section 2.2 of WCAP-14040-A.**

Given that the above concerns regarding the phrasing of the TS 5.6.6.b revisions have been addressed by the licensee and that the NRC staff finds the use of WCAP-18124-NP-A for determining neutron fluence acceptable in Section 3.2.1.6 of this safety evaluation, the NRC staff finds that the revision to TS 5.6.6.b is acceptable.

3.4.3 Conclusion

The NRC staff reviewed the licensee's requested RFOL and TS changes associated with the implementation of the MUR-PU and concludes that these changes are acceptable and continue to meet the regulatory requirements of 10 CFR 50.36.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Alabama State official was notified of the proposed issuance of the amendments on July 8, 2020. On July 16, 2020 the State official confirmed that the State of Alabama had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

The amendments change 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 amendments involve 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 amendments involve no significant hazards consideration, and there has been no public comment on such finding as published in the *Federal Register* on February 2, 2020 (85 FR 6231). Accordingly, the amendments meet 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 amendments.

7.0 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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1 Gayheart, Cheryl A., Southern Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Joseph M. Farley Nuclear Plants - Units 1 and 2, Submittal of License Amendment Request for Measurement Uncertainty Recapture Power Uprate," October 30, 2019 (ADAMS Package Accession No. ML19308A761).

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LIST OF ACRONYMS

AAC	alternate AC
AC	alternating current
AFW	auxiliary feedwater
ALARR	as low as reasonably achievable
AMP	aging management program
AOP	abnormal operating procedure
AOR	analysis of record
AOT	allowed outage time
AOV	air-operated valve
ARL	Alden Research Laboratories
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
BE/C	best-estimate-to-calculated
BHP	brake horsepower
BOP	balance-of-plant
BWI	Babcock & Wilcox International, Inc.
CAP	corrective action plan
CCWS	component cooling water system
CFR	*Code of Federal Regulations*
CHF	critical heat flux
CLRTP	Containment Leakage Rate Testing Program
CLTP	current licensed thermal power
CRDM	control rod drive mechanism
CSS	core support structure
CST	condensate storage tank
CVCS	chemical and volume control system
DBA	design-basis accident
DC	direct current
DG	diesel generator
DBA	design-basis accident
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
ECCS	emergency core cooling system
EDG	emergency diesel generator
EFPY	effective full power years
EMA	equivalent margins analysis
EOLE	end-of-life extension
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
EQ	environmental qualification
ESF	engineered safety feature
ETAP	Electrical Transient Analyzer Program
EVND	es-vessel neutron dosimetry
F	degrees Fahrenheit
FAC	flow accelerated corrosion
FEI	Fluid Elastic Instability
FIV	flow induced vibration
FR	*Federal Register*

FR	flatness ratio
FRD	fuel rod design analyses
FW	feedwater
GALL	Generic Aging Lessons Learned (Report)
GDC	General Design Criteria
gpm	gallons per minute
HELB	high energy line break
HEPA	high efficiency particulate air
HP	horsepower
HVAC	heating, ventilation and air conditioning
I&E	inspection and evaluation
IFM	intermediate flow mixer
ILRT	integrated leak rate test
ISA	Instrument Society of America
ISI	inservice inspection
IST	inservice test
kV	kiloVolt
LAR	license amendment request
LBB	leak-before-break
LEFM	leading edge flow meter
LLRT	local leak rate test
LOCA	loss-of-coolant accident
LOSP	loss-of-site power
LVAC	low voltage AC
LTOPS	low temperature overpressure protection system
M/C	measured-to-calculated
M&E	mass and energy
Mlbm/hr	million pounds per hour
MELB	moderate energy line breaks
M&E	mass and energy
MOV	motor-operated valve
mrem	millirem
MRP	Materials Reliability Program
MS	main steam supply system
MSLB	main steam line break
MSSS	main steam supply system
MSVR	main steam valve room
MT	main transformer
MUR	measurement uncertainty recapture
MVA	mega-voltamperes
MW	megawatt
MWe	megawatts-electric
MWt	megawatts-thermal
NIST	National Institute of Standards and Technology
NRC	U.S. Nuclear Regulatory Commission
NS	Nuclear Service Water System
NSCA	Nuclear Safety Capability Assessment
NSHC	no significant hazards consideration
NSPC	Nuclear Safety Performance Criteria
NSSS	Nuclear Steam Supply System
NSW	nuclear service water

OFA	optimized fuel assembly
OL	operating license
OM	ASME Code for Operation and Maintenance of Nuclear Power Plants
PCT	peak clad temperature
psia	pounds per square inch, absolute
psig	pounds per square inch, gauge
PLTB	pressure locking/thermal binding
P-T	pressure-temperature
PTC	Performance Test Code
PTLR	Pressure and Temperature Limits Report
PTS	pressurized thermal shock
PU	per unit
PWR	pressurized water reactor
QA	quality assurance
RAI	request for additional information
RCCA	Rod Cluster Control Assembly
RCL	reactor coolant loop
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
rem	roentgen equivalent man
RFOL	Renewed Facility Operating License
RG	Regulatory Guide
RHRS	residual heat removal system
RIP	rod internal pressure
RIS	Regulatory Issue Summary
RPV	reactor pressure vessel
RSS	root sum squares
RTD	resistance temperature detector
RTDP	revised thermal design procedure
RTP	rated thermal power
RT_{NDT}	Reference temperature for non-ductile transition
RT_{PTS}	Reference temperature for PTS
RV	reactor vessel
SATs	startup auxiliary transformer(s)
SB	Main Steam Vent to Condenser
SBAA	Southern Balancing Authority Area
SBO	station blackout
SE	safety evaluation
SER	safety evaluation report
SFP	spent fuel pool
SG	Steam Generator
SGBS	Steam Generator Blowdown System
SGTR	steam generator tube rupture
SL	Safety Limits
SLC	Select Licensee Commitments
SM	Main Steam Supply System
SNC	Southern Nuclear Operating Company
SNSWP	Standby Nuclear Service Water Pond
SRP	Standard Review Plan
SRSS	square-root-of-the-sum-of-the-squares

SSCs	structures, systems, and components
SSF	Standby Shutdown Facility
STDP	standard thermal design procedure
SV	Main Steam Vent to Atmosphere
SWS	service water system
T-cold	reactor coolant vessel core inlet temperature
TCOA	Time Critical Operator Action(s)
T/H	thermal hydraulic
T-hot	reactor coolant vessel outlet temperature
TEDE	Total Effective Dose Equivalent
TID	Total Integrated Dose
TR	technical report
TRM	Technical Requirements Manual
TS	Technical Specification
TSO	Transmission System Operator
UAT	unit auxiliary transformers
UHS	Ultimate Heat Sink
UFM	ultrasonic flow meters
UFSAR	updated final safety analysis report
UPS	uninterruptible power supply
USE	upper shelf energy
V&V	verification and validation
V	volt
WG	Radioactive Waste Management Systems: Waste Gas
WL	Liquid Waste Recycle
WM	Liquid Waste Monitor and Disposal
WS	Nuclear Solid Waste Disposal System

SUBJECT: JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENTS RE: MEASUREMENT UNCERTAINITY RECAPTURE POWER UPRATE (EPID L-2019-LLS-0002) DATED OCTOBER 9, 2020

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