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

January 26, 2021

Mr. J. Ed Burchfield, Jr.
Vice President, Oconee Nuclear Station
Duke Energy Carolinas, LLC
7800 Rochester Highway
Seneca, SC 29672-0752

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 - ISSUANCE OF AMENDMENT NOS. 420, 422, AND 421 RE: MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (EPID L-2020-LLS-0000)

Dear Mr. Burchfield:

The U.S. Nuclear Regulatory Commission (the Commission) has issued the enclosed Amendment No. 420 to Renewed Facility Operating License (RFOL) No. DPR-38, Amendment No. 422 to RFOL No. DPR-47, and Amendment No. 421 to RFOL No. DPR-55 for Oconee Nuclear Station (Oconee), Units 1, 2, and 3, respectively. The amendments are in response to the application from Duke Energy Carolinas, LLC dated February 19, 2020, as supplemented by letters dated April 6, July 23, and August 17, 2020.

The amendments consist of changes to the Oconee RFOLs and Technical Specifications to implement a measurement uncertainty recapture power uprate. Specifically, the amendments authorize an increase in the maximum licensed rated thermal power from 2568 megawatts thermal (MWt) to 2610 MWt, which is an increase of approximately 1.64 percent.

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

If you have any questions, please contact me at (301) 415-1009 or by e-mail at Shawn.Williams@nrc.gov.

Sincerely,

/RA/

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

Docket Nos. 50-269, 50-270, 50-287

Enclosures:
1. Amendment No. 420 to DPR-38
2. Amendment No. 422 to DPR-47
3. Amendment No. 421 to DPR-55
4. Safety Evaluation

cc: Listserv

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 420
Renewed License No. DPR-38

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility) Renewed Facility Operating License No. DPR-38 filed by Duke Energy Carolinas, LLC (the licensee), dated February 19, 2020, as supplemented by letters dated April 6, July 23, and August 17, 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 Renewed Facility Operating License and Technical Specifications as indicated in the attachment to this license amendment, and Paragraphs 3.A. and 3.B. of Renewed Facility Operating License No. DPR-38 are hereby amended to read as follows:

   3.A. Maximum Power Level

   The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2610 megawatts thermal.

   3.B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 420 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Caroline L. Carusone

Digitally signed by Caroline L. Carusone
Date: 2021.01.26 16:40:34 -05'00'

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: January 26, 2021

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 422
Renewed License No. DPR-47

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility) Renewed Facility Operating License No. DPR-47 filed by Duke Energy Carolinas, LLC (the licensee), dated February 19, 2020, as supplemented by letters dated April 6, July 23, and August 17, 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 Renewed Facility Operating License and Technical Specifications as indicated in the attachment to this license amendment, and Paragraphs 3.A. and 3.B. of Renewed Facility Operating License No. DPR-47 are hereby amended to read as follows:

   3.A. Maximum Power Level

   The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2610 megawatts thermal.

   3.B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 422 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Caroline L. Carusone

Digitally signed by Caroline L. Carusone
Date: 2021.01.26 16:40:55 -05'00'

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: January 26, 2021

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 421
Renewed License No. DPR-55

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

A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility) Renewed Facility Operating License No. DPR-55 filed by Duke Energy Carolinas, LLC (the licensee), dated February 19, 2020, as supplemented by letters dated April 6, July 23, and August 17, 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 Renewed Facility Operating License and Technical Specifications as indicated in the attachment to this license amendment and, Paragraphs 3.A. and 3.B. of Renewed Facility Operating License No. DPR-55 are hereby amended to read as follows:

 3.A. Maximum Power Level

 The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2610 megawatts thermal.

 3.B. Technical Specifications

 The Technical Specifications contained in Appendix A, as revised through Amendment No. 421 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 120 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Caroline L. Carusone

Digitally signed by Caroline L. Carusone
Date: 2021.01.26 16:41:23 -05'00'

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: January 26, 2021

ATTACHMENT TO

AMENDMENT NO. 420 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 422 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 421 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-55

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

Replace the following pages of the Renewed Facility Operating Licenses and the 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
DPR-38, page 3	DPR-38, page 3
DPR-47, page 3	DPR-47, page 3
DPR-55, page 3	DPR-55, page 3
TSs	TSs
1.1-5	1.1-5
3.3.1-5	3.3.1-5
3.3.1-6	3.3.1-6
3.4.3-5	3.4.3-5
3.4.3-6	3.4.3-6
3.4.3-7	3.4.3-7
3.4.3-8	3.4.3-8
3.4.3-9	3.4.3-9
3.4.3-10	3.4.3-10
3.4.3-11	3.4.3-11
3.4.3-12	3.4.3-12
3.4.3-13	3.4.3-13
3.4.4-1	3.4.4.-1

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2610 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 420 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

Any particular bulk power supply transaction may afford greater benefits to one participant than to another. The benefits realized by a small system may be proportionately greater than those realized by a larger system. The relative benefits to be derived by the parties from a proposed transaction, however, should not be controlling upon a decision with respect to the desirability of participating in the transaction. Accordingly, applicant will enter into proposed bulk power transactions of the types hereinafter described which, on balance, provide net benefits to applicant. There are net benefits in a transaction if applicant recovers the cost of the transaction (as defined in ¶1 (d) hereof) and there is no demonstrable net detriment to applicant arising from that transaction.

1. As used herein:

   (a) “Bulk Power” means electric power and any attendant energy, supplied or made available at transmission or sub-transmission voltage by one electric system to another.

   (b) “Neighboring Entity” means a private or public corporation, a governmental agency or authority, a municipality, a cooperative, or a lawful association of any of the foregoing owning or operating, or proposing to own or operate, facilities for the generation and transmission of electricity which meets each of

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2610 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 422 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

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A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2610 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 421 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

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1.1 Definitions (continued)

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation.

These tests are:

a. Described in the UFSAR;

b. Authorized under the provisions of 10 CFR 50.59; or

c. Otherwise approved by the Nuclear Regulatory Commission.

QUADRANT POWER TILT (QPT)

QPT shall be defined by the following equation and is expressed as a percentage.

$$QPT = 100\left(\frac{\text{Power in any Core Quadrant}}{\text{Average Power of all Quadrants}} - 1\right)$$

RATED THERMAL POWER (RTP)

RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2568 MWt.*

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 full length CONTROL RODS (safety and regulating) are fully inserted except for the single CONTROL ROD of highest reactivity worth, which is assumed to be fully withdrawn. However, with all CONTROL RODS verified fully inserted by two independent means, it is not necessary to account for a stuck CONTROL ROD in the SDM calculation. With any CONTROL ROD not capable of being fully inserted, the reactivity worth of these CONTROL RODS must be accounted for in the determination of SDM;

b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level; and

c. There is no change in APSR position.

*Following implementation of MUR on the respective unit, the value of RTP shall be 2610 MWt.

Table 3.3.1-1 (page 1 of 2)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION B.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Nuclear Overpower				
a. High Setpoint	1,2[a]	C	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7[d][e]	≤ 105.5% RTP with four pumps operating, and ≤ 80.5% RTP[g] when reset for three pumps operating per LCO 3.4.4, "RCS Loops - MODES 1 and 2"[f]
b. Low Setpoint	2[b],3[b] 4[b],5[b]	D	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≤ 5% RTP
2. RCS High Outlet Temperature	1,2	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≤ 618°F
3. RCS High Pressure	1,2[a]	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≤ 2355 psig
4. RCS Low Pressure	1,2[a]	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≥ 1800 psig
5. RCS Variable Low Pressure	1,2[a]	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	As specified in the COLR
6. Reactor Building High Pressure	1,2,3[c]	C	SR 3.3.1.1 SR 3.3.1.6 SR 3.3.1.7	≤ 4 psig
7. Reactor Coolant Pump to Power	1,2[a]	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	>2% RTP with ≤ 2 pumps operating
8. Nuclear Overpower Flux/Flow Imbalance	1,2[a]	C	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	As specified in the COLR

Table 3.3.1-1 (page 2 of 2)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION B.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
9. Main Turbine Trip (Hydraulic Fluid Pressure)	≥ 30% RTP	E	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≥ 800 psig
10. Loss of Main Feedwater Pumps (Hydraulic Oil Pressure)	≥ 2% RTP	F	SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≥ 75 psig
11. Shutdown Bypass RCS High Pressure	2(b),3(b) 4(b),5(b)	D	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7	≤ 1720 psig

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal.

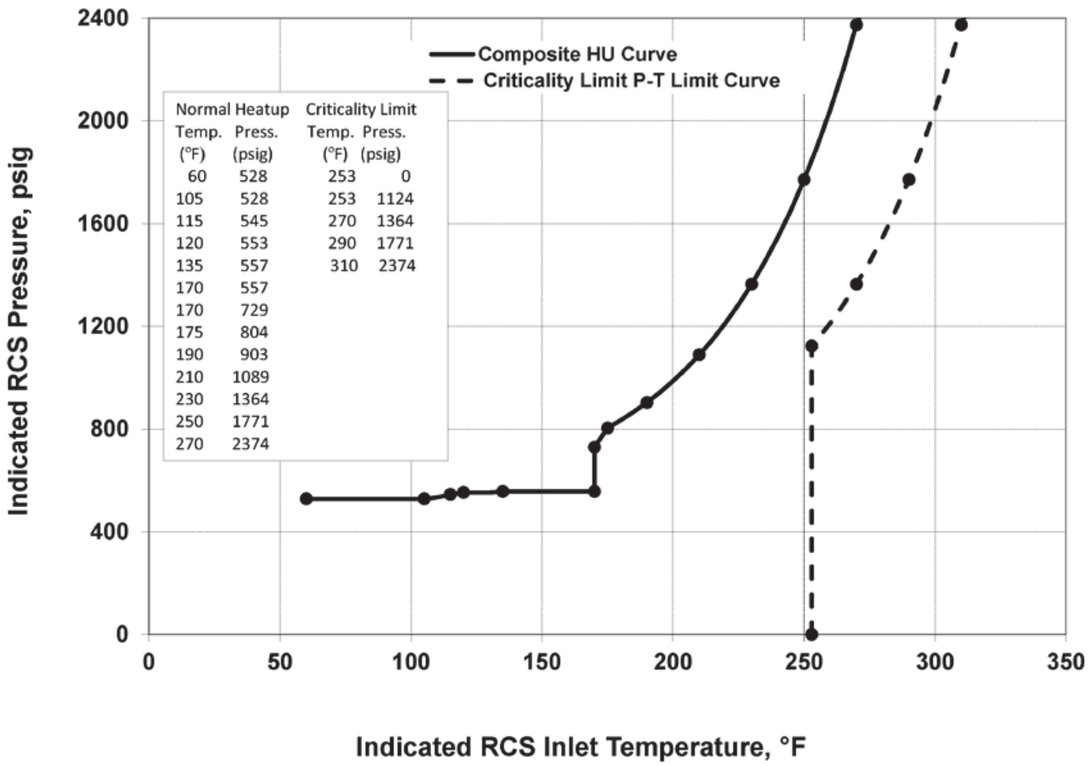
(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) If the as-found channel setpoint is conservative with respect to the Allowable Value but outside its predefined as-found acceptance criteria band, then the channel shall be evaluated to verify it is functioning as required before returning the channel to service.

(e) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint or a value that is more conservative than the Nominal Trip Setpoint; otherwise the channel shall be declared inoperable. The Nominal Trip Setpoint and the methodologies used to determine the predefined as-found acceptance criteria band and the as-left setpoint tolerance band are specified in the Selected Licensee Commitments Manual.

(f) If the high accuracy indication (including the Leading Edge Flow Meter) is unavailable, reduce the overpower trip setpoint as specified in Selected Licensee Commitment 16.7.18, “Leading Edge Flow Meter.”

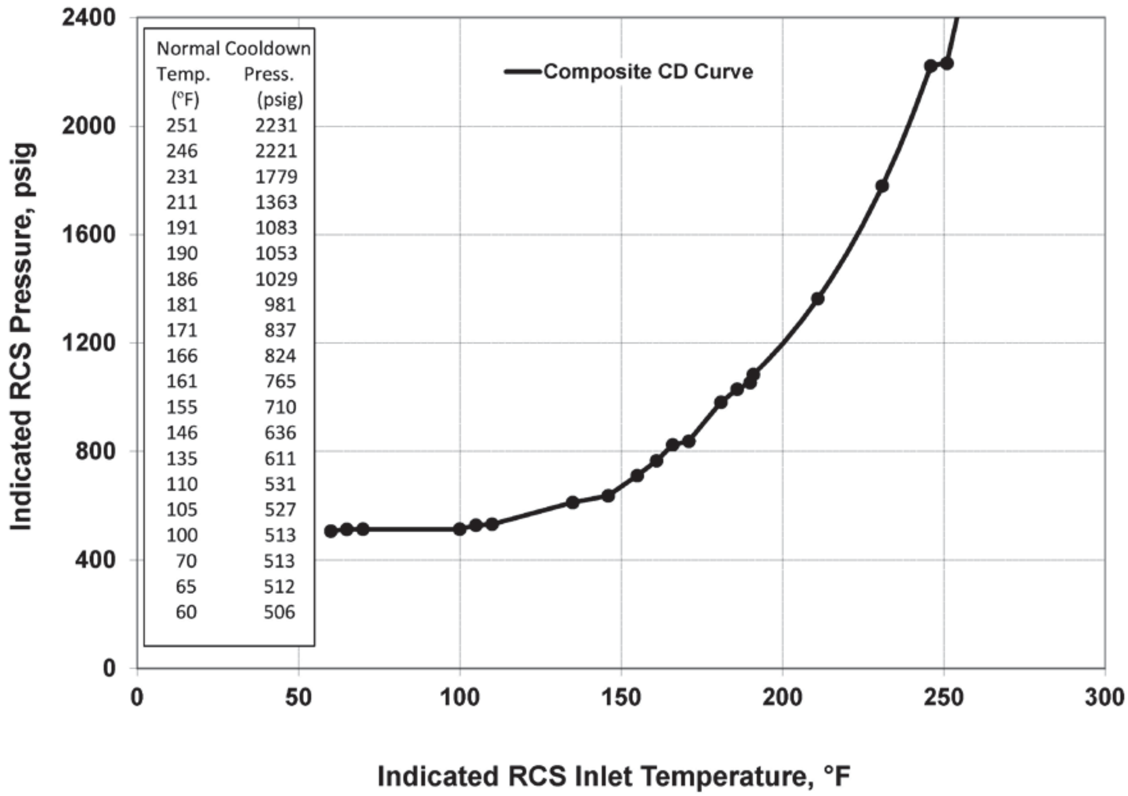
(g) Following implementation of MUR on the respective unit, the value shall be 79.3% RTP.



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. <u>Margins for instrument error are not included</u>.

<u>Note</u>: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

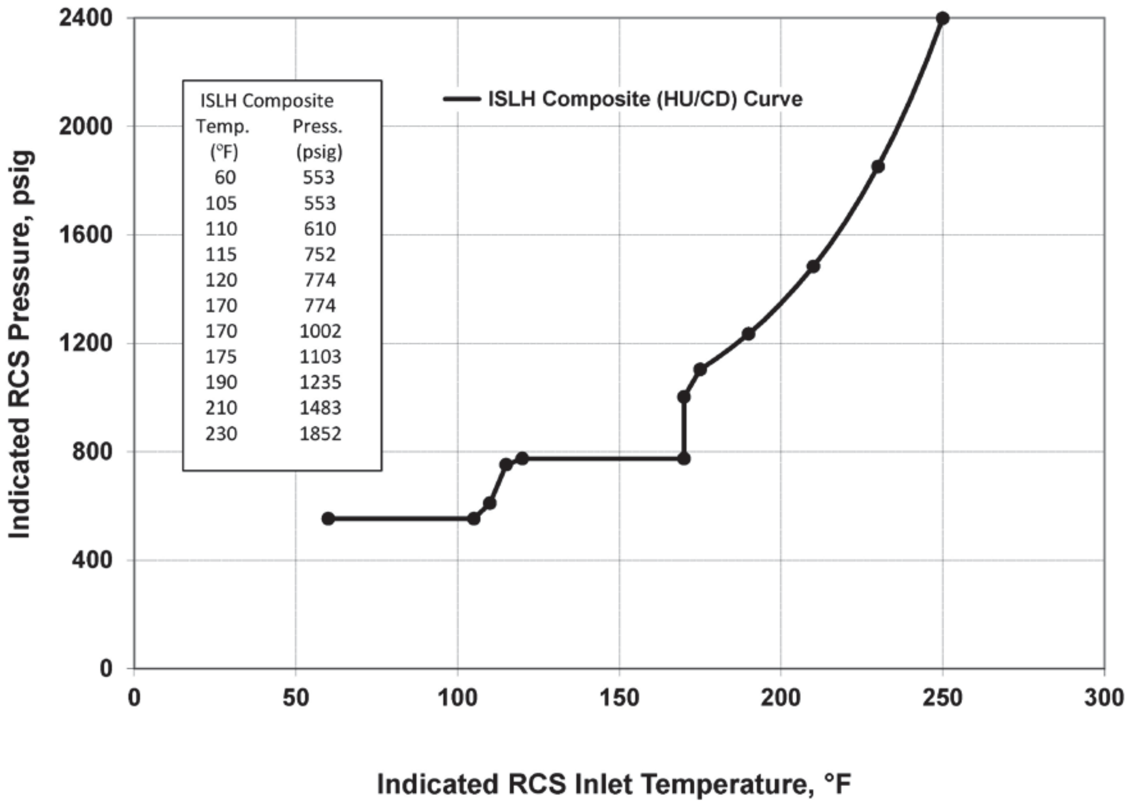
Figure 3.4.3-1 (page 1 of 1)
RCS Normal Operational Heatup Limitations
Applicable for the First 44.6 EFPY - Oconee Nuclear Station Unit 1



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

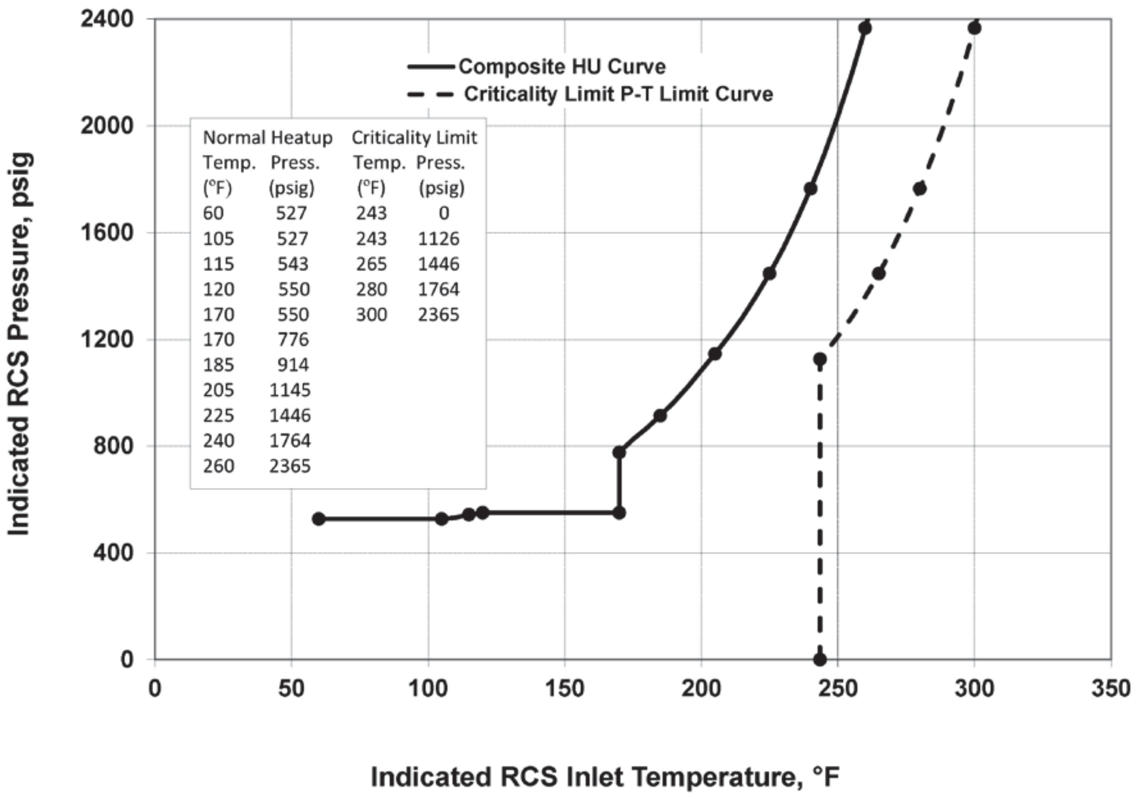
Figure 3.4.3-2 (page 1 of 1)
RCS Normal Operational Cooldown Limitations
Applicable for the First 44.6 EFPY - Oconee Nuclear Station Unit 1



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

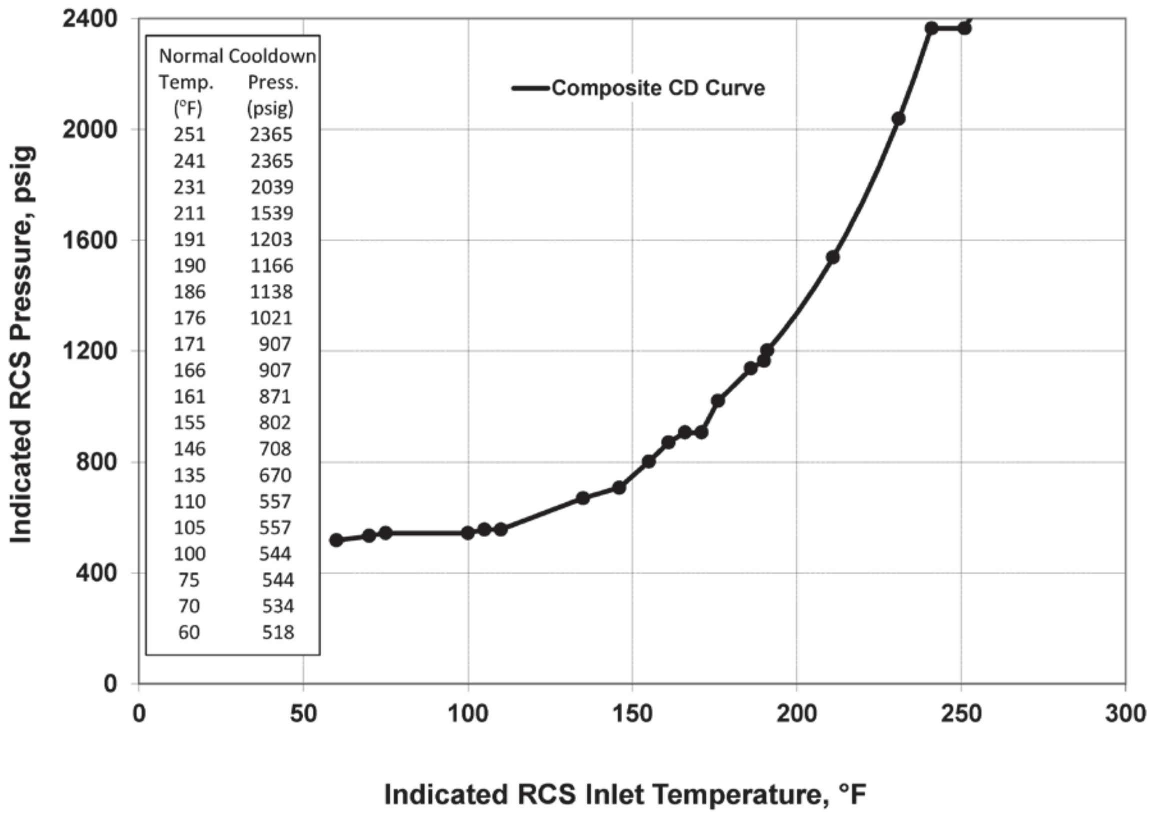
Figure 3.4.3-3 (page 1 of 1)
RCS Leak and Hydrostatic Test Heatup and Cooldown Limitations
Applicable for the First 44.6 EFPY - Oconee Nuclear Station Unit 1



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. <u>Margins for instrument error are not included</u>.

<u>Note</u>: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

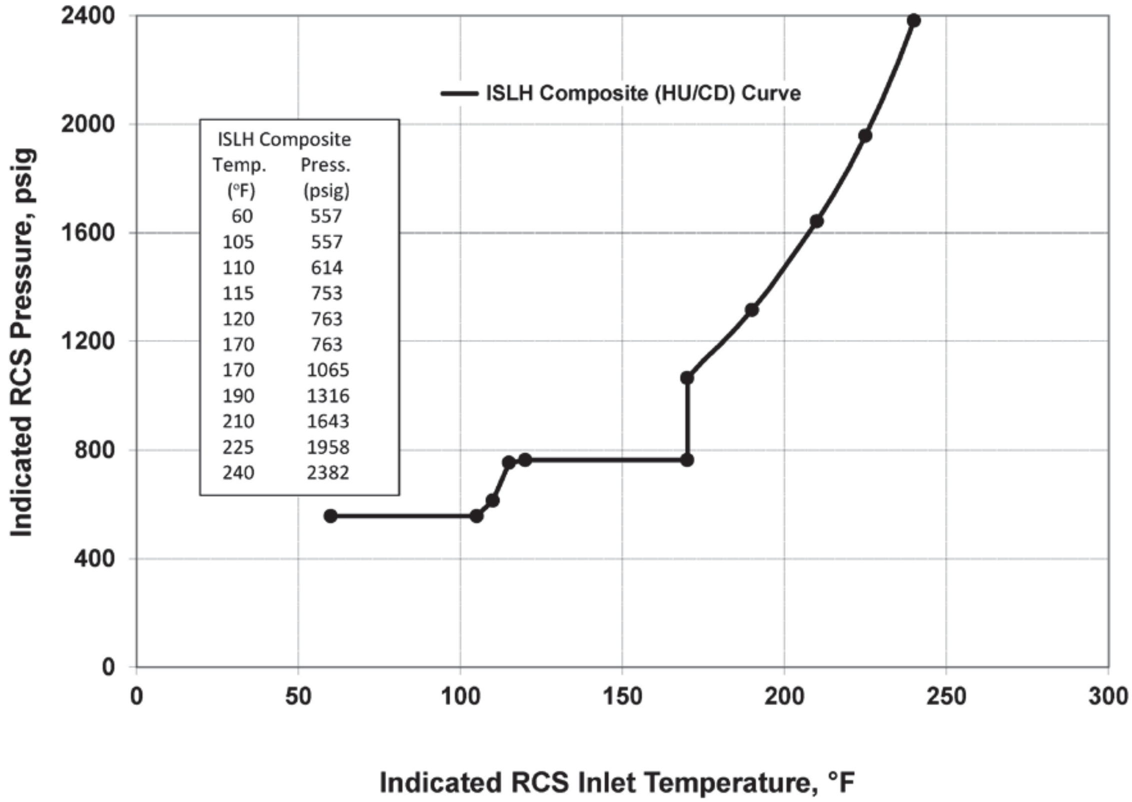
Figure 3.4.3-4 (page 1 of 1)
RCS Normal Operational Heatup Limitations
Applicable for the First 45.3 EFPY - Oconee Nuclear Station Unit 2



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

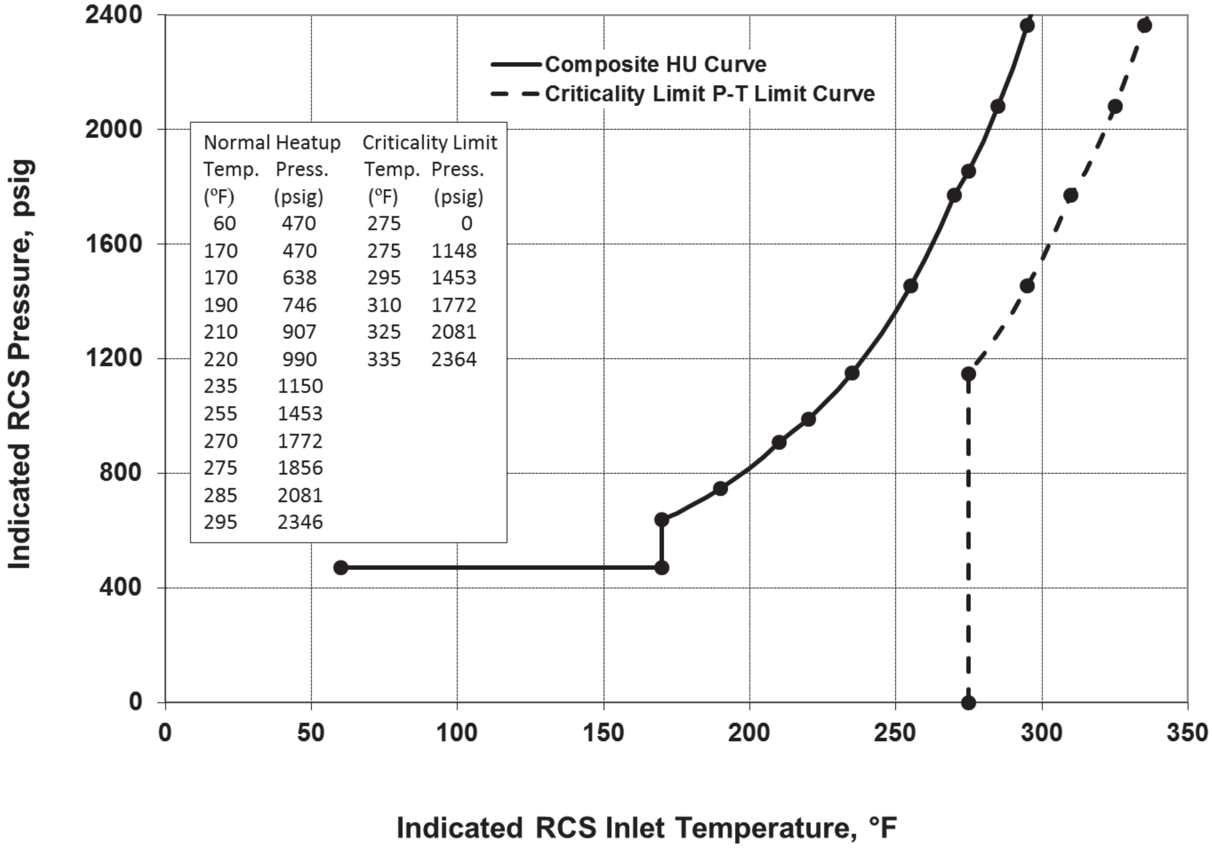
Figure 3.4.3-5 (page 1 of 1)
RCS Normal Operational Cooldown Limitations
Applicable for the First 45.3 EFPY - Oconee Nuclear Station Unit 2



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

Figure 3.4.3-6 (page 1 of 1)
RCS Leak and Hydrostatic Test Heatup and Cooldown Limitations
Applicable for the First 45.3 EFPY - Oconee Nuclear Station Unit 2

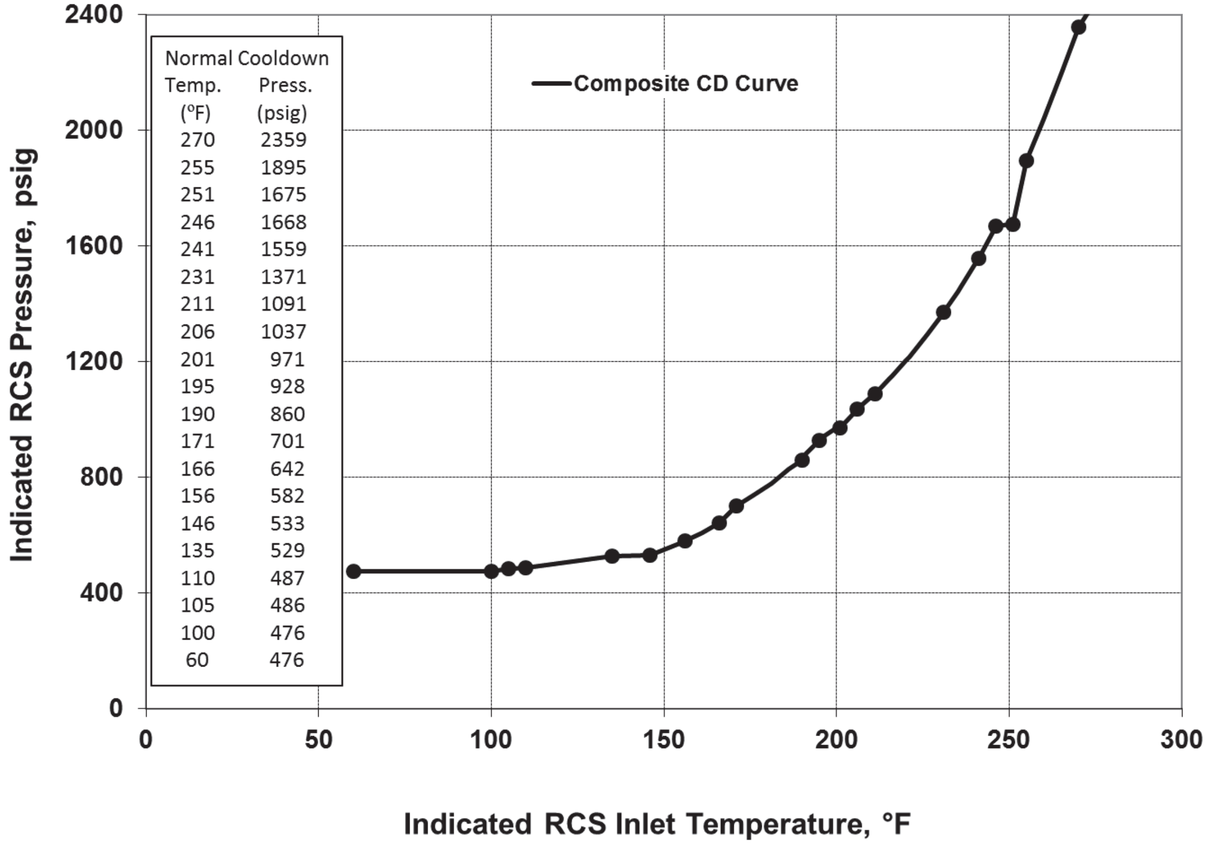


Normal Heatup Temp. (°F)	Normal Heatup Press. (psig)	Criticality Limit Temp. (°F)	Criticality Limit Press. (psig)
60	470	275	0
170	470	275	1148
170	638	295	1453
190	746	310	1772
210	907	325	2081
220	990	335	2364
235	1150		
255	1453		
270	1772		
275	1856		
285	2081		
295	2346		

The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

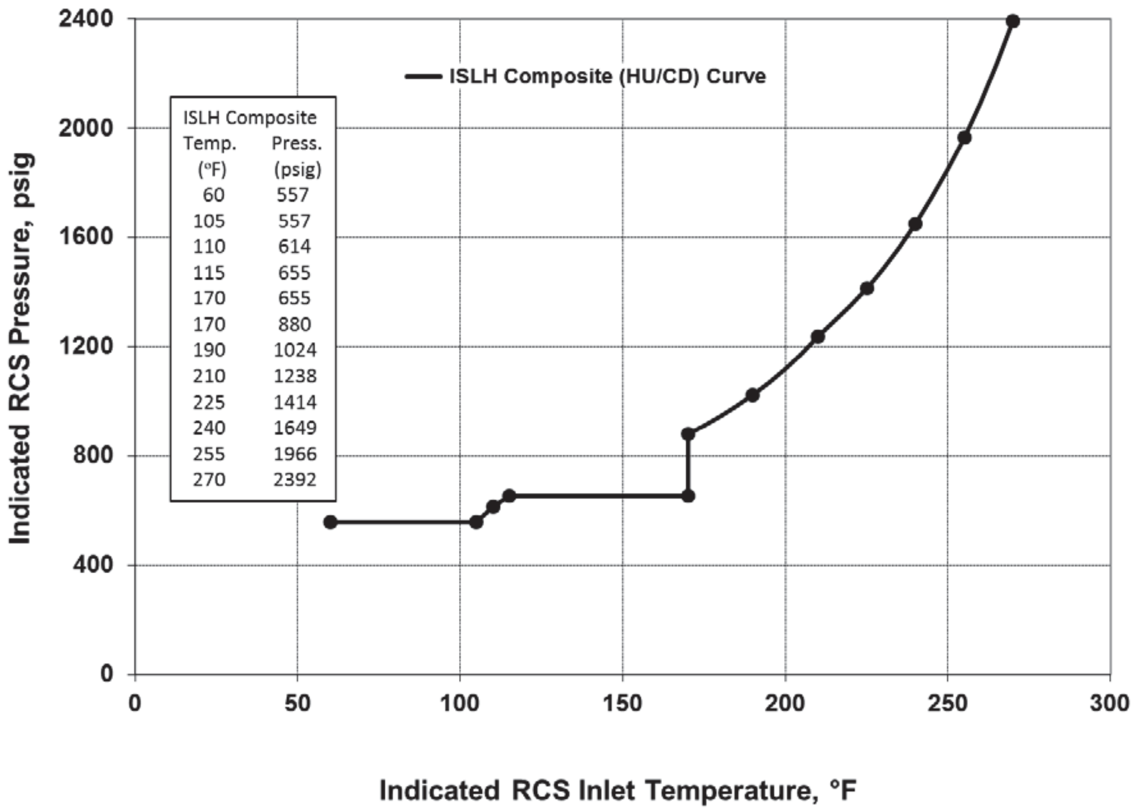
Figure 3.4.3-7 (page 1 of 1)
RCS Normal Operational Heatup Limitations
Applicable for the First 43.8 EFPY - Oconee Nuclear Station Unit 3



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

Figure 3.4.3-8 (page 1 of 1)
RCS Normal Operational Cooldown Limitations
Applicable for the First 43.8 EFPY - Oconee Nuclear Station Unit 3



The regions of acceptable operation are below and to the right of the limit curves. Margins are included for the pressure differential between point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. Margins for instrument error are not included.

Note: Heatup and Cooldown rate restrictions and Reactor Coolant Pump combination restrictions during Heatup and Cooldown are required, as identified in text.

Figure 3.4.3-9 (page 1 of 1)
RCS Leak and Hydrostatic Test Heatup and Cooldown Limitations
Applicable for the First 43.8 EFPY - Oconee Nuclear Station Unit 3

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops – MODES 1 and 2

LCO 3.4.4 Two RCS Loops shall be in operation, with:

a. Four reactor coolant pumps (RCPs) operating; or

b. Three RCPs operating and:
   1. THERMAL POWER is ≤ 75% RTP*; and
   2. LCO 3.3.1, “Reactor Protection System (RPS) Instrumentation,” Function 1.a (Nuclear Overpower – High Setpoint), Allowable Value of Table 3.3.1-1 is reset for 3 RCPs operating.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO 3.4.4.b not met.	A.1 Reset the RPS to satisfy the requirements of LCO 3.4.4.b.2.	10 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Requirements of LCO not met for reasons other than Condition A.	B.1 Be in MODE 3.	12 hours

* Following implementation of MUR on the respective unit, the value shall be 73.8% RTP.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION FOR

AMENDMENT NO. 420 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 422 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-47

AMENDMENT NO. 421 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-55

DUKE ENERGY CAROLINAS, LLC

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

1.0 INTRODUCTION

By application dated February 19, 2020 (Reference 1), as supplemented by letters dated April 6, 2020 (Reference 2), July 23, 2020 (Reference 3), and August 17, 2020 (Reference 4), Duke Energy Carolinas, LLC (Duke Energy, the licensee) submitted a license amendment request (LAR) proposing changes to the Renewed Facility Operating Licenses (RFOLs) and the Technical Specifications (TSs) for Oconee Nuclear Station (ONS, Oconee), Units 1, 2, and 3.

The licensee proposed to revise the RFOLs and TSs to implement a measurement uncertainty recapture power uprate (MUR-PU) that would increase the maximum licensed rated thermal power (RTP) from 2568 megawatts thermal (MWt) to 2610 MWt, which is an increase of 42 MWt, approximately 1.64 percent.

On June 2, 2020, the U.S. Nuclear Regulatory Commission (NRC, the Commission) staff published a proposed no significant hazards consideration (NSHC) determination in the *Federal Register* (85 FR 33745) for the proposed amendments that included the application dated February 19, 2020, and the supplemental letter dated April 6, 2020. The supplemental letters dated July 23 and August 17, 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 REGULATORY EVALUATION

2.1 Measurement Uncertainty Recapture Power Uprates

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

A change to 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 adequately accounts 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 Part 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 an LAR in accordance with 10 CFR 50.90, "Application for amendment of license construction permit, or early site permit," which must be reviewed before approval 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 5). RIS 2002-03 provides guidance to licensees on the scope and detail of the information that should be provided to the NRC staff for MUR-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 for the conduct of its review. The licensee stated that its submittal followed the guidance of RIS 2002-03.

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

The instruments originally installed to measure FW flow rate in existing nuclear power plants were usually a venturi or a flow nozzle, each of which generates a differential pressure proportional to the FW velocity in the pipe. However, errors in the determination of flow rate can be introduced due to venturi fouling and, to a lesser extent, flow nozzle fouling, the transmitter, and the analog-to-digital converter. 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 Oconee

Oconee, Units 1, 2, and 3 are currently licensed with a 2-percent margin in the ECCS evaluation model to allow for uncertainties in RTP measurement. The proposed MUR-PU LAR would reduce this uncertainty to 0.34 percent.

The licensee intends to use UFMs developed by the Cameron International Corporation (Cameron, formerly known as Caldon Ultrasonic Inc. (Caldon)),[1] specifically, the leading edge flow meter CheckPlus System (LEFM), which provides a more accurate measurement of FW flow as compared to the accuracy of the venturi flow meter-based instrumentation originally installed at Oconee, Units 1, 2, and 3. 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 current licensed thermal power (CLTP). Based on the use of the Caldon instrumentation to determine core power level with a power measurement uncertainty of approximately 0.34 percent, the licensee proposes to reduce the licensed power uncertainty within the requirements of 10 CFR Part 50, Appendix K, resulting in an approximately 1.64 percent increase in RTP.

The LEFM system will be installed in Oconee, Units 1, 2, and 3. The LEFM system will be used in lieu of the current venturi-based FW flow indication to provide FW flow input for the plant thermal heat balance calculation. Although the CheckPlus UFM system is part of the implementation of the MUR-PU, existing FW flow and temperature instrumentation will be retained and used for comparison monitoring of the LEFM system and as a backup FW flow measurement, when needed.

The LEFM system was developed over a number of years. Caldon submitted Topical Report (TR), Engineering Report (ER)-80P, Revision 0 (ER-80P), “Improving Thermal Power Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM System,” in March 1997 (Reference 6), 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 Part 50, Appendix K, in a safety evaluation (SE) dated March 8, 1999 (ER-80P SE) (Reference 7), which allowed a 1 percent power uprate using the LEFM. The NRC staff’s SE included four criteria that should be addressed by licensees incorporating ER-80P in their plant licensing basis.

Following the publication of the changes to 10 CFR Part 50, Appendix K, which allowed for an uncertainty less than 2 percent, Caldon submitted TR ER-160P, Revision 0, “Supplement to Engineering Report ER-80P: Basis for a Power Uprate with the LEFM System” (Reference 8). The NRC staff approved a plant-specific precedent using ER-160P by letter dated January 19, 2001 (Reference 9), for a power uprate of up to 1.4 percent at Watts Bar Nuclear Plant, Unit 1.

Subsequently, in an SE dated December 20, 2001 (Reference 10), the NRC staff approved a plant-specific precedent using ER-157P, Revision 5, “Supplement to Engineering Report ER-80P: Basis for a Power Uprate with the LEFM or LEFM CheckPlus System” (Reference 11), for a power uprate of up to 1.7 percent using the CheckPlus system.

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

Caldon submitted ER-157P, Revision 8, “Supplement to Caldon Topical Report ER-80P: Basis for Power Uprates with an LEFM Check or an LEFM CheckPlus System” (ER-157P, Rev. 8), in May 2008 (Reference 12) and an associated errata on October 15, 2010 (Reference 13). The ER-157P, Rev. 8 corrected minor errors in Revision 5, provided clarifying text, and incorporated revised analyses of coherent noise, non-fluid delays, and transducer replacement. It also added two new appendices, Appendix C and Appendix D, which describe the assumptions and data that support the coherent noise and transducer replacement calculations, respectively.

In its SE for ER-157P, Rev. 8 (Reference 14), dated August 16, 2010 (ER-157P SE), the NRC staff approved ER-157P, Rev. 8 and associated errata, subject to five criteria that should be addressed by licensees incorporating ER-157P, Rev. 8 in their plant licensing basis.

2.3 Regulatory Requirements and Guidance

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

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

RIS 2002-03, “Guidance on the Content of Measurement Uncertainty Recapture Power Uprate Applications,” provides a method to satisfy the requirements of 10 CFR Part 50, Appendix K.

The Atomic Energy Commission (AEC) issued the construction permits for Oconee on November 6, 1967. The plants’ design approval for the construction phase was based on the proposed general design criteria (GDC) published by the AEC in the *Federal Register* (32 FR 10213) on July 11, 1967. The GDC that constitute the licensing basis for Oconee are those described in the Updated Final Safety Analysis Report, Revision 28 (UFSAR) (Reference 15), Chapter 3.1, and in applicable UFSAR sections. As discussed in the UFSAR, the licensee made changes to the facilities and committed to some of the GDC from 10 CFR Part 50, Appendix A.

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, ER-157P, describe the Cameron LEFM CheckPlus System for the measurement of FW 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 Part 50, Appendix K.

Regulatory Guide (RG) 1.105, Revision 3, “Setpoints for Safety-Related Instrumentation,” (Reference 16) describes a method acceptable to the NRC for complying with the NRC regulations for ensuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits.

3.1.1.2 Technical Evaluation

The NRC staff’s review addressed the proposed plant-specific implementation of the FW flow measurement technique and the power increase gained as a result of implementing this technique in accordance with 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 FW 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 submittal included proprietary and non-proprietary versions of the following Cameron Engineering Reports:

- ER-813P, Revision 6, “Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 1 Using the LEFM CheckPlus System,” submitted on February 19, 2020.
    - ER-813P, Revision 7, was submitted on July 23, 2020, to correct an error in proprietary markings.
- ER-824P, Revision 6, “Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 2 Using the LEFM CheckPlus System,” submitted on February 19, 2020.
    - ER-824P, Revision 7, was submitted on July 23, 2020, to correct errors in proprietary markings.
- ER-825P, Revision 6, “Bounding Uncertainty Analysis for Thermal Power Determination at Oconee Unit 3 Using the LEFM CheckPlus System,” submitted on February 19, 2020.
    - ER-825P, Revision 7, was submitted on July 23, 2020, to correct errors in proprietary markings.
- ER-855P, Revision 0, “Meter Factor Calculation and Accuracy Assessments for the LEFM CheckPlus Meters at Oconee Units 1, 2, and 3,” submitted on February 19, 2020.

3.1.1.2.1 Feedwater Flow Measurement Technique and Power Measurement Uncertainty

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

The licensee provided an analysis of the uncertainty contribution of the LEFM CheckPlus system when operating in the check plus mode, as well as when operating in the Check mode, to the overall calculated thermal power uncertainty (ER-813P, ER-824P, ER-825P, ER-855P). These uncertainties were calculated using the methodology described in ER-157P, Rev. 8. These uncertainties were ± 0.30 percent in the heat balance uncertainty calculation. These uncertainties combine to give an overall secondary heat balance power measurement uncertainty of 0.34 percent RTP.

Since June 1, 2000, the rulemaking regarding Appendix K to 10 CFR Part 50 has allowed a licensed power level that maintains a demonstrated level of margin to 102 percent of CLTP. The licensee used 102 percent of the Oconee CLTP (2568 MWt), or 2619 MWt as a maximum value to determine the MUR-PU value. The total thermal power heat balance calculation uncertainty and bias is obtained by combining the following input uncertainties as random terms and biased: reactor coolant pump (RCP), makeup/letdown power, steam enthalpy and feedwater pressure. The NRC staff reviewed the ONS Units 1, 2, and 3 specific heat balance related uncertainty calculations, found them to be acceptable and, accordingly, found the thermal power uncertainties and proposed power level to have been conservatively determined.

As part of the MUR-PU application, the licensee provided a discussion of the ONS procurement specifications and the Duke Energy procedures, oversight activities, and quality assurance requirements, which combine to ensure that the reactor thermal power licensing requirements are not exceeded. The NRC staff reviewed this information and found that the licensee has implemented robust controls to avoid exceeding the reactor thermal power licensing requirements.

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 response.

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

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

The NRC staff reviewed the application and found that the licensee identified the applicable approved topical reports as ER-80P, Rev. 0, ER-157P, Rev. 8, and ER-157P, Rev. 8 Errata.

The NRC staff also found that the licensee identified the following NRC SEs that approved the topical reports:

> NRC letter from John Hannon, to C. Lance Terry, Texas Utilities Electric, “Comanche Peak Steam Electric Station, Units 1 and 2 – Review of Caldon Engineering Topical Report ER 80P, ‘Improving Thermal Power Accuracy and Plant Safety While Increasing Power Level Using the LEFM System’ (TACS Nos. MA2298 and MA2299),” March 8, 1999 (Reference 17).

> NRC letter from Thomas Blount, to Ernest Hauser, Cameron, "Final Safety Evaluation for Cameron Measurement Systems Engineering Report ER-157P, Revision 8, 'Caldon Ultrasonics Engineering Report ER-157P, 'Supplement to Topical Report ER-80P: Basis for a Power Uprate with the LEFM Check or CheckPlus System',' (TAC No. ME1321)," August 16, 2010 (Reference 18).

The licensee indicated that the LEFM CheckPlus system will be installed permanently in Oconee and operated in accordance with the manufacturer's requirements as described in Topical Reports ER-80P and ER-157P.

The licensee indicated that the Oconee LEFM CheckPlus system consists of an electronic cabinet located in the turbine building and two measurement section/spool pieces consisting of four electronic transmitters and four pressure transmitters. The LEFMs will be installed in horizontal runs of main FW piping upstream of the existing venturis. The Oconee LEFMs installation configuration meets or exceeds the required four Length/Diameter (L/D) upstream distance from the existing venturis.

The licensee indicated that the Oconee LEFM CheckPlus system testing was performed at Alden Research Laboratory (ARL) and that the bounding uncertainty analysis results were documented in Caldon Engineering Report ER-813P, ER-824P, and ER-825P for Units 1, 2, and 3, respectively, and that the meter factor calculation was documented in ER-855P for all three units, all of which were included as part of the application.

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

Based on the above, the NRC staff determined that the licensee has adequately addressed the plant-specific implementation of the Cameron LEFM CheckPlus system that is supported by NRC-approved topical reports. Therefore, the NRC staff concludes that the licensee's description of the FW flow measurement technique and implementation of the power uprate using this technique follows the guidance in Items A through C of Section I of Attachment 1 to RIS 2002-03 and meets the regulatory requirements of 10 CFR Part 50, Appendix K.

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

Item D in Section I of Attachment 1 to RIS 2002-03 guides licensees in addressing the criteria established by the NRC staff in its approval of the FW flow measurement uncertainty technique used by the licensee in the application.

When the NRC staff approved ER-80P and ER-157P, Rev. 8, it established nine criteria (four criteria for ER-80P and five criteria for ER-157P) that licensees were to address in order to implement these topical reports at their facilities. The licensee addressed these criteria in Enclosure 2 to its application. The NRC staff evaluated the licensee's approach to addressing each of these criteria as follows:

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

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

In its application, the licensee stated that implementation of the MUR-PU would include developing the necessary procedures and documents required for operation and maintenance at the uprated power level with the new LEFM CheckPlus system. Implementation would also include training of operating and maintenance personnel. A preventive maintenance program would be developed prior to implementing the LEFM CheckPlus system using Cameron's maintenance and troubleshooting manual and Duke Energy's established procedure program.

Typical preventive maintenance activities would include the following checks:

- General inspection of the terminal and cleanliness
- Power Supply inspection of magnitude and noise
- Central Processing Unit inspection
- Acoustic Processor Unit Checks of the 5 MHz clock and LED status
- Analog Input checks of the analog-to-digital converter
- Alarm Relay checks
- Watchdog Timer checks that ensure the software is running
- Transducer Cable checks
- Calibration checks of each of the FW pressure transmitters

The preventive maintenance program and continuous monitoring of the LEFM would ensure that the LEFM operation remains bounded by analysis and assumptions set forth by the vendor.

Evaluation of Criterion 1 and Criterion 2 from the SE for ER-157P, Rev. 8, discussed below in this SE section, describes plant operation for inoperable LEFM instrumentation.

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

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

> For plants that currently have LEFMs installed, the licensee should provide an evaluation of the operational and maintenance history of the 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.

In its application, the licensee stated that Oconee currently uses flow venturis to measure FW flow to support secondary calorimetric power measurements.

Based on the NRC staff's review of the licensee's application, Criterion 2 from the SE for ER-80P does not apply to Oconee and, therefore, the NRC staff concludes that the licensee adequately addressed Criterion 2 from the SE for ER-80P. Additionally, the licensee stated that after the LEFM CheckPlus system is installed and operational, 30 days of data would 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.

Criterion 3 from the SE for ER-80P

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

In its application, the licensee stated that the LEFM uncertainty calculation is based on the American Society of Mechanical Engineers (ASME) Performance Test Code (PTC) 19.1 and International Society of Automation (ISA, formerly known as Instrumentation, Systems, and Automation Society) Recommended Practice (RP) ISA RP 67.04 and ARL calibration tests. The methodology is consistent with the plant setpoint methodology and the current heat balance uncertainty calculation that uses the FW flow venturis and Resistance Temperature Detectors.

The FW flow and temperature uncertainties are combined with other plant measurement uncertainties to calculate the overall heat balance uncertainty, consistent with the accepted plant setpoint methodology. As stated in the Unit-Specific Bounding Uncertainty Analysis reports, random and systemic uncertainty terms are combined by the means of the root sum squared approach, provided they are independent, zero-centered, and normally distributed, while uncertainty terms caused by a common (systemic) condition are combined algebraically. This approach is consistent with RG 1.105, Revision 3.

Based on the above, the NRC staff concludes that the licensee adequately addressed Criterion 3 from the SE for ER-80P.

Criterion 4 from the SE for ER-80P

> 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.

In its application, the licensee stated that this criterion does not apply to Oconee because the flow elements were tested and calibrated in a full-scale model of the Oconee Units 1, 2, and 3, hydraulic geometry at the ARL. The bounding calibration factor for the Oconee Units 1, 2, and 3, spool pieces was established by these tests and is included as attachments to the LAR in ER-813P, ER-824P, and ER-825P for each unit, respectively. ER-855P was also included as an attachment to the LAR and summarizes the testing and evaluates the test data.

Based on the above, the NRC staff finds that Criterion 4 from the SE for ER-80P does not apply to Oconee and, therefore, the NRC staff concludes that the licensee adequately addressed Criterion 4.

Criterion 1 from the SE ER-157P

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

In its application, the licensee stated that:

> A Selected Licensee Commitment (SLC) will be added to address functional requirements for the LEFMs and appropriate Required Actions and Completion Times when an LEFM is not functional. This is identified as a Regulatory Commitment in Attachment 1. If a non-functional LEFM is not restored to functional status within 72 hours, then within 6 hours, the unit will be reduced to no more than 2568 MWt (the previously licensed rated thermal power).
>
> The existing feedwater flow venturi-based signals will be corrected to the last valid data from the LEFM system during this period. Any slight drift of the feedwater flow venturi measurements due to fouling would result in a higher than actual indication of feedwater flow and an overestimation of the calculated calorimetric power level. This is conservative since the reactor will be operating below the calculated power level. A sudden de-fouling event during the 72-hour inoperability period is unlikely and any significant sudden de-fouling would be detected by other plant parameters. It is expected that most issues rendering an LEFM system non-functional could be resolved within a 72-hour AOT [Allowed Outage Time].

The NRC staff reviewed the information provided above for the proposed AOT. Based on the above, 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 likely to be exceeded. Therefore, the NRC staff concludes that the licensee adequately addressed Criterion 1 from the SE for ER-157P.

Criterion 2 from the SE for ER-157P

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

In its application, the licensee stated that:

> ONS will not consider a CheckPlus system with a single failure as a separate category; such a failure will be considered as a non-functional LEFM and the same actions identified in response to Criterion 1 from ER-157P, Rev. 8 above will be implemented.

Based on the above, the NRC staff finds that Criterion 2 from the SE for ER-157P, Rev. 8 does not apply to Oconee and, therefore, the NRC staff concludes that the licensee adequately addressed Criterion 2.

Criterion 3 from the SE for ER-157P

> [A 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 [Research] Laboratory tests.

In its application, the licensee stated that:

> As stated in response to Criterion 2 from ER-157P, Rev. 8 above, ONS will not consider a CheckPlus system with disabled components as a separate category; such a condition will be considered as an inoperable LEFM and the same actions identified in response to Criterion 1 above will be implemented.

Based on its review of the licensee's application, the NRC staff finds that Criterion 3 from the SE for ER-157P, Rev. 8 does not apply to Oconee and, therefore, the NRC staff concludes that the licensee adequately addressed Criterion 3.

Criterion 4 from the SE for ER-157P

> [A licensee] that requests a MUR with the upstream flow straightener configuration discussed in Section 3.2.2 should provide justification for claimed CheckPlus uncertainty that extends the justification provided in Reference [62]. Since the Reference [62] 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.

In its application, the licensee stated that the Oconee units have no flow straighteners upstream (or downstream) of the LEFM installation.

Based on the above, the NRC staff finds that Criterion 4 from the SE for ER-157P, Rev. 8 does not apply to Oconee and, therefore, the NRC staff concludes that the licensee adequately addressed Criterion 4.

Criterion 5 from the SE for ER-157P

> [A 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 [61].

In its application, the licensee stated that:

> The ONS Nuclear Steam Supply Systems (NSSSs) use Once-Through Steam Generators (OTSGs) that produce superheated steam as shown in Table IV-1 below. Thus, uncertainty associated with the steam moisture content at ONS is not a factor in the heat balance uncertainty calculation. This criterion is not applicable to ONS.

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 for Oconee. This is considered an insignificant factor because of the approach to calculate the total power uncertainty and the contribution of this steam moisture is negligible to the total power uncertainty. Additionally, if more moisture is present, the calculated power will be greater than the actual power and would, therefore, be a conservative calculation.

Based on the above, the NRC staff finds that Criterion 5 from the SE for ER-157P, Rev. 8 does not apply to Oconee and, therefore, the NRC staff concludes that the licensee adequately addressed Criterion 5.

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

In this section, the NRC staff evaluated the licensee's responses to Item D of Section I, Attachment 1 to RIS 2002-03. The licensee stated that Criteria 2 and 4 from the SE for ER-80P, and Criteria 2, 3, 4, and 5 from the SE for ER-157P, Rev. 8 were not applicable to Oconee. The NRC staff reviewed these assessments by the licensee and determined these criteria did not apply to Oconee. The NRC staff also reviewed the licensee's evaluation of Criteria 1 and 3 from the SE for ER-80P and the licensee's evaluation of Criteria 1 from the SE for ER-157P and determined that they were acceptable. Therefore, the NRC staff concludes that the licensee adequately addressed the guidance in Item D of Section I, Attachment 1 to RIS 2002-03 to implement the FW flow measurement uncertainty technique from the NRC staff approved ER-80P and ER-157P, Rev.8 and errata referenced topical reports, and would meet the regulatory requirements of 10 CFR Part 50, Appendix K.

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

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

To address Item E, the licensee provided ER-813P, ER-824P, and ER-825P. In addition, the licensee provided Table I.1.E-1, "Total Thermal Power Uncertainty Determination," in Enclosure 2 to its application, which provides key parameters and their uncertainty, including both the uncertainties associated with parameters provided by the LEFM CheckPlus system as well as uncertainties associated with the other plant parameters that are used by the plant computer to calculate the calorimetric. The licensee stated that acceptance testing following the installation of the CheckPlus System in the Oconee units would confirm that the as-built parameters are within the bounds of the error analysis.

Consistent with the accepted plant setpoint methodology, random and systemic uncertainty terms are combined by the means of the root sum squared approach, provided that they are independent, zero-centered and normally distributed, while uncertainty terms caused by a common (systemic) condition are combined algebraically. This approach is consistent with RG 1.105, Revision 3.

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

The NRC staff reviewed the application and determined that the licensee has provided the plant-specific total power measurement uncertainty calculation for Oconee, and has identified explicitly the parameters and their individual contributions to the overall thermal power uncertainty. The overall thermal power uncertainty was then calculated using an approach consistent with RG 1.105, Revision 3. Therefore, the NRC staff concludes that the licensee adequately addressed the guidance in Item E of Section I, Attachment 1 to RIS 2002-03, and would meet the regulatory requirements of 10 CFR Part 50, Appendix K.

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

Item F in Section I of Attachment 1 to RIS 2002-03 guides licensees in providing information to specifically address the following aspects of the calibration and maintenance procedures related to all instruments that affect the power calorimetric. Each aspect is followed by the information provided by the licensee to address Item F, as stated in Enclosure 2 to its application.

(1) Maintaining Calibration

Calibration of the LEFM will be ensured by preventative maintenance activities previously described in Section I.1.D.i [in the application], Response to Criterion 1 [from the SE for] ER-80P.

(2) Controlling Software and Hardware Configuration

The Cameron LEFM CheckPlus Systems were procured to the requirements of ANSI Std 7-4.3.2-2003 … and ASME NQA-1, 2008…. Hardware configuration will be controlled in accordance with Duke Energy procedures.

LEFM software will be classified in accordance with Duke Energy procedures. Software will be classified, developed, tested, and controlled in accordance with Duke Energy procedures. Implementation of the software will be performed under the design control process.

Instruments that affect the power calorimetric, including the Cameron LEFM CheckPlus System inputs, are monitored by ONS personnel. Equipment problems for plant systems, including the Cameron LEFM CheckPlus System equipment, fall under site work control processes. Conditions that are adverse to quality are documented under the corrective action program. Corrective action directives, which ensure compliance with the requirements of 10 CFR [Part] 50, Appendix B, include instructions for notification of deficiencies and error reporting.

(3) Performing Corrective Actions

Corrective actions will be monitored and performed in accordance with Duke Energy procedures and the Work Process Manual.

(4) Reporting Deficiencies to the Manufacturer

Reporting deficiencies to the manufacturer will be performed in accordance with Duke Energy procedures.

(5) Receiving and Addressing Manufacturer Deficiency Reports

> Manufacturer deficiency reports will be received and addressed in accordance with Duke Energy procedures.

NRC Staff Conclusion Regarding Item F of Section I, Attachment 1 to RIS 2002-03

Based on the above, the NRC staff determined that the licensee has 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 adequately addressed the guidance in Item F of Section I, Attachment 1 to RIS 2002-03, and would meet the regulatory requirements of 10 CFR Part 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 in Section I of Attachment 1 to RIS 2002-03 guide licensees in providing a proposed AOT for the instrument, along with the technical basis for the time selected, and proposed actions to reduce power if the AOT is exceeded. Items G and H refer to Criterion 1 from the SE for ER-157P.

To address Items G and H, in the application the licensee refers to its response to Criterion 1 from the SE for ER-157P.

NRC Staff Conclusion Regarding Items G and H of Section I, Attachment 1 to RIS 2002-03

The NRC staff reviewed the information provided in the application in response to Criterion 1 from the SE for ER-157P discussing the proposed AOT and the actions if the AOT is exceeded. Based on the review, the NRC staff found that the licensee provided sufficient justification for the proposed 72-hour AOT and the actions to reduce power level to 2568 MWt (i.e. the pre-MUR reactor power limitations) if the AOT is exceeded. Therefore, the NRC staff concludes that the licensee adequately addressed the guidance in Items G and H of Section I, Attachment 1 to RIS 2002-03, and thus meets the regulatory requirements of 10 CFR Part 50, Appendix K.

3.1.1.3 NRC Staff Conclusion

The NRC staff's evaluation of the identified instrumentation for new power levels for Oconee Units 1, 2, and 3, is based on the analytical limits documented by the licensee in its submittal. Based on its review of the application and Caldon ERs (including uncertainty calculations and referenced topical reports), the NRC staff determined that the licensee provided sufficient justification for the proposed TS changes. The licensee's proposed amendment is consistent with ER-80P and its supplement, ER-157P. The licensee followed the guidance in Items A through H in Section 1 of Attachment 1 to RIS 2002-03 and would meet the applicable regulatory requirements and guidance. Therefore, the NRC staff concludes that the instrumentation and controls aspect of the proposed MUR-PU is acceptable.

3.1.2 Containment Analysis

3.1.2.1 Regulatory Evaluation

The NRC staff identified the following Oconee UFSAR Plant Design Criteria as being applicable to the proposed amendment.

UFSAR Section 3.1.10, "Criterion 10 - Containment (Category A)," states, in part:

> Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity, and, together with other engineering safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

UFSAR Section 3.1.49, "Criterion 49 - Containment Design Basis (Category A)," states, in part:

> The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of Emergency Core Cooling Systems.

UFSAR Section 3.1.52, "Criterion 52 - Containment Heat Removal Systems (Category A)," states, in part:

> Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

UFSAR Section 3.1.70, "Criterion 70 - Control of Releases of Radioactivity to the Environment (Category B)," states, in part:

> The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified: a) on the basis of 10 CFR [Part] 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and b) on the basis of 10 CFR [Part] 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

3.1.2.2 Technical Evaluation

The NRC staff reviewed the following areas of containment design and analysis for the LAR: short-term 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-term pressurization of the containment following a LOCA is discussed in UFSAR Section 6.2.1.1.3.1. The long-term temperature response within containment following a LOCA is discussed in UFSAR Section 6.2.1.1.3.2. A discussion of containment pressure and temperature following a steam line break is contained in UFSAR Section 6.2.1.1.3.3.

The licensee stated that the analyses discussed in the above listed UFSAR sections are performed to ensure that the containment pressure limit is not exceeded and that the temperature response assumed in the environmental qualification (EQ) analyses remain bounding. Small break LOCAs are analyzed to verify that a large break LOCA is more limiting.

The long-term mass and energy release data input to the containment analysis of record (AOR) for Oconee assumes a core thermal power of 2619 MWt, which corresponds to the licensed core power of 2568 MWt with an additional 2 percent uncertainty, that bounds the proposed MUR-PU. The licensee stated that the peak containment pressure is below the design limit and the temperature profile assumed in the EQ analyses is not challenged.

Based on the above, the NRC staff finds that the effects of the proposed MUR-PU on the short- and long-term LOCA mass and energy (M&E) release are bounded by the AOR and are, 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 of performing their intended safety function following the proposed MUR-PU.

In Section II.1.D.iii. Items 26 and 27 of Enclosure 2 to the LAR, the licensee described a loss of main FW and a FW line break, respectively. The loss of main FW event is analyzed using the NRC-approved RETRAN-3D transient analysis computer code to demonstrate the adequacy of the emergency feedwater (EFW) system. The licensee stated that the event is initiated from 102 percent of 2568 MWt and is analyzed for peak reactor coolant system (RCS) pressure. The proposed MUR-PU would not affect the analysis because the initial power level, stored energy of the structures, systems, and components (SSCs), and decay heat values used in it are the result of operation at 102 percent of 2568 MWt. The licensee stated that the analysis is reflected in the Oconee UFSAR and would remain acceptable for the proposed MUR-PU.

The licensee stated that FW line breaks are evaluated as part of the high energy line break (HELB) program and that the current licensing basis for FW line break analyses are bounded by the transient analyses performed for FW line breaks in support of the future licensing basis for HELB. The HELB scenario is discussed in Section 3.1.4.2.6 of this evaluation.

The licensee's analysis bounds the proposed MUR-PU power level of 2610 MWt and, therefore, the NRC staff finds that the analysis would be unaffected by the MUR-PU.

3.1.2.2.3 Fuel Handling Accidents in the Containment

The fuel handling accidents (FHAs), as described in UFSAR Section 15.11, include four base accidents, with one base having three separate conditions. One of the base accidents is the FHA inside containment, which is described in UFSAR Section 15.11.2.2.

The licensee described the FHAs in Section II.1.D.iii. Item 11 of Enclosure 2 to the LAR. The FHA radiological dose analysis uses a bounding fission product inventory based on operation at 102 percent RTP (i.e., 2619 MWt), which bounds the proposed MUR-PU. Therefore, the licensee stated that the FHA analysis would remain acceptable for the MUR-PU. The FHA analysis was performed using the NRC reviewed and approved Alternative Source Term methodology in RG 1.183 (Reference 19).

Since the source term was calculated at an initial power level of 102 percent RTP, which bounds the proposed MUR-PU power level of 2610 MWt, the NRC staff finds that the analysis would be unaffected by the MUR-PU.

3.1.2.2.4 Environmental Qualification

With respect to EQ, the licensee stated, in part, in Section II.1.D.iii. Item 21 of Enclosure 2 to the LAR, that the proposed MUR-PU would affect the M&E release data input to the various structure analyses and consequently affect the EQ analyses. However, the containment response analyses following a LOCA or MSLB all obtain the M&E data from analyses performed at 2619 MWt. The containment response analysis following a National Fire Protection Association (NFPA) Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition" (NFPA-805) based fire also obtains the M&E data from analyses performed at 2619 MWt.

The licensee also stated that the penetration room response analyses following a main feedwater line break (MFWLB) (large break and critical crack) or MSLB in the penetration room obtain the M&E data from analyses performed at 2619 MWt. The large break MFWLB penetration room pressure/temperature results bound the critical crack penetration room results. The large break MFWLB and MSLB M&E analyses are generated using NRC reviewed and approved methods at 2619 MWt.

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

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, containment isolation system, reactor building spray, and reactor building cooling system.

The licensee evaluated the impact of the proposed MUR-PU on these systems as discussed in Section IV.1.B of Enclosure 2 to the LAR. The LAR indicates 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 proposed MUR-PU. Therefore, the NRC staff concludes that containment systems 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.E of Enclosure 2 to the LAR, the licensee stated that:

> The Containment Leakage Rate Testing Program is discussed in ONS Technical Specifications Section 5.5.2. The MUR power uprate does not have any impact on the programmatic aspects of the Appendix J Program. It does not change any of the regulatory requirements of the program or change the scope of the program. The MUR power uprate does not change containment peak pressure following a large break LOCA as discussed in Section II, above.

The NRC staff reviewed the licensee's response and determined that the Appendix J integrated leakage rate test/local leak rate test (ILRT/LLRT) Containment Leakage Rate Testing Program would be unchanged with the proposed MUR-PU and is, therefore, acceptable.

3.1.2.3 NRC Staff Conclusion Regarding Containment Analysis

The NRC staff determined that the current containment analyses remain bounding for the proposed MUR-PU. The NRC staff determined that the current peak containment pressure is less than the containment design pressure and that the EQ envelope would remain bounding. In addition, the previously approved analytical methods would remain acceptable. Further, the NRC staff determined that the criteria identified in Section 3.1.2.1 of this SE would remain satisfied at the MUR-PU conditions. Therefore, the NRC staff concludes that the LAR is acceptable regarding containment analysis.

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

3.1.3.1 Regulatory Evaluation

The NRC staff identified the following Oconee UFSAR Plant Design Criteria as being applicable to the proposed amendment.

UFSAR Section 3.1.11, "Criterion 11 - Control Room (Category B)," states, in part:

> The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR [Part] 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

UFSAR Section 3.1.17, "Criterion 17 - Monitoring Radioactive Releases (Category B)," states, in part:

> Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths and the facility environs for radioactivity that could be released from normal operations, from anticipated transients and from accident conditions.

UFSAR Section 3.1.18, "Criterion 18 - Monitoring Fuel and Waste Storage (Category B)," states, in part:

> Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

UFSAR Section 3.1.37, "Criterion 37 - Engineered Safety Features Basis for Design (Category A)," states, in part:

> Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

UFSAR Section 3.1.51, "Criterion 51 - Reactor Coolant Pressure Boundary outside Containment (Category A)," states, in part:

> If part of the reactor coolant pressure boundary is outside the containment, appropriate features, as necessary, shall be provided to protect the health and safety of the Public in case of an accidental rupture in that part. Determination of the appropriateness of features, such as isolation valves and additional containment, shall include consideration of the environmental and population conditions surrounding the site.

UFSAR Section 3.1.70, "Criterion 70 - Control of Releases of Radioactivity to the Environment (Category B)," states, in part:

> The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified: a) on the basis of 10 CFR [Part] 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and b) on the basis of 10 CFR [Part] 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

In its review, the NRC staff used specific criteria relevant to the evaluation of Engineered Safety Features Heating, Ventilation, and Air Conditioning (ESF HVAC) Systems found in NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR [Light Water Reactor] Edition” (SRP), Section 6.4, “Control Room Habitability System” (Reference 20); Section 6.5.2, “Containment Spray as a Fission Product Cleanup System” (Reference 21); Section 9.4.1, “Control Room Area Ventilation System” (Reference 22); Section 9.4.2, “Spent Fuel Pool Area Ventilation System” (Reference 23); Section 9.4.3, “Auxiliary and Radwaste Area Ventilation System” (Reference 24); Section 9.4.4 (Reference 25), “Turbine Area Ventilation System”; and Section 9.4.5, “Engineered Safety Feature Ventilation System” (Reference 26).

3.1.3.2 Technical Evaluation

The NRC staff reviewed the impact of implementing the proposed MUR-PU on the control room ventilation system, the spent fuel pool ventilation system, the auxiliary building ventilation system, the reactor building purge system, and the reactor building penetration room ventilation system.

In Section VI.1.F of Enclosure 2 to the LAR, the applicant described that these systems where evaluated at the proposed MUR-PU conditions and stated that these systems would remain capable of performing their intended functions without modifications and that all AOR would remain bounding.

The Control Room Ventilation and Air Conditioning Systems are described in UFSAR Section 9.4.1. There are two separate control room ventilation systems (CRVS) (one for Units 1 and 2 and a separate CRVS for Unit 3). The Chilled Water (WC) system, which serves all three units, provides WC to ensure that the heat loads, considered vital loads, are removed from the areas served by the CRVSs when air is circulated via the respective cooling coils of the associated air handling units. The CRVSs provide cooling to other areas besides the control rooms, including the cable rooms, the electrical equipment rooms, and areas designated as the control room zones. The CRVSs have a safety function to provide cooling and filtration to the operators and equipment in the control rooms and associated areas. The licensee evaluated the impact of the proposed MUR-PU on the system and stated that the CRVSs and the WC system have sufficient design and operational margin to accommodate the MUR-PU and that the systems remain bounded by the existing AOR.

The Spent Fuel Pool (SFP) Ventilation System is discussed in UFSAR Section 9.4.2 and is designed to maintain a suitable environment for the operation, maintenance, and testing of equipment and for personnel access. Two methods of exhausting air from the SFP area are provided—a filtered exhaust system and an unfiltered exhaust system. Normal operation is with the unfiltered exhaust system in operation. In the filter mode, the SFP area ventilation air passes through a filter train consisting of prefilters, high efficiency particulate filters, charcoal filter, and two 100 percent vane axial fans. The UFSAR Chapter 15 FHA in the SFP analysis does not credit filtration by the SFP Ventilation System. The licensee evaluated the impact of the proposed MUR-PU on the system and stated that the SFP Ventilation System has sufficient design and operational margin to accommodate the MUR-PU and that the system remains bounded by the existing AOR.

The Auxiliary Building (AB) Ventilation System is discussed in UFSAR Section 9.4.3 and is designed to provide a suitable environment for the operation, maintenance, and testing of equipment and for personnel access. The AB ventilation System is not credited during accident conditions.

The Reactor Building (RB) Purge System is discussed in UFSAR Section 9.4.5. The RB Purge System provides the RB with fresh air during outages to reduce airborne contaminant levels inside the RB. The RB Purge System is only used in MODES 5 and 6 or when the fuel has been completely offloaded from the reactor vessel. During MODES 1, 2, 3, and 4, the RB Purge System containment penetrations are required to be sealed closed thereby prohibiting operation of the system. The only safety function of the RB Purge System is containment isolation. The RB Purge System will continue to operate to perform its safety-related functions of maintaining containment isolation after the proposed MUR-PU. The containment isolation valve leakage test pressure, which was established to bound post-LOCA peak pressure as determined by analyses performed at 2619 (MWt) [megawatts thermal], is unchanged. The licensee stated that there would be no adverse impact to this system due to the proposed MUR-PU.

The RB Penetration Room Ventilation System is discussed in UFSAR Section 9.4.7 and is actuated by the Engineered Safeguards System. The licensee evaluated the impact of the proposed MUR-PU on the system and stated that the RB Penetration Room Ventilation System has sufficient design and operational margin to accommodate the MUR-PU and that the system remains bounded by the existing AOR.

3.1.3.3 NRC Staff Conclusion Regarding ESF HVAC

The NRC staff determined that the increase in heat loads in the ESF HVAC Systems would be minimal and bounded by the current analyses and that the criteria identified in Plant Design Criteria 11, 17, 18, 37, 51, and 70 would remain satisfied at the proposed MUR-PU conditions. The NRC staff also determined that the applicable guidance in the SRP for evaluating the increase in heat loads in the control room and on the ESF HVAC Systems has been adequately addressed. Therefore, the NRC staff concludes that the LAR is acceptable regarding ESF HVAC.

3.1.4 <u>Plant Systems</u>

3.1.4.1 Regulatory Evaluation

The NRC staff's review in the area of plant systems covers the impact of the proposed MUR-PU on the 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 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 Enclosure 2 to the LAR.

The NRC staff review below covers the impact of the proposed MUR-PU on the following major plant systems and events:

- NSSS interface systems,
- safety-related cooling water systems,

- SFP cooling analyses and systems,
- radioactive waste systems,
- flooding analyses, and
- high energy line breaks.

The NRC staff's review concerning containment and ESF HVAC systems, which are also listed in RIS 2002-03, Attachment 1, Section VI, can be found in Sections 3.1.2 and 3.1.3 of this SE, respectively. The NRC staff conducted its review to verify that the licensee's analyses bound the proposed plant operation at the proposed MUR-PU power level of 2610 MWt and that the results of the licensee's analyses related to the areas under review continue to meet the applicable acceptance criteria following the implementation of the proposed MUR-PU.

3.1.4.2 Technical Evaluation

3.1.4.2.1 NSSS Interface Systems

The NSSS interface systems include the Main Steam (MS) System, the Main Turbine-Generator, the Condensate and Main FW System, and the EFW System.

3.1.4.2.1.1 Main Steam System

The MS System is described in Section 10.3 of the UFSAR. The MS System includes not only piping from the steam generators to the main turbines, EFW pump turbines, and other loads, but also the Main Steam Safety Valves, the Main Steam Atmospheric Dump Valves, the Turbine Bypass valves, and the Moisture Separator Reheaters.

The licensee evaluated the impact of the proposed MUR-PU and stated that the MS System has sufficient design and operational margin to accommodate the MUR-PU.

Based on the above, the NRC staff concludes that there is reasonable assurance that the MS System would remain capable of performing its function following the proposed MUR-PU.

3.1.4.2.1.2 Main Turbine-Generator

As discussed in UFSAR Section 10.2, the turbine-generator converts the thermal energy of steam produced in steam generators into mechanical shaft power and then into electrical energy. The turbine-generator consists of a tandem (single shaft) arrangement of a double-flow, high pressure turbine and three identical double-flow low pressure turbines driving a direct-coupled generator at 1800 revolutions per minute (rpm).

The licensee stated that the main electrical generators were reviewed at each of the Oconee units, and it was determined that the electrical generators were acceptable for the proposed MUR-PU. The increase in megawatts electric (MWe) due to the MUR-PU can be accommodated within the present generator capability curve and would result in modest reduction in available reactive power output. Additionally, the licensee stated that the turbine-generator was reviewed and found to be acceptable for the proposed MUR-PU level and the unit design rating of 1038 MVA.

Based on the above, the NRC staff concludes that the Main Turbine-Generator can accommodate the MWe increase following the proposed MUR-PU.

3.1.4.2.1.3 Condensate and Main Feedwater System

The Condensate and Main FW System is described in Section 10.4.6 of the UFSAR. Three motor-driven hotwell pumps deliver condensate from the condenser hotwell through the condensate polishing demineralizers, the condensate and other coolers to the suction of the condensate booster pumps. Three motor-driven condensate booster pumps deliver condensate through four stages of feedwater heating to the main feedwater pumps. Two steam turbine-driven main FW pumps deliver FW through two high pressure heaters to a single FW distribution header where feedwater is divided into two single lines to the steam generators.

The licensee stated that a comparison between operating requirements for proposed MUR-PU conditions and current conditions demonstrates that the Condensate and Main FW System has sufficient design and operational margin to accommodate the MUR-PU. The proposed MUR-PU conditions remain bounded by design as described in the Oconee UFSAR.

The NRC staff evaluated the justification provided in the LAR and determined that the systems and components credited to provide containment isolation capability were evaluated in the AOR at 102 percent RTP, which bounds the proposed MUR-PU. The Condensate and Main FW system operating parameters would not significantly change at proposed MUR-PU conditions; therefore, the NRC staff concludes that the Condensate and Main FW system would remain capable of performing its function following the proposed MUR-PU.

3.1.4.2.1.4 Emergency Feedwater System

The EFW System is described in Section 10.4.7 of the UFSAR. It includes two Alternating Current (AC) motor-driven pumps and one turbine-driven pump per unit that is independent of AC power. The EFW System provides sufficient FW supply to the steam generators of each unit during events that result in a loss of main FW. EFW removes energy stored in the core and primary coolant. The accident analyses crediting EFW were evaluated at 102 percent RTP and bound the proposed MUR-PU.

The licensee stated that the current design and capabilities of the EFW System would remain bounding for the proposed MUR-PU conditions. The MUR-PU conditions do not impose any necessary changes to system design flow rates, volumes, temperatures, or pressures.

Based on the above, the NRC staff concludes that the EFW System would remain capable of performing its function following the proposed MUR-PU.

3.1.4.2.1.5 NRC Staff Conclusion Regarding the NSSS Interface Systems

The NRC staff reviewed the information and evaluations provided by the licensee showing that the design of the NSSS interface systems at the increased MUR-PU power level would be bounded by existing plant analyses and, based on this information, determined that they are acceptable. The licensee indicated that there would be no adverse impact on the NSSS interface systems from the proposed MUR-PU because there is sufficient operating margin to produce an additional 1.64 percent RTP. The NRC staff determined that the proposed MUR-PU would 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 (CC) System, Condenser Circulating Water (CCW) System, High Pressure Service Water (HPSW) System, Low Pressure Service Water (LPSW) System, Standby Shutdown Facility (SSF) Auxiliary Service Water (ASW) System, Protected Service Water (PSW) System, and Recirculated Cooling Water (RCW) System.

3.1.4.2.2.1 Component Cooling System

The CC System is described in Section 9.2.1 of the UFSAR. The CC System provides closed-loop cooling water to various components in the RB as follows: letdown coolers, RCP cooling jacket and seal coolers, quench tank cooler, and control rod drive cooling coils. The heat that is transferred to the CC water from these heat exchangers is in turn transferred from the CC water to the LPSW System in the Component Coolers.

The licensee stated that the CC System's only safety function is to provide containment isolation to ensure that RB atmosphere leakage is minimized during a DBA.

The CC System will continue to perform its safety function of containment isolation. The licensee stated that the margin between the design and operating temperatures and pressures is large enough to allow for the increase in power level from the current power level to the proposed MUR-PU power level and that there is no impact to this system due to the MUR-PU.

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

3.1.4.2.2.2 Condenser Circulating Water System

The CCW System is described in Section 9.2.2.2.1 of the UFSAR and consists of four pumps per unit which take water from Lake Keowee via the intake canal to supply plant systems that use raw water.

The licensee stated that the CCW System performs the following safety functions:

- Provides a suction source for the LPSW pumps during normal operations and emergencies. Ensures suction is provided to LPSW Pumps during loss of power to all CCW Pumps.
- Provides a suction source for the PSW booster pump via water in the Unit 2 inlet piping. The CCW System is capable of transferring water from all three units' CCW inlet and discharge piping to support 30 days of decay heat removal during a loss of lake event.
- Provides a suction source for SSF ASW System via water in the Unit 2 CCW inlet piping. Ensures flow paths are maintained for makeup to CCW piping via SSF submersible pumps.
- Provides a suction source for HPSW pumps to support fire suppression for 4 hours during a fire.

The licensee stated that the MUR-PU would have no impact on the LPSW suction source. Suction to the LPSW pumps is provided from connections to the CCW inlet crossover piping. The fluid conditions in this crossover are not impacted by the proposed MUR-PU conditions.

In Section VI.1.C.ii of Enclosure 2 to the LAR, the licensee stated:

> The MUR power uprate will have no impact on the PSW System suction source. Suction to the PSW booster pump is provided from a connection to the CCW Unit 2 inlet piping. The fluid conditions in the inlet piping are not impacted by the MUR power uprate conditions. The water available to the PSW booster pump in the CCW piping will continue to last approximately 30 days because the volume available in the piping is unchanged due to the MUR power uprate and the volume required is based on decay heat loads assuming 102 [percent] of 2568 MWt.
>
> The MUR power uprate will have no impact on the SSF ASW System suction source. Suction to the SSF ASW System is provided from a connection to the CCW Unit 2 inlet piping. The fluid conditions in the inlet piping are not impacted by the MUR power uprate conditions. Also, the CCW System configuration is not impacted by the MUR power uprate; therefore, the CCW System can still ensure a flow path to the SSF submersible pumps is maintained.
>
> The MUR power uprate will have no impact on HPSW suction source. Suction to the HPSW pumps is provided from connections to the CCW inlet crossover piping. The conditions of the fluid in this crossover are not impacted by the new MUR power uprate conditions. Also, the CCW System configuration is not impacted by the MUR power uprate; therefore, still allowing the CCW System to supply suction to the HPSW pumps for a minimum of two hours.
>
> A comparison between operating conditions for the 2610 MWt MUR power uprate and the current 2568 MWt conditions demonstrates that the Condenser Circulating Water System has sufficient design margin to accommodate the MUR. The safety functions of the Condenser Circulating Water System have been determined not to be adversely impacted by the MUR. The systems remain bounded by the existing analyses of record.

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

3.1.4.2.2.3 High Pressure Service Water System

The HPSW System is described in Section 9.2.2.2.2 of the UFSAR and supplies raw lake water for fire protection and cooling/sealing of various loads.

The HPSW System performs the following safety and fire protection functions:

- Prevents air in-leakage from air binding the LPSW pumps if the elevated water storage tank is depleted.
- Provides a source of water for fire suppression systems. This is a regulatory requirement.

- Provides fire protection during all plant conditions, for both safety and non-safety related SSCs.

The licensee stated that the safety and fire protection functions of the HPSW System would not be adversely impacted by the MUR-PU and that the system would remain bounded by the existing AOR.

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

3.1.4.2.2.4 Low Pressure Service Water System

The LPSW System is described in Section 9.2.2.2.3 of the UFSAR and is designed to provide cooling water for normal and emergency services throughout the station. Oconee Units 1 and 2 share three LPSW pumps and Unit 3 has two LPSW pumps.

The LPSW System performs the safety-related function of providing cooling to the low pressure injection coolers, the high pressure injection pump motor bearing coolers, the motor-driven EFW pump motor air cooler, the RB cooling units, and the siphon seal water headers.

The licensee stated that the proposed MUR-PU conditions do not impose any necessary changes to system design and capabilities of the LPSW System and that components would remain bounding for the MUR-PU conditions.

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

3.1.4.2.2.5 Standby Shutdown Facility Auxiliary Service Water System

The SSF is described in Section 9.6 of the UFSAR. The ASW System portion of the SSF is a high head, high volume system designed to provide sufficient steam generator inventory for adequate decay heat removal for all three units. The Unit 2 CCW piping serves as the supply source for the SSF ASW System. The SSF, which includes the SSF ASW System, serves as a backup for existing safety systems to provide an alternate and independent means to achieve and maintain Mode 3. The SSF is capable of maintaining all three units at Mode 3 for 72 hours. The licensee stated that the SSF ASW System has no functions related to the design basis events described in Chapter 15 of the UFSAR.

The licensee stated that the current design and capabilities of the SSF ASW System and components remain bounding for the proposed MUR-PU conditions and that the MUR-PU does not impose any necessary changes to system design flow rates, volumes, temperatures, or pressures.

The NRC staff reviewed the information provided in the LAR regarding the SSF ASW System and determined that it will continue to meet its safety functions upon implementation of the proposed MUR-PU.

3.1.4.2.2.6 Protected Service Water System

As described in Section 9.7 of the UFSAR, the PSW System is designed as a standby system for use under emergency conditions. The PSW System design includes a dedicated power system and independent control functions and provides additional defense-in-depth protection by serving as a backup to existing safety systems. The PSW System is provided as an alternate means to achieve and maintain safe shutdown conditions for one, two, or three units following certain postulated scenarios. The PSW System reduces fire risk by providing a diverse power supply to power safe shutdown equipment in accordance with the NFPA-805 safe shutdown analyses. The PSW System requires manual activation and can be activated if normal emergency systems are unavailable.

The licensee stated that the PSW System is not adversely impacted by the MUR-PU and that the PSW System can continue to perform all required functions at the proposed MUR-PU conditions while remaining within its current design limits.

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

3.1.4.2.2.7 Recirculated Cooling Water System

The RCW System, which performs no safety function, is described in Section 9.2.2 of the UFSAR and supplies corrosion-inhibited closed-loop cooling water to various primary and secondary components in the Auxiliary and Turbine Buildings. The RCW System performs a risk significant function for decay heat removal from the SFP while the core is offloaded. Major components within the RCW System include surge tanks, RCW pumps, RCW heat exchangers, and Spent Fuel Coolers. The Spent Fuel Coolers are analyzed at 102 percent RTP. The RCW System also includes several minor heat exchangers which provide cooling to both primary and secondary components in the Auxiliary and Turbine Buildings.

The NRC staff reviewed the information provided in the LAR regarding the RCW System and determined that it will continue to meet its functions upon implementation of the proposed MUR-PU.

3.1.4.2.2.8 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 were evaluated with sufficient margin such that the AOR remains bounding. Therefore, the NRC staff concludes that there is reasonable assurance that the systems will perform acceptably after implementation of the proposed MUR-PU.

3.1.4.2.3 SFP Storage and Cooling

The licensee addressed the effects of the proposed MUR-PU on the SFP storage and cooling systems in Section VI.1.D of Enclosure 2 to the LAR.

3.1.4.2.3.1 Refueling System

The Refueling System (RFS), which is comprised of the spent fuel storage and fuel handing systems, is described in Sections 9.1.2 and 9.1.4 of the UFSAR and consists of plant facilities for storing both new and spent fuel as well as a means for transferring fuel to and from the RB from the SFP. The RFS does not perform a safety-related function with respect to safe shut down of any of the units. The licensee stated that because the RFS equipment handles nuclear fuel with the potential to release radioactive fission products if damage occurs from a fuel handling accident, the system is considered “risk significant” from the standpoint of offsite dose limits.

The RFS will continue to perform its risk significant functions of storing new and spent fuel in the SFPs and transporting fuel into and out of the RB. The licensee stated that the existing analysis for determining radiation levels of spent fuel was performed at 2619 MWt. This analysis bounds radiation levels to be encountered by the fuel storage racks at the proposed MUR-PU power level. Spent fuel being stored in the SFP after being irradiated at the higher power level associated with the MUR-PU will be maintained in the storage racks in a subcritical condition.

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

3.1.4.2.3.2 Spent Fuel Cooling System

The Spent Fuel Cooling (SF) System is described in Section 9.1.3 of the UFSAR. Oconee has two SFPs, one for Unit 1 and 2 and one for Unit 3. Each SFP is composed of three pumps, three heat exchangers, filters, valves, and interconnecting piping whose function includes cooling, purifying, and maintaining water level in the SFPs and the refueling canal.

The licensee stated that the SF System does not perform a safety-related function; however, it is credited with meeting the Extensive Damage Mitigation Strategy condition H of the licenses for ONS Units 1, 2, and 3.

The licensee stated that the SF System will continue to perform its risk significant functions of spent fuel decay heat removal and SFP inventory control after the proposed MUR-PU. Analysis demonstrates that the increase in SFP heat load resulting from fuel assumed to have been irradiated at maximum thermal power of 2619 MWt is still within the design parameters of the SF System and its components following the MUR-PU.

The NRC staff determined that the proposed MUR-PU will not result in a significant change to the operation of the SF System. Therefore, the NRC staff finds that the SF System will not be impacted by implementation of the MUR-PU.

3.1.4.2.4 Radioactive Waste Management Systems

The radioactive waste management systems are described in Chapter 11 of the UFSAR. Section VI.1.E of Enclosure 2 to the LAR describes the licensee’s evaluation of the impact of the proposed MUR-PU on the radioactive waste management systems.

The Gaseous Waste Disposal (GWD) System contains waste gases and its failure is addressed in safety analyses. The GWD System containment isolation function does not directly or indirectly interface with the steam cycle and, therefore, is not impacted by the MUR-PU.

The Liquid Waste Disposal System piping provides pressure boundary piping and containment isolation functions for mitigating events. The system is also credited to store and minimize leakage of radioactive fluid to the environment. With no direct interface with the steam cycle, these system functions are unaffected by the MUR-PU.

The Solid Waste Management System is designed to contain solid radioactive waste materials that are produced in the station and to provide for their storage and preparation for eventual shipment to an NRC or Agreement State licensed offsite disposal facility. The Solid Waste Management System has no direct interface with the power cycle and, therefore, the MUR-PU will have no impact on this system.

Based on the above and evaluations indicating the radioactive waste management systems are unaffected by the proposed MUR-PU power level, the NRC staff concludes that there is reasonable assurance that the radioactive waste management systems will perform acceptably after implementation of the MUR-PU.

3.1.4.2.5 Flooding Analyses

In Section II.1.D.iii.22 of Enclosure 2 to the LAR, the licensee stated that the NRC-approved flooding analyses for the SSF Event Turbine Building and an External Flood bounded the proposed MUR-PU conditions.

A reconstituted ONS Turbine Building Flood Mitigation Strategy was approved by the NRC on December 17, 2018 (Reference 27). Transient analyses in support of the Turbine Building Flood mitigation were performed using the RETRAN-3D computer code. The NRC-approved licensing basis for Turbine Building Flood events mitigated by the SSF includes events occurring when the Oconee units are not at nominal full-power conditions that allows use of off-nominal success criteria. Turbine Building Flood events initiated at full-power conditions were analyzed with an initial reactor power of 102 percent RTP, which bounds the proposed MUR-PU conditions.

Transient analyses in support of External Flood mitigation strategies due to a postulated Jocassee Dam failure were performed using the RELAP5/MOD2-B&W computer code. These analyses were evaluated as part of the mitigating strategies for Beyond-Design Basis external events which credit use of FLEX equipment. The NRC SE for these mitigating strategies is documented in a letter dated August 30, 2017 (Reference 28). These events were analyzed with an initial reactor power of 102 percent RTP, which bounds the proposed MUR-PU conditions.

Based upon the information provided by the licensee to show that the effects of flooding at the proposed MUR-PU power level described in the LAR are bounded by existing plant analyses, the NRC staff concludes that the flooding analyses will remain acceptable following implementation of the MUR-PU.

3.1.4.2.6 High Energy Line Breaks

In Section II.1.D.ii.29 of Enclosure 2 to the LAR, the licensee discussed the HELB analyses performed in support of the proposed MUR-PU.

Transient analyses were performed using the RELAP5/MOD2-B&W computer code using NRC reviewed and approved methods. The HELB analyses assumed an initial reactor power of 102 percent RTP, which bounds the proposed MUR-PU conditions.

The NRC staff reviewed the information provided in the LAR regarding the impact of the proposed MUR-PU on the HELB and determined that the HELB analyses remain acceptable.

3.1.4.3 NRC Staff Conclusion Regarding Plant Systems

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

3.1.5 Accident Analysis

3.1.5.1 Regulatory Evaluation

For the review of the proposed MUR-PU the NRC staff used the following:

The RIS 2002-03 guidance to licensees on the scope and detail of the information that should be provided to the NRC staff for MUR-PU LARs.

The SRP Chapter 15 (Reference 29) 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 LAR, the licensee generally concluded that existing analyses bounded the proposed MUR-PU conditions with reduced uncertainty. The analyses were shown to be bounding in different ways:

- For analyses that assume steady-state plant operation with a core power of 2619 MWt, the licensee evaluated the accident or transient, and reanalyzed as necessary.
- Analyses that analyzed with 3 reactor coolant pumps initially operating at a power level of 75 percent RTP were reanalyzed to use a power level of 73.8 percent RTP, consistent with the proposed change to TS 3.4.4.
- Zero-power transients were not reanalyzed.
- Reload analyses are performed for each fuel cycle in accordance with normal cycle design practice and included in the Core Operating Limits Report in accordance with TS 5.6.5.

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

RIS 2002-03 states the following:

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

The NRC staff's review utilized the approach discussed above. The licensee's analyses that were performed at 102 percent RTP. 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.

The Table 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.

Evaluation of Accident and Transient Analyses

Transient/Accident	**Analytic Power Level (percent RTP)**	**Review Determination**
Start-up Accident	0 MWt	Acceptable
Rod Withdrawal At Power Accident	2619 MWt (102% of 2568)	Acceptable
Moderator Dilution Accidents	Mode 1, Mode 6	Acceptable
Cold Water Accident	2054 MWt (80% of 2568)	Acceptable
Loss of Coolant Flow Accidents – Flow Coastdown	2054 MWt (80% of 2568) 2619 MWt (102% of 2568)	Acceptable
Loss of Coolant Flow Accidents – Locked Rotor	1926 MWt (75% of 2568) 2619 MWt (102% of 2568)	Acceptable

Transient/Accident	**Analytic Power Level (percent RTP)**	**Review Determination**
Control Rod Misalignment Accidents (Dropped Rod)	1926 MWt (75% of 2568) 2619 MWt (102% of 2568)	Acceptable
Turbine Trip Accident	2105.8 MWt (82% of 2568) 2619 MWt (102% of 2568)	Acceptable
Steam Generator Tube Rupture Accident	2619 MWt (102% of 2568)	Acceptable
Waste Gas Tank Rupture Accident	2619 MWt (102% of 2568)	Acceptable
Fuel Handling Accidents	2619 MWt (102% of 2568)	Acceptable
Rod Ejection Accident	0 MWt 1977 MWt (77% of 2568) 2619 MWt (102% of 2568)	Acceptable
Steam Line Break Accident	2619 MWt (102% of 2568)	Acceptable
Loss of Coolant Accident	2619 MWt (102% of 2568)	Acceptable
Maximum Hypothetical Accident	2619 MWt (102% of 2568)	Acceptable
Post-Accident Hydrogen Control	2619 MWt (102% of 2568)	Acceptable
Small Steam Line Break Accident	1926 MWt (75% of 2568) 2619 MWt (102% of 2568)	Acceptable
Anticipated Transients Without Scram	2619 MWt (102% of 2568)	Acceptable
Natural Circulation Cooldown	0 MWt	Acceptable
Containment Performance	2619 MWt (102% of 2568)	Acceptable
EQ parameters	2619 MWt (102% of 2568)	Acceptable

Transient/Accident	Analytic Power Level (percent RTP)	Review Determination
SSF Event Turbine Building Flood (TBF)	2619 MWt (102% of 2568)	Acceptable
External Flood	2619 MWt (102% of 2568)	Acceptable
NFPA-805 Fire	2619 MWt (102% of 2568)	Acceptable
Spent Fuel Pool Accidents (loss of pool cooling)	Decay Heat	Acceptable
Loss of Main Feedwater	2619 MWt (102% of 2568)	Acceptable
Main Feedwater Line Break	2619 MWt (102% of 2568)	Acceptable
Low-Temperature Overpressure Protection (LTOP)	0	Acceptable
High Energy Line Break/Pipe Rupture (HELB)	2619 MWt (102% of 2568)	Acceptable
RPS/ESF Instrument Uncertainties	0 MWt	Acceptable
Natural: Tornado, Wind, Hurricane	2619 MWt (102% of 2568)	Acceptable
Double Steam Line Break with SSF	2619 MWt (102% of 2568)	Acceptable

LOCA Analysis

The current LOCA analyses in the UFSAR was reviewed for the impact of the proposed MUR-PU. A failure of the RCS pressure boundary will result in a loss of primary coolant inventory and the potential for the core to uncover. These hypothetical failures are considered to occur in all piping and components up to and including a double-ended rupture of the largest pipe in the system. If the core is not rapidly reflooded and long-term heat removal established, decay heat will cause the fuel cladding to fail and release the fission product inventory. The Emergency Core Cooling System (ECCS) is designed to deliver sufficient cooling to provide the necessary core decay heat removal for all credible LOCA. The large break loss-of-coolant accident (LBLOCA) and small break loss-of-coolant accident (SBLOCA) were analyzed with the RELAP5/MOD2-B&W computer code. The LBLOCA and SBLOCA evaluation model has been shown to conform to the requirements of 10 CFR Part 50, Appendix K.

The loss-of-coolant accidents analyzed have been reviewed for the impact of the proposed MUR-PU. Based on the power levels assumed in the current safety analyses, the licensee determined that all LOCA analyses bound the MUR-PU. Since the proposed change relies on less than 0.4 percent uncertainty, the nominal power level of 100.34 percent of 2610 MWt is bounded by the analysis power of 2619 MWt, and all five criteria of 10 CFR 50.46 continue to be met following a LOCA initiated at the post-MUR-PU power level. Therefore, LOCA analyses performed at this power remain applicable. The LOCA analyses described in UFSAR Section 15.14 were performed with NRC-approved methods.

A complete spectrum of LOCAs has been analyzed conservatively with the NRC-approved evaluation models which conform to 10 CFR Part 50, Appendix K. The results of these analyses meet the acceptance criteria of 10 CFR 50.46. Therefore, the consequences of all design basis LOCAs have been shown to be acceptable.

Departure from Nucleate Boiling Analyses in UFSAR Chapter 15

There are several analyses in the ONS UFSAR Chapter 15 that are evaluated for Departure from Nucleate Boiling Ratio (DNBR) concerns. These DNB events are evaluated using Duke Energy's Statistical Core Design (SCD) methodology. The treatment of core power initial conditions for SCD analyses are described in DPC-NE-3005-PA. DPC-NE-2005-PA describes how a core power uncertainty of 2 percent is combined with other statistically treated uncertainties when developing the statistical DNBR design limit used in SCD analyses for the Mark-B-HTP fuel at ONS. No credit was taken for reduced core power uncertainty due to installation of the LEFMs when establishing the statistical DNBR design limit for SCD analyses. Therefore, the NRC staff determined that the proposed changes are acceptable.

3.1.5.3 NRC Staff's 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 that the licensee stated were based on operation at 102 percent RTP, the NRC staff determined that the existing analyses will continue to bound plant operation after implementation of the proposed MUR-PU. Therefore, the NRC staff concludes that for the MUR-PU, the Chapter 15 events would meet the applicable acceptance criteria.

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, the 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 has 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 pressurized-water reactors (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 reference nil-ductility transition temperature for pressurized thermal shock (RT_{PTS}). The regulation in 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). Specifially, 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. The regulations in 10 CFR 50.61(b)(1) require that PWR licensees have projected values of RT_{PTS} accepted by the NRC for each RV beltline material. The regulations in 10 CFR 50.61(c)(2) require 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). The regulations in 10 CFR 50.61(b) specify 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 RIS 2014-11, "Information on Licensing Applications for Fracture Toughness Requirements for Ferritic Reactor Coolant Pressure Boundary Components" (Reference 30).

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 MUR-PU 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

<u>Licensee Evaluation</u>

The licensee discussed the impact of the MUR-PU on the Oconee Units 1, 2, and 3, PTS evaluation in Section IV.1.C.i of Enclosure 2 to its LAR. The licensee's current licensing basis (CLB) RT_{PTS} calculations were performed using the procedures required by 10 CFR 50.61 for 48 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 Oconee Units 1, 2, and 3, RV beltline materials using 72 EFPY with MUR-PU neutron fluence values. The licensee provided a summary of the RT_{PTS} calculations associated with the MUR-PU for all Oconee Units 1, 2, and 3, RV beltline materials in Tables IV.1-2, IV.1-3, and IV.1-4 of Enclosure 2 to the LAR. The licensee stated 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 72 EFPY. The licensee further stated, while 72 EFPY with MUR-PU fluences were used in the RT_{PTS} calculations, the submittal does not extend the applicability of these analyses beyond the CLB duration of 48 EFPY for RT_{PTS}.

NRC Staff Evaluation

The CLB RT_{PTS} values for the Oconee Units 1, 2, and 3, RV beltline materials are provided in its license renewal application for 48 EFPY (Reference 31). All CLB 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 Oconee Units 1, 2, and 3 RV beltline materials using the procedures required by 10 CFR 50.61 and incorporating 72 EFPY with MUR-PU neutron fluence values. The NRC staff's technical review of the licensee's neutron fluence evaluation is discussed in Section 3.2.6 of this SE. The NRC staff verified that the RT_{PTS} values for all Oconee Units 1, 2, and 3, RV beltline materials will continue to remain less than the applicable PTS screening criteria.

The NRC staff concludes that after implementation of the MUR-PU, the Oconee Units 1, 2, and 3, RV beltline materials would continue to meet the PTS screening criteria requirements described in 10 CFR 50.61 and maintain structural integrity during a postulated PTS event.

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 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. 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 shift 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, "Radiation Embrittlement of Reactor Vessel Materials" (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 1×10^{17} n/cm^2 (E > 1.0 MeV). The NRC staff's basis for this fluence threshold is provided in RIS 2014-11.

For MUR-PU amendment 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 Oconee Units 1, 2, and 3 P-T limits in Section IV.1.C.iii of Enclosure 2 to its LAR. The CLB P-T limits were developed in accordance with 10 CFR Part 50, Appendix G and are applicable for 54 EFPY. Table IV.1-5 of Enclosure 2 to the LAR shows the limiting adjusted reference temperature (ART) values for the Oconee Units 1, 2, and 3 reactor vessel materials used in its CLB P-T limits. The licensee stated that the updated 54 EFPY fluence values with the MUR-PU are greater than the fluences used to support the CLB P-T limits for numerous reactor vessels. To address MUR-PU conditions, the licensee proposed to maintain its limiting ART values and resulting CLB P-T limits by reducing the EFPY term corresponding to the limiting ART values used in the existing analysis. The licensee determined that the MUR-PU will result in the limiting ART values used for the CLB P-T limit calculations being reached at 44.6 EFPY, 45.3 EFPY, and 43.8 EFPY for Oconee, Units 1, 2, and 3, respectively. Therefore, the licensee proposed to decrease the EFPY applicability term for the CLB P-T limits to these values.

The licensee stated that the decrease in EFPY applicability terms for the CLB P-T limits are not due solely to increase in fluence from MUR-PU but is due primarily to consideration of RV nozzles on the bounding P-T limits and RIS 2014-11. RIS 2014-11 states, in part, "the beltline definition in 10 CFR Part 50, Appendix G is applicable to all reactor vessel ferritic materials with projected neutron fluence values greater than 1×10^{17} n/cm^2 (E > 1.0 MeV), and this fluence threshold remains applicable for the design life as well as throughout the licensed operating period." The licensee stated that the EFPY applicability terms were reduced until the RV nozzle fluence reached 1×10^{17} n/cm^2 (E > 1.0 MeV) in order to maintain the CLB P-T limits.

In addition to its P-T limits evaluation, the licensee also discussed the impact of the MUR-PU on its LTOP system requirements in Section IV.1.C.iv of Enclosure 2 to its LAR. Since MUR-PU conditions are addressed through a reduction to the EFPY applicability term for the CLB P-T limits, the licensee determined that the current LTOP limits in the existing P-T limits do not need to be modified beyond the same reduction in EFPY applicability.

NRC Staff Evaluation

For a given EFPY operating term, an increase in the rate of neutron fluence accumulation (i.e., neutron flux) for MUR-PU conditions will result in an increase in the limiting ART values used for calculating P-T limits, thereby requiring a rightward shift in the P-T limits for that EFPY term. For MUR-PU amendment applications, licensees frequently choose to maintain their CLB P-T limits for uprated conditions by recalculating the EFPY term corresponding to the neutron fluence and ART 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 MUR-PU, 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 EFPY applicability term for the CLB P-T limits based upon the RV nozzles.

The LTOP system applicability temperature is determined based on the ART for the limiting RV beltline material, per the ASME Code, Section XI, Appendix G. Thus, the EFPY term for the LTOP limits should be consistent with that used to determine limiting ART value. The NRC staff determined that the licensee's reduction to the EFPY term for the LTOP system applicability temperature, consistent with that used for the limiting ART values and P-T limits, ensures that the applicability temperature is valid for MUR-PU conditions.

Therefore, the NRC staff concludes that the licensee's CLB P-T limits would remain in compliance with 10 CFR Part 50, Appendix G, for the proposed MUR-PU conditions through 44.6 EFPY, 45.3 EFPY, and 43.8 EFPY for Oconee Units 1, 2, and 3, respectively.

3.2.1.3 Upper-Shelf Energy

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, also 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, Revision 2. For MUR-PU amendment 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 provided the USE evaluation in Section IV.1.C.v of Enclosure 2 to its LAR. The CLB USE at the ¼-thickness wall locations for Oconee Units 1, 2, and 3, RV beltline and extended beltline materials were calculated per RG 1.99, Revision 2 and are for 54 EFPY. To evaluate the effects of the proposed MUR-PU on USE, the licensee recalculated the USE evaluations using 72 EFPY with MUR-PU neutron fluence values. The licensee provided a summary of the USE calculations associated with the MUR-PU for all Oconee Units 1, 2, and 3, RV beltline and extended beltline materials in Tables IV.1-6, IV.1-7, and IV.1-8 of Enclosure 2 to the LAR.

The licensee identified the beltline and extended beltline materials with a projected USE value above 50 ft-lbs at 72 EFPY with the proposed MUR-PU. The licensee further stated that, while 72 EFPY with MUR-PU fluences were used in the USE calculations, the submittal does not extend the applicability of these analyses beyond the CLB duration of 54 EFPY for USE.

The licensee identified two materials in Oconee Unit 3 that required further discussion, as designated in Table IV.1-8. For the Oconee Unit 3 dutchman forging, the licensee evaluated the dutchman forging using 54 EFPY with MUR-PU fluences and determined that the predicted USE value remained above 50 ft-lbs. For the Oconee Unit 3 outlet nozzle forgings, the licensee noted that the CLB fluence was less than 1×10^{17} n/cm^2 and, therefore, did not conduct an embrittlement assessment. The licensee noted that the limiting postulated flaw in the outlet nozzle is a corner flaw. The licensees stated that the projected EFPY with MUR-PU fluence to reach 1×10^{17} n/cm^2 for a postulated corner flaw in the Oconee Unit 3 outlet nozzle is 43.8 EFPY. The licensee reduced the applicability of the Oconee Unit 3 USE and EMA evaluations considering the MUR-PU to 43.8 EFPY. The Oconee Units 1 and 2 USE and EMA evaluations are applicable for 54 EFPY with the proposed MUR-PU.

The licensee identified the Oconee Units 1, 2, and 3, beltline and extended beltline materials with a projected USE value below 50 ft-lbs at 54 EFPY with the proposed MUR-PU neutron fluence values, as designated in Tables IV.1-6, IV.1-7, and IV.1-8 with "EMA." The licensee stated that it performed an equivalent margin analysis for these locations to demonstrate that the lower USE values would provide acceptable margins of safety against fracture toughness. The licensee stated that it completed the EMA using the projected 72 EFPY (without MUR-PU neutron fluence) as documented in BAW-2192, Revision 0, Supplement 1P-A, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels A & B Service Loads," (Reference 34) and BAW-2178, Revision 0, Supplement 1P-A, Revision 0, "Low Upper-Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of B&W Owners Reactor Vessel Working Group for Levels C & D Service Loads" (Reference 35). The NRC staff approved these EMAs, as discussed in the SE included in each report. Because the 72 EFPY fluences used in these evaluations bound the 54 EFPY fluences with the MUR-PU, the licensee stated that these identified beltline and extended beltline materials provide adequate margins of safety through 54 EFPY with the MUR-PU. The licensee further stated that, while 72 EFPY fluences were used in these EMA calculations, the submittal does not extend the applicability of these analyses beyond the CLB duration of 54 EFPY.

NRC Staff Evaluation

The NRC staff verified independently that the licensee appropriately recalculated the projected USE values for all of the RV beltline and extended beltline materials to address the proposed MUR-PU conditions for 54 EFPY. The NRC staff verified that the licensee's 54 EFPY USE calculations for the MUR-PU conditions demonstrate that the RV beltline and extended materials will maintain USE values greater than 50 ft-lbs or that the lower values of USE would provide margins of safety against fracture. The NRC staff determined that the licensee's reduction to the EFPY term for Oconee Unit 3 applicability ensures that the USE and EMA evaluations will be valid for Oconee Units 1, 2, and 3 for the MUR-PU conditions. Therefore, the NRC staff concludes that the licensee's USE and EMA evaluation for the Oconee Units 1, 2, and 3, RV beltline and extended materials would continue to meet the requirements of 10 CFR Part 50, Appendix G, for the proposed 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, provide 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, require 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 MUR-PU amendment 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 proposed MUR-PU on the surveillance capsule withdrawal schedules in Section IV.1.C.vi of Enclosure 2 to its LAR. The licensee stated that it participates in the Pressurized Water Reactor Owner's Group Master Integrated Reactor Vessel Surveillance Program (MIRVP). By letter dated March 27, 2018 (Reference 38), the NRC concluded that the MIRVP met the criteria in 10 CFR Part 50, Appendix H and that the current withdrawal schedule satisfies ASTM E185-82. The licensee stated that a “fifth” surveillance capsule supports the period of extended operation for Oconee Units 1, 2, and 3 and provided the respective fluences. The licensee stated that the updated 54 EFPY fluence values with the MUR-PU are less than the fluence values for these capsules. The licensee stated that this comparison shows that the current surveillance capsule withdrawal schedule will continue to meet ASTM E185-82, which states that the capsule may be removed when the capsule neutron fluence is between one and two times the limiting fluence calculated for the vessel at end of life.

NRC Staff Evaluation

The NRC staff independently confirmed that the licensee evaluated the increase in fluence as a result of the proposed MUR-PU and verified that the surveillance capsule withdrawal schedule remains acceptable for meeting ASMT E185-82 as required by 10 CFR Part 50, Appendix H. The NRC staff also verified that the surveillance capsule withdrawal schedule meets the recommended criteria in GALL Revision 2. Therefore, the NRC staff concludes that the license’s RV material surveillance program would continue to meet the requirements of 10 CFR Part 50, Appendix H, for the proposed 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 managed adequately so that the intended design functions will be maintained consistent with the CLB 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 Revision 2 are established in the Electric Power Research Institute (EPRI) Materials Reliability Program Topical Report MRP-227, Revision 1-A (Reference 39).

3.2.1.5.2 Technical Evaluation

Licensee Evaluation

The licensee discussed the impact of the proposed MUR-PU on the structural integrity of the RV internals in Section IV.1.A.ii of Enclosure 2 to its LAR. The licensee stated that the core delta temperature will experience a nominal increase, but that the revised core parameters are bounded by the design values plus uncertainty that were used in the current analyses. The licensee also stated that it reviewed the MUR-PU conditions for impact on the existing design basis. The licensee stated that RCS pressure and operating transients will be unchanged as a result of the MUR-PU and, therefore, the MUR-PU conditions are bounded by the design conditions and existing loads, stresses, and fatigue values remain valid. The licensee stated that it reviewed its Reactor Vessel Internals Inspection Plan, which implements the inspection and evaluation guidelines in MRP-227, Revision 1-A, and determined that it will not be affected by the MUR-PU.

NRC Staff Evaluation

The NRC staff reviewed the licensee's evaluation to assess the impact of the proposed MUR-PU on the effectiveness of ISI and PWR internals AMP activities for ensuring RV internal component functionality during periods of extended operation. The NRC staff noted that the small changes to the RCS fluid temperatures and service conditions (e.g., loads, stresses, fatigue cycles) for the MUR-PU will not invalidate the existing analyses of the structural integrity and functional performance of the RV internals components. Therefore, the NRC staff concludes that the MUR-PU will not have an adverse impact on the effectiveness of ISI and PWR internals 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 proposed 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 program criteria required by 10 CFR Part 50, Appendix H. The NRC staff determined that the licensee's evaluation demonstrates that these requirements will continue to be satisfied following implementation of the MUR-PU at Oconee, Units 1, 2, and 3. The NRC staff also 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 determined that 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 Oconee, Units 1, 2, and 3. 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

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

3.2.1.6.1 Regulatory Evaluation

Regulations in 10 CFR 50.61 describes the requirements for maintaining the integrity of the reactor pressure vessel with respect to PTS. The NRC RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," (Reference 40) provides guidance on methods for determining reactor pressure vessel fluence that are acceptable to the NRC staff, based on the requirements identified above.

3.2.1.6.2 Technical Evaluation

The fracture toughness of pressure vessel materials is related to a parameter called the material's reference temperature for nil-ductility transition. The reference temperature is defined by a correlation of the fluence, material chemistry, initial reference temperature, and margin to account for uncertainties in the correlation and input values. Methods for determining the fast neutron fluence are therefore necessary to estimate the fracture toughness of the pressure vessel materials. A slight increase in core thermal power associated with the proposed MUR-PU can lead to an increase in the core peripheral neutron flux, which would increase the neutron flux on the reactor vessel and, accordingly, cause the fluence to increase over time.

The licensee stated that the fluence was calculated in accordance with BAW-2241P-A, Revision 2, "Fluence and Uncertainty Methodologies" (Reference 41). The fluence values calculated by the licensee consider the effects of an MUR-PU through 72 EFPY. Note that the licensee has not received a subsequent license renewal for any of the three units, so the fluence values should be considered through 48 EFPY with the MUR-PU. However, the fluence values at 72 EFPY will be higher than the fluence values at 48 EFPY; therefore, the data provided by the licensee regarding neutron fluence considers the effect of an MUR-PU and remains bounding in regard to the 10 CFR 50.61 screening criteria. The fluence values were extrapolated out to 54 and 72 EFPY using the most recent cycle design and considered the effects of an MUR-PU.

In order to maintain the validity of the existing limiting ART and P-T limits analyses, the licensee had to reduce the applicability of the CLB from 54 EFPY to 44.6, 45.3, and 43.8 EFPY for units 1, 2, and 3, respectively. This reduction corresponds to the 1×10^{17} n/cm^2 neutron fluence criteria for embrittlement for the most limiting extended beltline material. The Oconee CLB was approved before the issuance of RIS 2014-11 and, therefore, did not consider all beltline materials, as described in RIS 2014-11. Instead, the CLB only considered core-adjacent materials. The CLB will remain applicable with the proposed MUR-PU through the above EFPYs, which corresponds to when the first extended beltline materials are projected to exceed 1×10^{17} n/cm^2.

3.2.1.6.3 NRC Staff Conclusion Regarding Neutron Fluence

Based on the above, the NRC staff concluded that the licensee used acceptable methods to estimate the fluence for the reactor vessel materials. Thus, the NRC staff determined that the licensee's fluence estimate was acceptable for accounting for the effects of irradiation on the reactor pressure vessel materials, consistent with the requirements of 10 CFR 50.61 and the guidance in RG 1.190. The NRC staff therefore concludes that the proposed MUR-PU is acceptable with respect to the reactor vessel neutron fluence.

3.2.2 Mechanical and Inservice Testing

3.2.2.1 Regulatory Evaluation

The NRC staff's review of the LAR in the area of mechanical and inservice testing (IST) focused on verifying that the licensee has provided reasonable assurance that the structural and pressure boundary integrity of the SSCs at Oconee will continue to be adequately maintained following the implementation of the proposed MUR-PU under normal, upset, emergency, and faulted operating conditions, as applicable.

The NRC staff's assessment of the Oconee LAR in the area of mechanical engineering and IST considered the following design criteria:

UFSAR Section 3.1.1, “Criterion 1 - Quality Standards (Category A),” states, in part:

> Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

UFSAR Section 3.1.2, “Criterion 2 - Performance Standards (Category A),” states, in part:

> Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area and, (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

UFSAR Section 3.1.9, “Criterion 9 - Reactor Coolant Pressure Boundary (Category A),” states, in part:

> The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

UFSAR Section 3.1.40, “Criterion 40 - Missile Protection (Category A),” states, in part:

> Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

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 the proposed 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, “Mechanical/Structural/Material Component Integrity and Design,” of RIS 2002-03 provides information to licensees on the scope and detail of the information that should be submitted to the NRC 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 area of mechanical and IST covers the structural and pressure boundary integrity of the piping, components and supports that make up the NSSS and the balance-of-plant (BOP) systems. The mechanical and IST review scope also includes an evaluation of other new or existing SSCs that 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 Oconee 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 HELB locations and corresponding dynamic effects resulting from the postulated HELBs, including pipe whipping and jet impingement. A review of the impact of the MUR-PU on Flow Induced Vibration (FIV) was also performed. The NRC staff's review focused on verifying that the licensee has provided reasonable assurance of the structural and pressure boundary integrity of the Oconee piping systems, components, component internals, and their supports under normal and transient loadings, including those due to postulated accidents and natural phenomena, such as earthquakes.

The proposed MUR-PU would increase the RTP level from 2568 MWt to 2610 MWt. In accordance with the 10 CFR Part 50, Appendix K requirements, the licensee noted in Section IV of Enclosure 2 to the LAR that the current ECCS AOR are based on a maximum analytical thermal power level of 102 percent RTP (2619 MWt), which bounds the proposed MUR-PU.

3.2.2.2.1 Power Uprate Evaluation Parameters and Design Bases

In Table IV-1 in Section IV.1 of Enclosure 2 to the LAR, the licensee provided the pertinent temperatures, pressures, and flow rates for the current and proposed MUR-PU conditions. The licensee evaluated the effects of the proposed MUR-PU at a bounding power level of 102 percent RTP (2619 MWt). This power level corresponds to the proposed level following the implementation of the proposed MUR-PU (i.e., 2610 MWt) plus the revised uncertainty of 0.34 percent.

As shown in Table IV-1, there is no change in the RCS operating pressure (2155 pounds per square inch absolute (psia)) as a result of the proposed MUR-PU. The change in RCS mechanical design flow is insignificant (an increase of 0.01 percent from 145.5×10^6 to 145.52×10^6 pounds/hour (lb/hr)) after implementation of the proposed MUR-PU. At full power, the implementation of the proposed MUR-PU would yield a hot leg temperature (T_{hot}) of 602.1 °F (from the current temperature of 601.7 °F) and a cold leg temperature (T_{cold}) of 556.0 °F (from the current temperature of 556.4 °F), resulting in no change to the average RCS temperature. The MS pressure would remain unchanged at 925 psia at the proposed MUR-PU conditions and the MS steam flow would increase from 10.91×10^6 lb/hr to 11.14×10^6 lb/hr at the proposed MUR-PU conditions. Steam temperature would decrease by 0.7 °F from 592.0 °F to 591.3 °F and the final FW temperature would increase by 3.3 °F from 455.6 °F to 458.9 °F as a result of proposed MUR-PU implementation.

The information related to the structural qualification of SSCs at Oconee is contained in Chapter 3 of the UFSAR. The UFSAR describes the design criteria applicable to the Oconee SSCs, including loads, load combinations, and acceptance criteria stipulated by the applicable codes of record for these SSCs. In Section IV.1.A of Enclosure 2 to the LAR, the licensee noted that implementation of the proposed MUR-PU would not change current operating transients, nor would it introduce additional transients. As such, loads, stresses, and fatigue values resulting from these transients that are used in the structural evaluations of SSCs would not be affected. Similarly, the proposed MUR-PU would have no effect on the deadweight and seismic loads of existing SSCs. Therefore, the NRC staff finds that the loads used in the existing AORs for these SSCs remain valid.

3.2.2.2.2 Pressure-Retaining Components and Component Supports

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

The Codes of record for Oconee are documented in Table IV.1.D-1 of Enclosure 2 to the LAR and are also stated in UFSAR, Table 5-4, “Reactor Coolant System Component Codes.” The licensee confirmed that the LAR did not revise any stress/fatigue analysis revisions, and hence no Code of record changed.

As discussed in UFSAR Section 3.7.3.10, the reactor internals for Oconee were designed as Class 1 structures. The existing AOR are based on the design conditions in the RCS functional specification and bound the MUR-PU conditions. Therefore, the RV internals remain acceptable under the MUR-PU conditions.

The pressure-retaining components and component supports, including piping and pipe supports, that must be evaluated in support of an MUR-PU include the following: the RV, including the RV shell, RV nozzles, and supports; the pressure retaining portions of the control rod drive mechanisms (CRDMs); NSSS piping, pipe supports, and branch nozzles associated with the RCS; BOP piping and supports; steam generators (SGs), including their supports, the SG shells, and secondary side internal support structures and nozzles; the pressure retaining portions of the RCPs; the pressurizer, including the pressurizer shell, nozzles, and the surge line; and safety-related valves. Further, Section IV.1.B of RIS 2002-03 indicates that the discussion should identify and evaluate any changes related to the power uprate in the following areas:

i. stresses
ii. cumulative usage factors
iii. flow induced vibration
iv. changes in temperature (pre- and post-uprate)
v. changes in pressure (pre- and post-uprate)
vi. changes in flow rates (pre- and post-uprate)

vii. high energy line break locations
viii. jet impingement and thrust forces

In reviewing the licensee's evaluation of pressure-retaining components and their supports, the NRC staff focused on determining whether those components and supports would be affected by the implementation of the proposed MUR-PU. Affected components and supports refer to those for which their AOR are 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 that are more limiting than those that 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, would affect the loads included in the AOR for the component and its supports. Based on its evaluations of the impact of proposed 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 Oconee: the RV, RV nozzles, and RV supports; the pressure retaining portions of the CRDMs; NSSS piping, pipe supports, and branch nozzles associated with the RCS; BOP piping and supports; SGs, including their supports, the SG shells, and secondary side internal support structures and nozzles; the pressure retaining portions of the RCPs; the pressurizer, including the pressurizer shell, nozzles, and the surge line; and safety-related valves.

The licensee evaluated CRDMs as shown in Section IV.1.A.iii of Enclosure 2 to the LAR. There are only small changes to the plant operating parameters, namely T-hot (hot leg temperature) and T-cold (cold leg temperature). It is also concluded that the design transient definitions and frequency of occurrences applicable to current power conditions would remain applicable for proposed MUR-PU conditions. Thus, the existing stress reports for the CRDMs would remain applicable for the uprated power conditions.

In Section IV.1.B.iv of Enclosure 2 to the LAR, the licensee evaluated the potential for thermal stratification of the pressurizer surge line. Thermal stratification in the surge line, attached to the primary side of the RCS, occurs mainly during plant heatup and cooldown and is driven by the temperature difference between the hot leg of the RCS and the pressurizer. The operating temperature of the RCS hot leg will increase very slightly above the current operating temperature due to the proposed MUR-PU. This gives a lower temperature differential between the RCS hot leg and the pressurizer, which in turn lessens the stratification effects. This means that stress and fatigue in the surge line, which is attributed to thermal stratification, is bounded by the existing analyses. The temperature changes of the RCS as a result of the proposed MUR-PU are negligible and will not have an adverse effect on the thermal stress or stratification conditions. In addition, the design RCS flow rates are unchanged for the proposed MUR-PU. Therefore, the effects of the turbulent penetration will not change as a result of the proposed MUR-PU.

The NRC staff reviewed BOP piping as discussed in Section IV.1.A.v of Enclosure 2 to 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 the proposed MUR-PU conditions for the insignificant changes in MS and FW parameters. The BOP piping systems remain acceptable for the proposed MUR-PU conditions. Based on the above, the NRC staff concludes that all pressure-retaining components including piping and pipe supports remain bounded at the proposed 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 MUR-PU 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 proposed 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 magnitude of plant parameter changes, as documented in Table IV-1 of Enclosure 2 to the LAR are generally minor and support the licensee's assessment that concludes that all pressure-retaining components remain bounded.

Based on the above, 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 proposed 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 Oconee RV internals. 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), 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 of Enclosure 2 to the LAR.

Mechanical and structural evaluations were performed by the licensee to determine any effects on the RV internals due to the conditions that would be present following the implementation of the proposed MUR-PU. The mechanical evaluations of the RV internals are summarized in Section IV.1.A.ii of Enclosure 2 to the LAR. The average temperature is unchanged, and the cold leg decreases 0.4 °F, while the hot leg temperature increases 0.4 °F. The core delta temperature will experience a nominal operating increase in order to reflect the MUR-PU, but the revised core parameters are bounded by the design values plus uncertainty that were used in the current analyses. The system operating pressures remain unchanged as shown in Table IV-1. The components currently analyzed for FIV include the in-core instrumentation nozzles, the flow distributor assembly, the thermal shield, and the inlet baffle. From the comparative analysis, the new operational conditions of Oconee after the proposed MUR-PU would be bounded by the current analysis (Topical Report BAW-10051) (Reference 42)). The RV internals and in-core instrument nozzles are structurally adequate with regard to FIV, including the effects of the proposed MUR-PU. Based on this assessment, the licensee noted that the RV internals performance after the MUR-PU would remain bounded by the current normal operation analyses.

The NRC staff has reviewed the licensee's assessment of the RV internals and considers the licensee's evaluation acceptable. Concerning the effects of the proposed MUR-PU on the FIV of the RV internals, the NRC staff finds that the licensee's assessment is acceptable given that the licensee's submittal indicates that the RCS operating parameters (flow, temperature, and pressure) that directly affect FIV, either do not change or do not change sufficiently to affect the FIV of the RV internals. For the structural evaluations, the NRC staff finds that the licensee's conclusion that the RV internals are bounded by the current AOR at the proposed MUR-PU conditions is acceptable, because the RV internals have previously been evaluated at a power level that is greater than the proposed MUR-PU power level. Additionally, a comparison between the RCS operating parameters before and after the proposed 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 operating loads (i.e. transient and seismic) would change as a result of the proposed MUR-PU. Therefore, the NRC staff concludes that the DBA of the RV internals will remain unaffected and bounding following implementation of the proposed 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 would result from the implementation of the MUR-PU. This assessment is summarized in Section IV.1.B.vii of Enclosure 2 to the LAR. The licensee stated in a summary to its assessment that the current AORs were reviewed to determine whether the proposed MUR-PU would have any impact on the current HELB AOR. The licensee concluded that because the design conditions are not changing, the stresses and cumulative usage factors would also remain unchanged and, therefore, there would be no impact on HELB locations inside and outside of containment as a result of the MUR-PU.

The licensee summarized its assessment of the impact of the implementation of the proposed MUR-PU on jet impingement and thrust forces (dynamic effects) in Section IV.1.B.viii of Enclosure 2 to the LAR. The licensee stated that it had applied the Leak-Before-Break (LBB) concept to its RCS primary piping. The licensee stated that the evaluation showed that a double-ended guillotine break would not occur and that postulated flaws producing detectable leakage exhibit stable growth and, thus, allow a controlled plant shutdown before the potential exists for catastrophic piping failure. The licensee also confirmed that inputs to the LBB analyses were negligibly impacted by the proposed MUR-PU conditions and concluded that the LBB evaluation would remain acceptable and would be bounded by the existing AOR. Due to the application of LBB, the breaks considered for dynamic effects were limited break ruptures of the smaller attached piping (core flood, decay heat, surge line, steam line, and feedwater line). The licensee concluded that the changes to the design conditions based on the implementation of the proposed MUR-PU would be bounded by the existing analyses and so no changes would be needed for the LBB evaluation.

The NRC staff reviewed the licensee's evaluations related to determinations of pipe rupture locations and their corresponding dynamic effects and determined that the licensee's assessments performed for these areas are acceptable. The AOR related to HELBs, LBB, and dynamic effects resulting from postulated pipe ruptures will remain bounding under the proposed MUR-PU. Given the small magnitude in temperature and pressure increases that accompany the implementation of the proposed MUR-PU, the NRC staff determined that 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, Structural, and Material Component Integrity and Design

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 Oconee HELB AORs, including associated dynamic effects. Based on its review, the

NRC staff concludes that the LAR is acceptable with respect to the structural integrity of the SSCs affected by the proposed MUR-PU. This acceptance is based on the licensee's demonstration that the intent of the regulatory requirements related to the mechanical/structural/material component integrity and design will continue to be satisfied following implementation of the proposed 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 Oconee are not affected by the proposed MUR-PU, as evidenced by the fact that their AORs are unaffected; (2) the RV internals at Oconee also remain unaffected, when considering the impact of the implementation of the proposed MUR-PU on the FIV effects and structural integrity of the RV internals; and (3) the Oconee AORs related to the postulation of HELB locations, including dynamic effects associated with these postulated pipe ruptures, would remain unaffected by the proposed MUR-PU.

Based on the above, the NRC staff concludes that there is reasonable assurance that the structural integrity of SSCs at Oconee will be adequately maintained following implementation of the proposed 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 in accordance with the guidelines in RIS 2002-03. The NRC staff performed this safety evaluation based on the following regulatory and licensing requirements:

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.

UFSAR Section 3.1.24, "Criterion 24 - Emergency Power for Protection Systems (Category B)," states, in part, "In the event of loss of all offsite power, sufficient alternate sources of power shall be provided to permit the required functioning of the Protective Systems."

UFSAR Section 3.1.39, "Criterion 39 - Emergency Power for Engineered Safety Features (Category A)," states, in part, "Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the on-site power system and the off-site power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system."

3.2.3.2 Technical Evaluation

The electrical equipment design information is provided in Section V of Enclosure 2 to the LAR. The NRC staff reviewed the licensee's evaluation of the impact of the proposed MUR-PU on the following electrical systems/components:

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

3.2.3.2.1 AC Distribution System

The AC distribution system is the source of power for the non-safety-related (NSR) buses, the safety-related emergency buses, and the loads supplied by them. According to Section V of Enclosure 2 to the LAR and UFSAR Section 8.2.1, "System Description," Units 1 and 2, are connected by separate main step-up transformers (MSUs) to the 230-kilovolt (kV) switching station then to the 230-kV transmission network. Unit 3 has a similar arrangement from its MSU to the 525-kV switching station to the 525-kV transmission network. A 525/230-kV autotransformer provides a source from the 525-kV switching station to the 230-kV switching station. Each Oconee unit is provided with two physically independent circuits from the 230-kV switching station. One is through each unit's start-up transformer (SUT), and the other one is the back feed through each unit's MSU, the main generator bus, and the unit auxiliary transformer (UAT) with the main generator disconnected.

In accordance with UFSAR Section 8.1.2, "Onsite Power Systems," the on-site power system for each Oconee unit consists of the main generator, UAT, SUT, Keowee Hydro Station (KHS), the SSF, the batteries, the CT4 transformer, and the auxiliary power system.

According to UFSAR Section 8.3.1, "AC Power Systems," if the main generator is available for an Oconee unit, its 19-kV output connects to its UAT which supplies the 6.9- and 4.16-kV auxiliary buses. Otherwise, the auxiliary buses are supplied from the unit's SUT with the latter fed from the 230-kV switching station which has three power sources: (1) the 230-kV transmission network, (2) the 525-kV switching station, and (3) the KHS. If the Oconee generating unit and 230-kV switchyard are unavailable, the KHS, which consists of two hydro units, can supply power to (1) the 230-kV switching station through a common step-up transformer or (2) by an underground feeder through step down transformer CT-4 to the 4.16-kV standby power buses (SPBs) which can supply each unit's main 4.16-kV buses. If the generating units, the 230-kV switching station, and KHS are unavailable, the 100-kV Central Tie Substation or Lee Steam Station can supply power by a 100-kV transmission line connected through transformer CT-5 to the 4.16- kV SPBs. The 4.16-kV SPBs, one supply source to each unit's main 4.16-kV buses, are supplied from KHS through CT-4 and from the 100-kV line through CT-5.

The 4.16-kV auxiliary system for each unit is arranged into two main buses with each bus able to be powered by three power sources: (1) its UAT, (2) its SUT, and (3) the SPBs. Each

4.16-kV main bus provides power to each of the three redundant engineered safeguards (ESG) switchgear bus sections which are arranged so that a single failure does not inhibit their safety functions. The ESG bus sections supply both 4.16-kV NSR loads and safety-related loads, and the 600-V AC, 208-V AC, and 120-V AC systems.

The AC distribution system for each Oconee unit consists of the electrical equipment to supply power to each unit's auxiliary system at the 6.9-kV, 4.16-kV, 600-V AC, 208-V AC, and 120-V AC levels. The licensee stated in Section V of Enclosure 2 to the LAR that each Oconee unit will not experience any load changes at the 6.9-kV, 600 V-AC and 120-V AC voltage levels, but only at the 4.16-kV and 208-V AC voltage levels for the proposed MUR-PU. The load increases at the 4.16-kV and 208-V AC buses are discussed directly below.

The 4.16-kV NSR pumps and associated motors that will experience flow changes at the proposed MUR-PU conditions are provided in Section V of Enclosure 2 to the LAR, and presented in Table 1B of Section 3.2.3.2.2 of this SE. Based on Table 1B, each NSR motor driving a pump experiencing flow changes for the MUR-PU will safely operate within its nameplate ratings under the MUR-PU conditions because its required brake horsepower (BHP) for MUR-PU conditions is less than its nameplate rating. The licensee stated that protective device settings for 4.16-kV motors are based on their nameplate data and, therefore, would be bounded by the current analysis based on Electrical Transient and Analysis Program (ETAP) model calculations. The licensee also stated that the voltages and currents at the 4.16-kV buses would still be bounded by current calculations. No new analyses were performed to determine acceptability of loading increases to the affected buses because the plant's current ETAP model already includes sufficient conservatism to bound all loading increases required to achieve the proposed MUR-PU.

As stated in Section V of Enclosure 2 to the LAR and in request for additional information (RAI) response dated August 17, 2020, the Cameron LEFM feedwater flow instrumentation required for the MUR-PU per Oconee unit consists of two cabinets mounted back-to-back with each cabinet containing a central processing unit (CPU). Each redundant CPU receives pressure transmitter data from two measurement sections/spool pieces with each measurement section supplied by a separate NSR 208-V AC power source. Each power source consists of a panelboard supplied by a 208-V AC motor control center which is supplied by a 112.5-kilovoltampere (kVA) transformer. The increased load requirement for each power source is approximately a 600-VA load on each transformer which is less than 0.5 percent of the transformer's rating. The licensee further stated that the small load addition to each panelboard would have negligible impact and is, therefore, acceptable.

Based on the above, the NRC staff concludes that the entire AC distribution system for each unit will have adequate capacity and capability for the proposed MUR-PU at the 6.9-kV, 600-V AC, and 120-V AC levels because there would be no load increases at these voltage levels, and at the 4.16-kV and 208-V AC levels because any load increases would be within the capability of the affected 4.16-kV and 208-V AC equipment.

3.2.3.2.2 Power Block Equipment

The main power system for each Oconee unit consists of the main generator, MSU, UAT, SUT, and CT-4 as indicated in Section V of Enclosure 2 to the LAR.

Main Generator

Because of the proposed MUR power uprate, the RTP will increase from the previously analyzed core power level of 2568 MWt to 2610 MWt. The turbine-generator for each Oconee unit converts the thermal energy of steam produced in the SGs into mechanical shaft power and then into electrical energy. Per Section V of Enclosure 2 to the LAR and UFSAR Section 10.2.2, the generator for each unit is rated at 934 megawatts electrical (MWe), 1037.937 megavolt-amperes (MVA), 452 megavolt-amperes (reactive) (MVAR), 0.90 power factor, 60 (psig) [pounds per square inch gauge] hydrogen pressure, 19-kV three-phase, and 60 Hz at 1800 rpm. The increase in thermal power of each Oconee unit would result in an increase in each unit's main generator electrical power output as shown in Table 1A, below, which includes updated information provided in the RAI response letter dated August 17, 2020.

Table 1A
Main Generator Outputs Before and After MUR-PU

Unit	Before MUR-PU			After MUR-PU		
1	1023 MVA	909 MWe	470 MVAR	1031.3 MVA	922.7 MWe	460.6 MVAR
2	1029 MVA	919 MWe	463 MVAR	1037 MVA	932.6 MWe	453.5 MVAR
3	1031 MVA	922 MWe	461 MVAR	1037.9 MVA	936.2 MWe	448.1 MVAR

The licensee stated that each Oconee unit generator would be fully functional at the expected outputs provided in Table 1A at the proposed MUR-PU conditions and would operate within the bounds of its capability curve. The generator capability curve is contained within Oconee operating procedures.

Under normal operating conditions, the main generator for each unit supplies power through an isolated phase bus (IPB) to its UAT. In Section V of Enclosure 2 to the LAR, the licensee stated that the IPB for each unit would have enough electrical capacity for the proposed MUR-PU but has experienced cooling problems. Section VI.1.C.vii of Enclosure 2 to the LAR states that the IPB ventilation system for each Oconee unit does not meet its original nameplate design flow resulting in IPB cooling capacity issues during periods of elevated outdoor temperature. As stated in the LAR, the licensee monitors IPB temperatures and provides supplemental cooling or limits the maximum thermal power during periods of elevated outdoor temperature. The licensee plans to upgrade the IPB cooling capability for each unit to eliminate this issue. The NRC staff finds this acceptable because the RCW System, which cools the IPB ventilation system, provides no safety-related function as described in Section 3.1.4.2.2.7 of this SE.

Main Step-Up Transformer (MSU)

As stated by the licensee in Section V of Enclosure 2 to the LAR and in RAI response dated August 17, 2020, the Unit 1 and 2 MSUs are rated at 1000/1120 MVA at 55°C/65°C. The Unit 3 MSU consists of 3 single-phase transformers; each are rated at 373.333 MVA at 65°C or approximately 1120 MVA total. At the proposed MUR-PU conditions, the Unit 1 would be loaded to 984.892 MVA and Unit 2 and 3 MSUs would each be loaded to 991.195 MVA. In each case, the load would be less than the rating of the MSU.

Unit Auxiliary Transformers (1T, 2T, 3T)

The licensee stated that all three UATs (one per unit) are 3-phase, 18.1/6.9/4.16-kV transformers. Units 1 and 2 UATs (1T, 2T) are each rated for 45/60 MVA, and Unit 3 UAT (3T) is rated for 35/70 MVA. Each UAT has two low-voltage windings to serve each unit's auxiliary loads at both the 4.16-kV and 6.9 kV buses. Each UAT is sized to supply its unit's own auxiliary load as well as the ESG equipment of another unit.

The 4.16-kV ESG buses are usually powered by either its UAT or its SUT with each experiencing any required load changes at those buses for the proposed MUR-PU. For that reason, any MUR-PU load changes at the 4.16-kV buses are significant contributors in determining the acceptability of each UAT and SUT for the MUR-PU. Per Section V of Enclosure 2 to the LAR, the 4.16 kV Essential Auxiliary Power system powers the motors of the pumps in Table 1B (based on a similar table provided by the licensee in the LAR) below which they will experience flow changes at the proposed MUR-PU conditions. The licensee uses ETAP to perform electrical calculations including those involving electrical load changes. The horsepower (HP) and BHP ratings for each motor supplying a pump in Table 1B below are for a per pump basis.

Table 1B

Pumps Experiencing Flow Changes for MUR-PU Conditions

Component	Pre-MUR-PU BHP (per pump)	MUR-PU BHP (per pump)	Rated Motor HP (per pump)	Motor BHP in ETAP (per pump)
Hotwell Pump (2)	700	700	1000	
Hotwell Pump (3)	650	650	1000	650
Condensate Booster Pump (2)	1850	1850	2000	2000
D Heater Drain Pump (2)	1321	1477	2000	1700
E Heater Drain Pump (2)	220	220	300	285

For all pumps listed in Table 1B (except the D Heater Drain Pumps), the licensee did not identify a discernable difference in the required BHP from pre- to post-MUR-PU operation due to small differences in their flow rates between the pre- and post-MUR-PU conditions and the minimum slope of their BHP curves in flow regions of interest. For the hotwell pumps, worst case loading is running all three pumps, which is representative of actual operating conditions.

For the loading of each UAT per unit prior to the MUR-PU, the licensee calculated its worst case loading using the values shown in Table 1B above for the ETAP program. The licensee used a BHP in each case in ETAP that was either the motor's nameplate rating or equal to or above what was required for pre-MUR-PU conditions. Even with the stated conservatism built into the calculations, the licensee determined that each UAT could perform successfully prior to the MUR uprate. Those calculations would still be bounding because the ETAP values used for BHP pre-MUR-PU exceed or are equivalent to those required under MUR-PU conditions. The licensee further stated that any increased current through each UAT due to MUR-PU conditions would remain within the UAT's existing rating.

<u>CT-1, 2, 3 Start-Up Transformers</u>

The licensee stated in Section V of Enclosure 2 to the LAR that the SUTs CT-1, CT-2, and CT-3 (one per unit) are three-winding 230 kV/6.9 kV/4.16-kV transformers rated for 33.6 MVA. Each SUT can supply the 4.16-kV loads experiencing changes due to the proposed MUR-PU as identified in the section for the UATs. Existing licensee calculations for the loading of the SUTs using the ETAP program would still be bounding for the proposed MUR-PU conditions for the same reasons as stated above for the UATs. The licensee also stated that secondary voltages for the SUTs would remain within acceptable limits of less than 105 percent loaded or less than 110 percent unloaded.

CT-4 Transformer

The licensee stated in Section V of Enclosure 2 to the LAR that the CT-4 transformer is a two-winding 13.2/4.16-kV transformer rated for 12/16/20 MVA. The licensee stated that there is no load increase from the existing analysis for the CT-4 transformer for either no LOCA or a LOCA at the proposed MUR-PU operation and that the present CT-4 transformer ratings are sufficient for operation at MUR-PU conditions.

Based on the above discussion for the power block equipment for each Oconee unit, the NRC staff concludes that the main generator (including its IPB), the MSU, UAT, SUT, and the CT-4 transformer will each operate within its capability for the proposed MUR-PU as evaluated above for that operating condition.

3.2.3.2.3 DC Power System

In accordance with UFSAR Section 8.3.2, each Oconee unit has a vital 125-V DC system that consists of two independent 125-V DC buses, each supplied by a battery charger and a floating battery. Each Oconee unit also has a separate 125/250-V DC system with a singular bus powered by three battery chargers and two 125-V DC batteries. The 230-kV switchyard, the 525-kV switchyard, and the KHS hydro units each have a separate 125-V DC power system like the Oconee unit.

The licensee stated in Section V of Enclosure 2 to the LAR that any MUR-PU change to the DC distribution systems for each Oconee unit would not be significant and would be within the capacity margins of the system. All DC systems would continue to have adequate capacity and capability for plant operation after the proposed MUR-PU and would be bounded by the existing analyses and calculations of record for the plant. The licensee further stated in its RAI response dated August 17, 2020, that the LEFM gateway computer server would be powered by non-safety AC-regulated power panels that receive backup power from the 250-V DC power distribution system as analyzed by the licensee. The licensee stated that there would be no impact to the vital DC power system.

Based on the above, the NRC staff concludes that the DC power system will operate within its capability after the implementation of the proposed MUR-PU.

3.2.3.2.4 Keowee Hydroelectric Station

In accordance with Section V of Enclosure 2 to the LAR, the equivalent EDG system for the Oconee station is performed by the KHS previously described in Section 3.2.3.2.1 of this SE. The KHS consists of two hydro units each rated at 87.5 MVA and provides a reliable emergency onsite power source for the three Oconee units through the 230-kV switching station and transformer CT-4 to each unit's 4.16-kV ESG buses. The licensee stated that existing analyses indicate there would be no increase in electrical loading of any KHS component under the proposed MUR-PU conditions.

The NRC staff finds the KHS as fully capable to perform its safety-related function as an emergency power source for the Oconee units since no KHS component experiences any load changes for the proposed MUR-PU.

3.2.3.2.5 Switchyard

The 230-kV and 525-kV switching stations were briefly described in Section 3.2.3.2.1 of this SE. Per UFSAR Section 8.2.1, Units 1 and 2, connect to the 230-kV switching station from the high-voltage side of their respective MSUs and overhead 230-kV lines, and Unit 3 connects from the high-voltage side of its MSU by a 525-kV overhead line to the 525-kV switching station. Eight 230-kV transmission lines connect the 230-kV switching station to the 230-kV transmission network. Three 525-kV lines connect the 525-kV switching station to the 525-kV transmission network. An autotransformer (AT-1) connects the 230-kV switching station to the 525-kV switching station.

The 230-kV and 525-kV switching stations connect respectively to the three Oconee units as stated above through their MSUs. The proposed MUR-PU would not impact the rating of each unit's MSU as discussed in Section V of Enclosure 2 to the LAR and in Section 3.2.3.2.2 of this safety evaluation. The respective 230-kV and 525-kV overhead lines connected to the MSUs for the three Oconee units, as described above, are each rated to carry the maximum current for each MSU's high-voltage side at 65°C. Therefore, each overhead line's rated current will not change since each MSU's ratings would not change for the proposed MUR-PU. This is also true of any motor-operated disconnect (MOD) switches and power circuit breakers (PCBs), respectively, in the 230-kV and 525-kV overhead lines' circuit paths since they all are rated to carry the same current as their overhead lines. Similarly, AT-1 is rated to carry the maximum current of the high-voltage side of Unit 3's MSU at 65°C.

The 230-kV switching station connects to each Oconee unit's SUT which provides one source to each unit's 6.9-kV and 4.16-kV auxiliary buses. These 230-kV circuits would be unaffected by the proposed MUR-PU because the ratings of the SUTs would remain unchanged at the proposed MUR-PU conditions based on Section V of Enclosure 2 to the LAR and Section 3.2.3.2.2 of this safety evaluation.

In accordance with Section V of Enclosure 2 to the LAR and the Oconee plant single line diagram (Figure 8.1 in the UFSAR), transformer 5T connects the low side of autotransformer AT-1 to the 4.16-kV auxiliary power upgrade buses 93T and 84T. Those buses have no connection with the 4.16-kV ESG buses which supply loads that would change as a result of the proposed MUR-PU for each unit. The licensee stated that loading on this transformer 5T would be bounded by existing analyses.

The NRC staff finds that the components for the 230-kV and 525-kV switchyards will not be affected by the proposed MUR-PU based on the following: (1) the overhead line from each unit's MSU, the MOD switches and PCBs in its circuit path, and AT-1 will operate within their existing rating which remains unchanged, (2) the current rating of the 230-kV circuit to each unit's SUT will remain unchanged since its SUT rating remains unchanged under MUR-PU conditions, and (3) transformer 5T will be capable of performing under MUR-PU conditions since it is not affected by the 4.16-kV loads that change as a result of the MUR-PU. Therefore, the NRC staff concludes that both 230-kV and 525-kV switchyards will be fully capable of performing for the proposed MUR-PU.

3.2.3.2.6 Grid Stability

Per Section V of Enclosure 2 to the LAR, the licensee completed a Generation System Impact Study that was found to be acceptable for an additional 45 MWe of generating capacity at Oconee station due to the proposed MUR-PU. A Grid Stability Impact Study for the Oconee units identified the need for four studies to be performed: (1) thermal analysis study, (2) short circuit study, (3) stability study, and (4) reactive capability study.

The thermal analysis studies determined that no network upgrades were required to mitigate thermal loading issues for the additional 45 MWe of generating capacity. A short circuit analysis was not performed since the impedance of each Oconee unit generator would not be changed for the proposed MUR-PU conditions and, thus, there would be no fault duty change from each unit generator to the transmission system.

The licensee performed stability studies using a Multiregional Modeling Working Group dynamics model that was updated for appropriate generator and equipment parameters for the additional generation. The estimated load was assumed to be for the 2022 summer modified by assuming some existing generation was turned off to offset the additional generation. The licensee found, based on the assumptions and models used in these stability studies, that the proposed Oconee MUR-PU would not negatively impact the stability of the Duke Energy transmission system.

The licensee performed a reactive capability study by modeling the Oconee plant's generation and MSUs at various taps and system voltage conditions. The study determined that adequate reactive support would exist in the vicinity of the Oconee units for the proposed MUR-PU. Based on the above studies, the licensee determined that the MUR-PU would have no significant effect on grid stability or reliability and that no modifications to the transmission system would be required.

Based on the above, the NRC staff finds that the proposed MUR-PU will not adversely impact the stability of the transmission/grid system.

3.2.3.2.7 Station Blackout

As stated in Section V of Enclosure 2 to the LAR and USFAR Section 8.3.2.2.4, Oconee's station black out (SBO) analysis assumes that all offsite power and both Keowee hydroelectric units are lost for the SBO coping duration. The SSF electrical system includes a diesel generator (DG) that is the credited alternate AC power source. The SSF provides for decay heat removal for the SBO duration for all three units. It can maintain RCS inventory and pressure, remove decay heat, and maintain shutdown margin. The licensee stated that neither the battery systems nor the SSF DG would be affected by the proposed MUR-PU in their design or supplied loads. Therefore, the licensee concluded that the capacity and capability of electrical power systems for an SBO event at the proposed MUR-PU conditions would be bounded by the existing AOR.

The NRC staff finds that the SSF electrical systems will be capable of functioning for an SBO at the proposed MUR-PU conditions since no changes will be made to them and that Oconee will continue to meet the requirements of 10 CFR 50.63 under the proposed MUR-PU conditions.

3.2.3.2.8 Environmental Qualification Program

3.2.3.2.8.1 Regulatory Evaluation

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

Regulations 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 [DBA] during or following which this equipment is required to remain functional.

Regulations 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 [DBA] during or following which the equipment is required to remain functional.

Regulations 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 [10 CFR 50.49] by the safety-related 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 will remain bounded as a result of the proposed MUR-PU.

The licensee evaluated the impact of the proposed MUR-PU on the areas/rooms/zones with regard to temperature, pressure, and radiation. The licensee noted that potential changes in ambient temperatures, system temperatures, system pressures, and potential accident external pressures (e.g., HELB) and accident temperatures were considered during its review. In Section II.1.D.iii of Enclosure 2 to the LAR, the licensee provided its assessment of the effect of the proposed MUR-PU on the UFSAR Chapter 15 transient analyses. The licensee’s response to Section II.1 of RIS 2002-03 is provided in Table II.1-1, “Oconee Analyses,” of the LAR. Table II.1-1 indicates that the existing AOR would remain bounding for all transients. Furthermore, the licensee provided its assessment of EQ of electrical equipment and concluded that its EQ evaluation for the proposed MUR-PU demonstrates that the qualification of EQ components at ONS would not be affected by the MUR-PU, and that temperature, pressure, and radiation parameters would remain enveloped by the qualification documentation.

The NRC staff reviewed Section II.1.D.iii, Table II.1-1, and Section V.1.C of Enclosure 2 to the LAR to determine whether the licensee adequately addressed the impact of the proposed MUR-PU on the EQ of electrical equipment inside and outside of containment (e.g., AB, etc.). The NRC staff confirmed that the current DBAs were performed at 102 percent RTP, which bounds the proposed MUR-PU, and that there would be no impact on the EQ of electrical equipment with respect to temperature, pressure, or radiation due to the MUR-PU in these areas. Based on the above, the NRC staff finds that the existing AORs that established the EQ of electrical equipment would remain bounding for both humidity and chemical spray.

According to 10 CFR 50.49(b)(2), certain NSR electric equipment also needs to be considered for EQ. Although not specifically discussed, the licensee noted that it performed a review of all EQ equipment and did not differentiate between safety and NSR equipment. The licensee also stated that "In addition, an evaluation was performed to determine if there were any EQ zones at ONS that are currently classified as mild environments (with TIDs [total integrated doses] less than 1000 (rads) [radiation absorbed doses] which may be impacted by the 2 [percent] assumed dose increase due to the MUR power uprate, and become harsh environments. No EQ zones were identified that will cross this threshold due to the MUR power uprate." The NRC staff reviewed this information and determined that there is no EQ impact with respect to NSR electric equipment due to the proposed MUR-PU. Based on the above, the NRC staff confirmed that no areas would transition from mild to harsh due to the proposed MUR-PU.

3.2.3.2.8.3 NRC Staff Conclusion Regarding Environmental Qualification of Electrical Equipment

Based on the above, the NRC staff determined that the proposed MUR-PU will not adversely impact the EQ of electrical equipment at ONS since the existing qualification for the normal and accident conditions for electrical equipment inside and outside containment remain adequate and bound the proposed MUR-PU conditions. Therefore, the NRC staff concludes that the proposed MUR-PU will have no adverse impact on the ONS EQ Program as it continues to comply with the requirements of 10 CFR 50.49 and is, therefore, acceptable.

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, the NRC staff determined that each Oconee unit would continue to meet the requirements of 10 CFR 50.49, 10 CFR 50.63, and Oconee's criteria stated in USFAR Sections 3.1.24 and 3.1.39. Therefore, the NRC staff concludes that the LAR is acceptable with respect to electrical engineering evaluations.

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

The NRC staff reviewed the LAR in accordance with RIS 2002-03 concerning the following areas: (1) Chemical Volume and Control System, (2) SG blowdown system, (3) SG tubes, secondary side internal support structures, shell, and nozzles, (4) rapidly propagating fatigue cracks in SG tubes, (5) flow-accelerated corrosion program, and (6) containment coatings program.

3.2.4.1 Chemical Volume and Control System

3.2.4.1.1 Regulatory Evaluation

The NRC staff reviewed the safety-related functional performance characteristics of the Chemical Volume and Control System (CVCS) components as they relate to control of primary water chemistry. The NRC staff's review criteria are based on:

UFSAR Section 3.1.9, “Criterion 9 - Reactor Coolant Pressure Boundary (Category A),” states, in part,

> The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture of significant leakage throughout its design lifetime.

Specific review criteria are contained in SRP, Section 9.3.4, “Chemical and Volume Control System (PWR) (Including Boron Recovery System).”

Note that for Oconee, the CVCS is comprised of multiple systems to include the Chemical Addition and Sampling (CA) System and the High Pressure Injection (HPI) System.

Additional information regarding the design of the CA and HPI systems can be found in Oconee UFSAR, Section 9.3.1 and Section 9.3.2, respectively.

3.2.4.1.2 Licensee Description

In Section IV.1.A.v of Enclosure 2 to the LAR, the licensee stated that the CA and HPI systems will continue to perform their safety functions. The licensee also stated that effects of operation temperature changes from the proposed MUR-PU would be within design limits and that there would be no change to the RCS design or operating pressure.

3.2.4.1.3 NRC Staff Evaluation and Conclusion

The NRC staff reviewed the impacts from the proposed MUR-PU on the CVCS and the effects on primary water chemistry. The NRC staff also reviewed UFSAR Section 9.3.1 and Section 9.3.2.

The NRC staff finds the CVCS design at the proposed MUR-PU conditions acceptable because the HPI system 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 proposed MUR-PU conditions are not expected to have a significant impact on primary water quality and, therefore, the NRC staff has reasonable assurance that the ability of the HPI system to purify the primary water will not be impacted. Additionally, per NUREG-1723, “Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2 and 3,” (Reference 43) the NRC staff has reviewed and approved the licensee’s water chemistry control program. The NRC staff finds that the proposed MUR-PU conditions will not impact the previously approved water chemistry control program. Based on the above, the NRC staff has reasonable assurance that UFSAR, Section 3.1.9, Criterion 9, would continue to be met at the proposed MUR-PU conditions and, therefore, the NRC staff concludes that the LAR is acceptable with respect to its review of the CVCS and primary water chemistry.

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

The NRC staff's review criteria for the SGBS are based on UFSAR, Section 3.1.9, Criterion 9. SRP Section 10.4.8, "Steam Generator Blowdown System" (Reference 44) provides review guidance.

3.2.4.2.2 Licensee Description

In Section IV.1.F of Enclosure 2 to the LAR, the licensee stated that the Replacement Once-Through Steam Generators (ROTSGs) do not operate with blowdown at full-power operation and, therefore, the proposed MUR-PU conditions will not impact the SGBS.
The SGBS and secondary water chemistry parameters are described in Chapters 10 and 18 of the Oconee UFSAR.

3.2.4.2.3 NRC Staff Evaluation and Conclusion

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 Section 10.4.8.

The NRC staff finds that the SGBS will not be impacted by the proposed MUR-PU conditions as the ROTSGs do not utilize blowdown during full-power operation. Therefore, an increase in power level will not impact the necessary SG blowdown flow capacity. Additionally, per NUREG-1723 (Reference 43), the NRC staff has reviewed and approved the licensee's water chemistry control program. The NRC staff finds that the proposed MUR-PU conditions will not impact the previously approved water chemistry control program.

The NRC staff reviewed the licensee's evaluation of the effects of the implementation of the proposed MUR-PU on the SGBS and determined that the licensee has adequately addressed changes in system flow and impurity levels and their effects on the SGBS. The NRC staff further determined that the licensee has demonstrated that the SGBS would continue to be acceptable and will continue to meet the requirements of UFSAR, Section 3.1.9, Criterion 9, following the implementation of the proposed MUR-PU. Therefore, the NRC staff concludes that the LAR is acceptable with respect to the SGBS and secondary water chemistry.

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 Oconee TSs. Oconee TSs 3.4.16, "Steam Generator (SG) Tube Integrity," and 5.5.10, "Steam Generator (SG) Program," govern the SG inspections for Oconee. Details of the Oconee SGs can be found in UFSAR Section 5.2.3.4, "Steam Generators." Specific review criteria for this topic are contained in SRP Section 5.4.2.1, "Steam Generator Materials," (Reference 45) for the SG materials and Section 5.4.2.2, "Steam Generator Program," (Reference 46) for the SG program. Additionally, RIS 2002-03 recommends that the licensee provide a discussion regarding the impacts of the proposed MUR-PU on the structural integrity of the SG tubes, secondary side internal support structures, shell, and nozzles.

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

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

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

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

3.2.4.3.2 Licensee Description

The Oconee ROTSGs are Babcock and Wilcox OTSGs. Each SG contains 15,631 thermally treated SB163 Ni-Cr-Fe (Alloy 690) tubes. In Section IV.1.A.vi of Enclosure 2 to the LAR, the licensee stated that the proposed MUR-PU conditions would be bounded by the design basis thermal hydraulic conditions for the ROTSGs. It also stated that existing loads, stresses, and fatigue values would remain valid and applicable at the proposed MUR-PU conditions. The proposed MUR-PU conditions (e.g., steam flow, temperature, pressure) are described in Table IV-1 of Enclosure 2 to the LAR. The table assumes 10 percent SG plugging. The table shows a slight increase in certain temperatures and flow rates from the current operating conditions.

3.2.4.3.3 NRC Staff Evaluation and Conclusion

The NRC staff evaluated the material provided by the licensee and determined that the changes in operating conditions at the proposed MUR-PU conditions would be relatively small. The changes in operating conditions for the proposed MUR-PU are described in Table IV-1 and assume 10 percent tube plugging. Further, the proposed new operating temperatures and pressures are typical of those used by other plants with OTSGs, which the NRC staff has already approved for use. Similar SGs have operated successfully under these conditions.

With respect to the SG materials, the NRC staff finds that the materials used in the SG will remain acceptable, the fracture toughness of the ferritic materials is adequate, the design will continue to limit the susceptibility of the materials to degradation and corrosion, the materials used in the SG will remain compatible with the environment, the design permits the removal of impurities, and that any degradation that could occur will either be avoided or can be managed. In addition, the NRC staff finds that the impact of the proposed MUR-PU on SG tube vibration and fatigue will remain within acceptable limits for safe operation.

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

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

The NRC staff reviewed the licensee’s evaluation of the effects of the implementation of the proposed MUR-PU on SG tube integrity and finds that the licensee 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 will not change as a result of the implementation of the proposed 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 to RIS 2002-03 states that licensees should address whether the effect of the power uprate on SG tube high cycle fatigue is consistent with NRC Bulletin 88-02, “Rapidly Propagating Fatigue Cracks in Steam Generator Tubes,” dated February 5, 1988 (Reference 47), which was addressed to all licensees of Westinghouse designed reactors with SGs that utilize carbon steel support plates. The bulletin described a steam generator tube rupture (SGTR) event at North Anna Unit 1 that was caused by rapidly propagating fatigue cracks due to high cycle fatigue. The bulletin stated that the presence of all the following conditions could lead to a rapidly propagating fatigue failure such as the one that occurred at North Anna; denting at the upper support plate, a fluid-elastic stability ratio approaching that for the tube that ruptured at North Anna, and absence of effective anti-vibration bar.

3.2.4.4.2 Licensee Description

In Section IV.1.F of Enclosure 2 to its LAR, the licensee stated that the Oconee FIV analyses are bounding for the proposed MUR-PU conditions. The licensee stated that the Oconee SGs are ROTSGs that do not have U-bend tubes. Additionally, the licensee stated that its FIV analyses show an acceptable fluid-elastic instability ratio, that the tube support broach plates cannot be mis-located (which is possible for U-bend anti-vibration bar supports), and that the Oconee broach plates are stainless steel which precludes tube denting. The licensee concluded that these conditions preclude the high cycle fatigue failure mode described in NRC Bulletin 88-02.

3.2.4.4.3 NRC Staff Evaluation and Conclusion

The NRC staff evaluated the information provided by the licensee regarding high cycle fatigue of the SG tubes. The NRC staff finds the licensee's evaluation acceptable because its FIV analysis demonstrated an acceptable fluid-elastic instability ratio, because the tube support broach plates cannot be mis-located, and because the broach plates are stainless steel which prevents tube denting. Because of these provisions and the fact that the Oconee SGs are not U-bend, the high cycle fatigue described in Bulleting 88-02 is not a concern for the Oconee SGs.

The NRC staff reviewed the licensee's evaluation of the effect of the implementation of the proposed MUR-PU on SG tube integrity with respect to SG tube high cycle fatigue concerns as well as tube vibration-induced fatigue. The NRC staff found that the Oconee SGs do not meet the criteria in Bulletin 88-02 for high cycle fatigue. Therefore, in conjunction with the NRC staff evaluation in Section 3.2.4.3 of this safety evaluation, the NRC staff has reasonable assurance that SG tube integrity will be maintained at the proposed MUR-PU with respect to SG tube high cycle fatigue concerns as well as tube vibration-induced fatigue.

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 Oconee is based on the most recent EPRI recommendations. Section 18.3.9, "Flow Accelerated Corrosion Program," of the licensee's UFSAR provides additional detail on the Oconee FAC program and the basis for the program.

3.2.4.5.2 Licensee Description

In Section IV.1.E.iii of Enclosure 2 to the LAR, the licensee stated that there is an established FAC program at Oconee based on EPRI recommendations and best practices. 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 proposed MUR-PU have been incorporated into the plant FAC model. Additionally, the licensee provided sample results from the updated CHECWORKS™ model considering the proposed MUR-PU conditions. The licensee stated that the increase in wear rates due to the proposed MUR-PU is minor and that the existing FAC program is adequate to handle the proposed MUR-PU conditions.

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, per Chapter 18 of the Oconee UFSAR, it is consistent with the initial revision of the GALL Report [see Reference 37], which is not impacted by the proposed MUR-PU. The GALL Report recommends the use of EPRI NSAC-202L, "Recommendations for an Effective Flow-Accelerated Corrosion Program," (Reference 48) which the NRC staff has found acceptable as a basis for a FAC program. The NRC staff finds the increases in predicted FAC wear rates acceptable because they are relatively small, which provides reasonable assurance that the licensee's FAC program will be able to account for these increases while planning inspections. The maximum wear rate change for a given location on the lines evaluated is a few percentage difference, which provides the NRC staff reasonable assurance that the licensee will be able to manage this change with its existing FAC program.

The NRC staff concludes that the licensee 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 the implementation of the proposed MUR-PU. The NRC staff found that the FAC program will provide reasonable assurance that components susceptible to FAC will be managed appropriately after the implementation of the proposed MUR-PU. Therefore, the NRC staff concludes that the proposed MUR-PU is acceptable with respect to the impacts 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 DBA LOCA conditions, considering temperature, pressure, radiation, and chemical effects on the ECCS. 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 SSCs.

Guidance for the NRC staff on specific review criteria is found in SRP Section 6.1.2, Revision 3, "Protective Coating Systems (Paints) – Organic Materials" (Reference 49).

The licensee stated that the containment coatings program is described in UFSAR Section 6.2.1.6, “Coating Materials,” [see Reference 15]. This section states that American National Standards Institute (ANSI) Standard N101.2, “Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities,” (Reference 50) is the licensing basis for Service Level I coatings in the Oconee containment.

3.2.4.6.2 Licensee Description

In Section VII.6.B of Enclosure 2 to the LAR, the licensee stated that the proposed MUR-PU would not have any impact on the containment coatings program. The licensee found that the proposed MUR-PU would have no effect on the containment analyses that would change the design pressure or temperature in a post-LOCA scenario. Therefore, the licensee concluded that the Service Level I coatings qualifications would remain valid at the proposed MUR-PU conditions.

3.2.4.6.3 NRC Staff Evaluation and Conclusion

The NRC staff reviewed the information provided by the licensee as well as the Oconee UFSAR with regard to the containment coatings program. The NRC staff finds that the program is acceptable at the proposed MUR-PU conditions as the proposed conditions in containment after a DBA LOCA due to the MUR-PU are bounded by the current analyses. Because the proposed post-DBA LOCA conditions in containment are bounded by current analyses, the coating qualifications will continue to bound the predicted conditions in containment after a DBA LOCA at the proposed MUR-PU conditions. Therefore, the NRC staff has reasonable assurance that the coatings in containment will not be adversely impacted by the proposed MUR-PU conditions and finds the MUR-PU acceptable with respect to protective coatings. The NRC staff concludes that the protective coatings will continue to meet the requirements of 10 CFR Part 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 of Enclosure 2 to the LAR, the licensee described its evaluation of the impact of the proposed MUR-PU on the ISI Program for ASME Class 1, 2, and 3 components at Oconee, stating, in part, that:

> The ISI Program is discussed in UFSAR Section 5.2.3.12.4. ASME Class 1, 2 and 3 components are examined in accordance with the provisions of the ASME Boiler and Pressure Vessel Code Section XI in effect as specified in10 CFR 50.55a(g) to the extent practical. The MUR power uprate conditions were reviewed for impacts on the ISI Program. The ISI Program will continue to

assess the operational qualification of ASME Class 1, 2, and 3 systems. The Program does not require revision as a result of the MUR power uprate.

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 the implementation of the proposed 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 IST Program for safety-related pumps and valves at Oconee during proposed MUR-PU operation. The Code of Record for Oconee is the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code), 2004 Edition through the 2006 Addenda (Reference 51) in compliance with the requirements of 10 CFR 50.55a. The IST Program provides reasonable assurance of the operational readiness of pumps, valves, and dynamic restraints to perform their safety functions.

3.2.6.2 Safety-Related Valves

The NRC staff reviewed the licensee's analyses of the impact of the proposed MUR-PU on safety-related valves at Oconee. The NRC staff's review included whether any design changes or plant-specific evaluations would be required as a result of the proposed MUR-PU.

The NRC staff examined the overall design change and included plant-specific evaluations of Generic Letter (GL) 89-10, "Safety-Related Motor-Operated Valve Testing and Surveillance," (Reference 52) and GL 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves" (Reference 53). The NRC staff's acceptance criteria for reviewing the safety-related valves analyses are based on 10 CFR 50.55a.

In its submittal, the licensee reviewed the impact of the proposed MUR-PU on the existing DBA for safety-related valves. No changes in RCS flow, design, or operating pressure were made as part of the MUR-PU. The licensee's evaluations concluded that the temperature changes due to the MUR-PU would be 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 MS 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 MU-PU.

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

3.2.6.3 Safety-Related Pumps

The NRC staff reviewed the effects of the implementation of the proposed MUR-PU on the licensee’s analysis for safety-related pumps.

3.2.6.3.1 Regulatory Evaluation

The NRC staff’s acceptance criteria for reviewing the licensee’s analysis of safety-related pumps are based on the requirements in 10 CFR 50.55a.

3.2.6.3.2 Technical Evaluation

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

3.2.6.3.3 NRC Staff Conclusion Regarding Safety-Related Pumps

Based on the above, the NRC staff concludes that the performance of existing safety-related pumps is acceptable with respect to the proposed MUR-PU.

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

In its submittal dated August 17, 2020 (Reference 4), the licensee described the review of the Inservice Examination and Testing Program for safety-related snubbers at Oconee during the proposed MUR-PU operation. The Code of Record for Oconee is the ASME Code for Operation and Maintenance of Nuclear Power Plants (OM Code) 2004 Edition through the 2006 Addenda (Reference 51), in compliance with the requirements of 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 proposed MUR-PU.

Based on the above, the NRC staff concludes that the snubber program will remain acceptable for the proposed MUR-PU conditions.

3.2.6.4.1 Safety-Related Snubbers (Dynamic Restraints)

The NRC staff reviewed the licensee’s evaluation of safety-related snubbers impacted by load increases due to the proposed MUR-PU. The Oconee IST Program includes snubbers that are required to ensure the integrity of the RCPB or required for systems and components that perform a specific function to bring the reactor to the safe shutdown condition, to maintain the safe shutdown condition, or to mitigate the consequences of an accident.

In its submittal dated August 17, 2020, the licensee stated that the assessed systems included those that have snubbers listed in the Oconee snubber program with one exception, the SGs Flush and Drain System was not reviewed. The licensee stated that this system is not inservice during power operation and is not an accident mitigation system and, therefore, is not impacted by the proposed MUR-PU. All other systems with snubbers within the snubber program were found to be acceptable for operation at the MUR-PU power level with no required modifications. No changes were identified for any of the associated piping analyses. The support loads are not changed and the snubbers are not affected by the proposed MUR-PU. Based on the above, the NRC staff concludes that the safety-related snubbers would not be impacted by the proposed MUR-PU.

3.2.6.5 NRC Staff Conclusion

The NRC staff finds that there are no significant changes to operating conditions or the design basis requirements that would affect component performance, test acceptance criteria, or reference values. Therefore, the NRC staff concludes that the IST Program at Oconee is acceptable for the proposed MUR-PU conditions.

3.3 <u>Safety Programs</u>

3.3.1 <u>Radiological Dose Assessment</u>

The NRC staff reviewed the impact of the proposed MUR-PU on analyzed DBA radiological consequences.

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 (Reference 5), which recommends that, for efficiency of review, licensees requesting an MUR-PU identify existing DBA AOR that 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 reanalysis.

For the radiation protection-related sections, the NRC staff conducted an evaluation to verify that annual doses are within the applicable 10 CFR Part 20 annual limit of 100 mrem (millirem), and the 40 CFR Part 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 (Reference 29), and RG 1.183 (Reference 19), at the proposed uprated power level.

3.3.1.1.1 Radiation Protection

The NRC staff’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) UFSAR Section 3.1.70, “Criterion 70 - Control of Releases of Radioactivity to the Environment (Category B),” 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,” (Reference 54) insofar as it relates to plant shielding for spaces/systems which may be used in post-accident operations. Specific review criteria are contained in SRP Section 11.1, “Coolant Source Terms” (Reference 55).

3.3.1.1.2 Design Basis Accident

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

Regulations in 10 CFR 100.11, “Determination of exclusion area, low population zone, and population center distances,” state, in part:

> (a) As an aid in evaluating a proposed site, an licensee should assume a fission produce release[2] from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following:
>
>> (1) An exclusion area of such size that an individual located at any point on its boundary for two hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem[3] or a total radiation dose in excess of 300 rem[2] 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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[2] 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.

[3] 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.

Regulations in 10 CFR 50.67, "Accident source term," state, in part:

> (i) An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (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.

UFSAR Section 3.1.11, "Criterion 11 - Control Room (Category B)," states, in part:

> The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10 CFR [Part] 20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

3.3.1.1.3 Other Regulatory Guidance

Regulatory Guide 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," dated July 2000 (Reference 19), 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 (Reference 5), recommends that to improve the efficiency of the NRC staff's review, licensees requesting an MUR-PU should identify existing DBA AOR that 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 SE documents the NRC staff's review of the impact of the proposed changes on analyzed DBA radiological consequences.

License Amendment Nos. 338, 339, and 339, dated June 1, 2004, for Oconee Nuclear Station, Units 1, 2, and 3, respectively (Reference 56), approved a full-scope implementation of the Alternative Source Term radiological analysis methodology in accordance with 10 CFR 50.67 to perform the radiological consequences analyses of DBAs as described in RG 1.183.

3.3.1.2 Technical Evaluation

3.3.1.2.1 Radiation Protection

3.3.1.2.1.1 Radioactive Waste Management Systems

The NRC staff reviewed the radioactive waste management systems (Liquid Waste, Gaseous Waste, and Solid Waste Management Systems) as described in UFSAR Chapter 11, Sections 11.2, 11.3, and 11.4, respectively, and summarized below. 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 Management System

The Liquid Waste Management System is designed to receive, segregate, process, recycle, and discharge liquid wastes. The system piping provides pressure boundary piping and containment isolation functions for mitigating events. The system also is credited to store and minimize leakage of radioactive fluid to the environment. The system design considers potential personnel exposure and ensures that quantities of radioactive releases to the environment are ALARA.

Liquid wastes are accumulated in storage tanks according to the waste source and expected process train. The AB coolant treatment header has been redesigned to facilitate the processing of liquid wastes from the high activity waste tanks, low activity waste tanks, and the miscellaneous waste holdup tanks in the Radwaste Facility. The liquid wastes are directed to the Radwaste Facility for processing by filtration and/or demineralization to segregate impurities for ultimate disposal.

The Liquid Waste Management 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 tailrace of the Keowee Hydroelectric Plant if the water is required to be monitored during the release, discharged to the Chemical Treatment Pond #3 if monitoring is not required, or retained for further processing. A permanent record of liquid releases is provided by analyses of known volumes of waste by the licensee.

The existing capacities of various holding, processing, and storage tanks would be sufficient at the proposed MUR-PU conditions because system flow rates and liquid inventories would not be affected by the MUR-PU. The volume of liquid waste primarily depends on reactor coolant bleed, SGs blowdown, and leakage from various components. The volume generated during normal operation will not change because of the MUR-PU. Implementing the MUR-PU will not increase the volume of liquid waste processed by the Liquid Waste Management System. The concentration of radioactive nuclides in the Liquid Waste Management System is expected to increase by a maximum of 1.64 percent. This increase in nuclide concentration would not significantly impact the Liquid Waste Management System operation.

3.3.1.2.1.1.2 Gaseous Waste Management System

All components in the AB and Interim Radwaste Building that can contain potentially radioactive gases are vented to a vent header. The vent gases are subsequently drawn from this vent header by one of two waste gas compressors or a waste gas exhauster. The waste gas compressor discharges through a waste gas separator to one of two waste gas tanks. The waste gas tanks and the waste gas exhauster discharge to the unit vent after passing through a filter bank consisting of a prefilter, an absolute filter, and a charcoal filter.

Oconee 1 and 2 share a GWD System. Oconee 3 has a separate Waste Gas Disposal System that can be interconnected to the GWD System for Oconee 1 and 2 through double isolation valves between the vent headers. These are normally operated separately but may be tied together to facilitate maintenance of either of the systems.

At the proposed MUR-PU conditions, the required containment, confinement, and filtering capacities of the Gaseous Waste Management System and the capacities of its various decay and storage tanks would be sufficient because the MUR-PU does not materially affect the system flow rates or gas volumes and, therefore, these systems would not be impacted by the MUR-PU.

3.3.1.2.1.1.3 Solid Waste Management System

The Solid Waste Management System provides the ability to process and package solid wastes for shipment to an offsite NRC-approved agreement state licensed burial facility. Low-level trash such as dry waste and low specific activity spent filters are prepared for shipment to a processor or directly to a licensed disposal facility.

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 licenses 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 Management System can be used as a decontamination area when needed.

The proposed MUR-PU would not have an effect on the generation of solid waste volumes; therefore, the quantities of low-level, compressible, radioactive wastes (e.g., paper, rags, plastics, clothing, respiratory filters) would not increase because of the MUR-PU. 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 would be no increase in solid waste generation due to the proposed MUR-PU, there is not an increase in solid waste storage requirements or shipments from the plant.

3.3.1.2.1.2 Radiological Effluents

The NRC staff reviewed the radioactive source term associated with the proposed MUR-PU 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.

During normal operations, the controls instituted by licensee programs and processes for the release rates of radwaste systems do not change with operating power. Thus, no impact on routine releases is anticipated due to the proposed MUR-PU. Actual, measured radiological activity 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 measured activities are a small fraction of annual limits. The effluent doses are determined in accordance with the offsite dose calculation manual which is a licensee-controlled document required under the administrative controls section of the Technical Specifications. The offsite dose calculation manual methodologies ensure that doses to the public remain within regulatory dose limits and are ALARA. The proposed MUR-PU will not result in resultant doses being more than a small fraction of annual limits because 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 proposed license amendment is acceptable with respect to the radiological effluents from radwaste systems.

3.3.1.2.1.3 Individual or Cumulative Occupational Radiation Exposure

Oconee was designed with sufficient margin for higher-than-expected radiation sources. During normal and post-accident conditions, radiation levels in most areas of the plant increase by no more than the percentage increase in power level. The licensee reviewed the radiological effects for the proposed MUR-PU to verify that expected radiological conditions at the MUR-PU 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 BOP 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 proposed MUR-PU.

The current ALARA program practices at Oconee, 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 does rates associated with the proposed MUR-PU. Thus, the increased radiation sources resulting from the proposed MUR-PU, 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 Part 20 and ALARA and is, therefore, considered acceptable. Therefore, the NRC staff concludes that the proposed license amendment is acceptable with respect to the effect of the proposed MUR-PU 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 the proposed MUR-PU, as they relate to the radiological consequences of DBA analyses. RIS 2002-03 recommends that to improve efficiency of the NRC staff's review, licensees requesting an MUR-PU should first identify existing DBA analyses of record that bound plant operation at the proposed uprated power level. Secondly, for any existing DBA analyses of record that do not bound the proposed uprated power level, the licensee should provide a detailed discussion of the re-analysis. Information regarding these analyses was provided by the licensee in Section II of Enclosure 2 to the LAR, consistent with RIS 2002-03. The findings of this SE are based on the descriptions and results of the licensee's analyses and other supporting information docketed by the licensee.

The Oconee accident and safety analyses including LOCA events are addressed in Chapter 15 of the Oconee UFSAR.

In its LAR, the licensee discussed each analysis in support of the proposed MUR-PU, including the assumed core power level in each analysis and whether the analysis remains bounding for the MUR-PU. As previously discussed, the MUR-PU would increase the Oconee authorized core power level from 2568 MWt to 2610 MWt, which is an increase of approximately 1.64 percent RTP, based on the use of the CheckPlus™ LEFM system. In accordance with the CLB that 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 2619 MWt (102 percent of the current RTP of 2568 MWt), which bounds operation at the proposed MUR-PU operating conditions.

The NRC staff reviewed the impact of the proposed 1.64 percent MUR-PU on DBA radiological consequence analyses, as documented in Chapter 15 of the UFSAR. The NRC staff confirmed that the CLB dose consequence analyses remain bounding at the proposed MUR-PU power level of 2610 MWt with a margin that is within the assumed approximately 0.34 percent uncertainty associated with advanced flow measurement techniques, including the use of the CheckPlus™ LEFM system credited by the licensee. Specific areas of review included:

- Major Reactor Coolant System Pipe Ruptures (Loss-of-Coolant Accidents) - UFSAR Section 15.14.4
- Fuel Handling Accident - UFSAR Section 15.11
- Major Secondary System Pipe Rupture - UFSAR Section 15.13
- Steam Generator Tube Rupture - UFSAR Section 15.9

- Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) – UFSAR Section 15.12
- Single Reactor Coolant Pump Locked Rotor - UFSAR Section 15.6.5
- Waste Gas Decay Tank Rupture - UFSAR Section 15.10

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. As part of the proposed MUR-PU, Oconee is adding an “LEFM System Trouble” alarm window to the control room alarm panel to alert the control room operators when there is a problem with the LEFM.

The NRC staff confirmed that the applicable CLB dose consequence analyses remain bounding at the proposed MUR-PU power level of 2610 MWt with a margin that is within the assumed 0.34 percent 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 Oconee UFSAR, in addition to information in the LAR, the NRC staff verified independently 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 Oconee at the proposed 1.64 percent MUR-PU power level will continue to meet the applicable dose limits. The NRC staff has reasonable assurance that with the approval of the LAR, Oconee will continue to provide sufficient safety margins, with adequate defense-in-depth, to address unanticipated events and to compensate for uncertainties in accident progression, analysis assumptions, and input parameters. Therefore, the NRC staff concludes that the proposed MUR-PU 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, or 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[4] (NSCA) to ensure that the SSCs

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

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

By a letter dated December 29, 2010, “Oconee Nuclear Station, Units 1, 2, and 3, Issuance of Amendments Regarding Transition to a Risk-Informed, Performance-Based Fire Protection Program in Accordance with 10 CFR 50.48(c)” (Reference 57), Oconee transitioned from the 10 CFR Part 50, Appendix R, to the NFPA-805 risk-informed, performance-based licensing basis.

3.3.2.2 Technical Evaluation

The NRC staff reviewed Section VII.6.A of Enclosure 2 to the LAR for the Oconee-specific evaluations of each item outlined in RIS 2002-03. 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 or safe-shutdown capability as noted in RIS 2002-03, Attachment 1, Sections II and III.

The licensee developed its LAR consistent with the guidelines in RIS 2002-03. In its LAR, the licensee re-evaluated the applicable SSCs and safety analyses at the proposed MUR-PU level of 2610 MWt against the previously analyzed core power level of 2568 MWt, an increase of 42 MWt or approximately 1.64 percent of RTP.

In Section VII.6.A of Enclosure 2 to the LAR, the licensee stated that the proposed MUR-PU would not change or modify the credited equipment necessary for post-fire safe shutdown or reroute or relocate any essential cables or components credited by the safe shutdown analysis. The licensee also stated that the installation of the LEFM components was reviewed under the administrative controls of the Oconee’s design change process and found to not adversely impact safe shutdown. The licensee stated that building heatup would be minimally impacted such that currently credited fire protection manual actions would not be prevented from being accomplished by their required time, and no new operator actions for fire protection were identified. Further, the HPSW System credited for supplying water to the fire headers as discussed in the Section VI.1.C.iii of Enclosure 2 to the LAR, would not be adversely impacted by the proposed MUR-PU since the system remains bounded by the existing AOR.

The NRC staff reviewed the above statements against the Oconee fire protection program NSCA and determined that the licensee has adequately accounted for the effects of the increase in decay heat and potential impacts to the ability of credited post-fire safe shutdown systems and functions to achieve and maintain fuel in a safe and stable condition in the event of a fire.

The NRC staff determined that the installation of the LEFM equipment would introduce a minimal increase in additional combustible loading and fire severity in affected fire areas. This minimal increase, however, is not expected to challenge any existing active or passive fire protection features in those fire areas. The proposed MUR-PU would also result in a minimal increase in decay heat load in the plant, which the NRC staff determined will not impede or prevent the performance of any credited post-fire operator manual actions in the affected fire areas. Further, since the proposed MUR-PU would not affect any cable or equipment credited in Oconee's NSCA, there will be no adverse impact on any systems or functions credited for achieving and maintaining the plant in a safe and stable condition post-fire. Therefore, NRC staff finds the proposed MUR-PU acceptable with respect to fire protection and concludes that the fire protection program will continue to meet the requirements of 10 CFR 50.48(c) following the implementation of the proposed MUR-PU.

3.3.2.3 NRC Staff Conclusion Regarding Fire Protection

Based on the above, the NRC staff concludes that the proposed MUR-PU will have no adverse impact on any active or passive fire protection features or the post-fire safe shutdown capability at Oconee. Therefore, NRC staff finds the proposed MUR-PU acceptable with respect to fire protection and concludes that the fire protection program would continue to meet the requirements of 10 CFR 50.48(c) following the implementation of the proposed MUR-PU.

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

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

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

Guidance in SRP Section 13.5.2.1, Revision 2, "Operating and Emergency Operating Procedures," (Reference 58) 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 the NRC staff's review included licensee-identified changes to operator actions, human-system interfaces, procedures, and training needed for the proposed MUR-PU.

Guidance for the 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 responses to these questions are described in Sections VII.1 through VII.4 of Enclosure 2 to the LAR. The following sections are the NRC staff's evaluation of the licensee's responses to these questions.

3.3.3.2.1 Operator Actions

RIS 2002-03, Attachment 1, Section VII.1, states that the licensee should make 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

In Section VII.1 of Enclosure 2 to the LAR, the licensee stated that the engineering change process ensures that all discipline, system, and program impacts including updates to emergency and abnormal operating procedures, Time Critical Operator Actions (TCOAs), control room controls, Human-Machine Interface displays (including the safety parameter display system) and alarms, the control room plant reference simulator, and the operator training program are captured when a modification occurs. Therefore, all modification packages identified as affected by the proposed MUR-PU will update these items. The design change process ensures that any impacted normal, abnormal, and emergency operating procedures having operator actions are revised prior to the implementation of the proposed MUR-PU, if required. An evaluation was performed of the operator actions and no impacts were identified. The TCOAs were evaluated individually in system evaluations and against the ONS licensing analyses presented in Enclosure 2, Section II, "Accidents and transients for which the existing analyses of record bound plant operation at the proposed uprated power level," to ensure that they would remain bounded. All of the TCOAs would remain unchanged following the proposed MUR-PU.

The NRC staff finds that the statements provided by the licensee are in conformance with Section VII.1 of Attachment 1 to RIS-2002-03 and concludes that the proposed MUR-PU 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

RIS 2002-03, Attachment 1, Section VII.2.A, states that the licensee should make a statement confirming that it has identified all required changes to the current emergency operating procedures and abnormal operating procedures to ensure that any changes do not adversely affect defense-in-depth or safety margins.

NRC Staff Evaluation

In Section VII.2.A of Enclosure 2 to the LAR, the licensee stated that the proposed MUR-PU would be implemented under the administrative controls of the Oconee license amendment request and design change processes which provide the administrative controls relevant to identifying impacted procedures. The design change process ensures that any impacted emergency and abnormal operating procedures would be revised prior to the implementation of the MUR-PU.

The NRC staff finds that the statements provided by the licensee are in conformance 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

RIS 2002-03, Attachment 1, Section VII.2.B, states that the licensee should make 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 changes do not adversely affect defense-in-depth or safety margins.

NRC Staff Evaluation

In Section VII.2.B of Enclosure 2 to the LAR, the licensee stated that a review of plant systems has indicated that only minor modifications are necessary. Oconee follows established engineering and licensing procedures to ensure that the necessary minor modifications are installed prior to implementing the proposed MUR-PU.

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

3.3.3.2.4 Control Room Plant Simulator

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

NRC Staff Evaluation

In Section VII.2.C of Enclosure 2 to the LAR, the licensee stated that a review of the plant simulator would be conducted, and necessary changes made, prior to implementing the proposed MUR-PU. The MUR-PU would be implemented under the administrative controls of the Oconee LAR and design change processes which provide the administrative controls relevant to identifying simulator and training impacts. The design change process ensures that any impacted emergency and abnormal operating procedures would be revised prior to the implementation of the MUR-PU. As part of this process, any necessary changes to the simulator would be identified and implemented during the design change review process.

The NRC staff finds that the statements provided by the licensee are in conformance 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

RIS 2002-03, Attachment 1, Section VII.2.D, states that the licensee should make a statement confirming that it has identified all required changes to the operator training program to ensure that any changes do not adversely affect defense-in-depth or safety margins.

NRC Staff Evaluation

In Section VII.2.D of Enclosure 2 to the LAR, the licensee stated that it is anticipated that the proposed MUR-PU would have no impact on the current operator training program. Operator training on the plant changes required to support the MUR-PU would be completed prior to the implementation of the MUR-PU. Training on the operation and maintenance of the Cameron® LEFM CheckPlus System would be developed and completed prior to the implementation of the MUR-PU.

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

3.3.3.2.6 Modifications

RIS 2002-03, Attachment 1, Section VII.3, states that the licensee should make a statement confirming its intent to complete the modifications identified in Section VII.2 (including the training of operators), prior to the implementation of the MUR-PU.

NRC Staff Evaluation

In Section VII.3 of Enclosure 2 to the LAR, the licensee stated that the changes/modifications to the simulator and the associated manuals and instructional materials would be implemented in accordance with the Oconee engineering change process to capture all plant changes as a result of the proposed MUR-PU. The licensee would complete the modifications related to the proposed MUR-PU and complete the training of operators prior to the implementation of the MUR-PU.

The NRC staff finds that the statements provided by the licensee are in conformance with Section VII.3 of Attachment 1 to RIS-2002-03 and concludes that the licensee has confirmed its intent to complete the modifications identified in Section VII.2 prior to the implementation of the proposed MUR-PU.

3.3.3.2.7 Temporary Operation above Licensed Full Power Level

RIS 2002-03, Attachment 1, Section VII.4, states that the licensee should make 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

In Section VII.4 of Enclosure 2 to the LAR, the licensee stated that operating procedures have been reviewed and required changes would be documented and implemented as part of the normal engineering change and licensing processes. Further, the licensee stated that the procedure related to temporary operation above full steady-state licensed power levels would be reviewed and modified, as necessary.

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

3.3.3.3 NRC Staff Conclusion Regarding Human Factors

The NRC staff has completed its human factors review of the LAR and has determined that the licensee has adequately considered, or will consider, the impact of the proposed MUR-PU on operator actions, emergency and abnormal operating procedures, control room components, the control room plant simulator, and operator training programs. The NRC staff has determined that the statements provided by the licensee 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 proposed MUR-PU is acceptable with respect to human factors.

3.4 Spent Fuel Pool Criticality Analysis

The licensee addressed the effects of the proposed MUR-PU on SFP criticality analysis in its supplement dated April 6, 2020. The SFP criticality analysis is described in Section 9.1.2.3.2 of the Oconee UFSAR (Reference 15). The Oconee AOR for the SFP criticality analysis was submitted for NRC approval by letter dated December 28, 2000 (Reference 59). By NRC letter dated April 22, 2002 (Reference 60), the NRC issued Amendment Nos. 323, 323, and 324 for Oconee, Units 1, 2, and 3, respectively, approving the SFP criticality analysis.

The criticality analysis ensures that there is a 95 percent probability at a 95 percent confidence level (95/95) that the effective neutron multiplication factor (k-effective) will remain below 1.0 in unborated water and below 0.95 with partial credit for soluble boron. 10 CFR 50.68, "Criticality accident requirements," requires that if credit for soluble boron is taken, the k-effective of the SFP must not exceed 0.95 at 95/95, and must not exceed 1.0 at 95/95 if flooded with unborated water.

The effects of an MUR-PU would result in slightly different discharge characteristics of the spent fuel, such as reactivity. In its supplement dated April 6, 2020, the licensee stated that the SFP criticality AOR was reviewed for the proposed MUR-PU conditions and was determined to remain bounding. The licensee evaluated the effects of the proposed MUR-PU on SFP criticality using its NRC-approved criticality analysis methodology. This analysis considered changes in parameters such as fuel temperature, moderator temperature, and fuel power density due to the proposed MUR-PU. The licensee concluded that discharged spent fuel assemblies at MUR-PU conditions would remain less reactive than spent fuel assemblies discharged at non-MUR-PU conditions. With the MUR-PU, fuel assemblies would experience a higher burnup, which, in general, would result in reduced reactivity at end of life. The NRC staff reviewed the NRC-approved criticality safety analysis and determined that it will remain bounding at the proposed MUR conditions.

Based on the above, the NRC staff determined that the proposed MUR-PU will not affect the applicability of the SFP criticality analyses, because Oconee will continue to demonstrate that the k-effective of the SFP storage racks remains below the regulatory limits. Therefore, the NRC staff concludes that the proposed MUR-PU is acceptable with respect to SFP criticality.

3.5 Changes to the RFOL and Technical Specifications Associated with the Proposed MUR-PU

3.5.1 Regulatory Evaluation

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

3.5.2 Technical Evaluation

In its LAR, the licensee proposed changes to the RFOL and TS as discussed below. These changes are proposed for each of the three ONS units.

3.5.2.1 RFOLs and TS 1.1

The RFOLs Item 3.A and TS 1.1 would be revised to increase the maximum core power level from 2568 MWt to 2610 MWt.

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

> RTP shall be a total reactor core heat transfer rate to the reactor coolant of 2568 MWt*.

The asterisk (*) would be added and refers to the following footnote that would be added to the section:

> *Following implementation of MUR on the respective unit, the value of RTP shall be 2610 MWt.

NRC Staff Evaluation

As discussed throughout this safety evaluation, the NRC staff has determined that the licensee’s proposal to increase the RTP from 2568 MWt to 2610 MWt as part of an MUR-PU is acceptable. Therefore, the NRC staff finds that changing the RTP from 2568 MWt to 2610 MWt, as stated in the RFOLs and in TS 1.1 is acceptable.

3.5.2.2 TS Table 3.3.1-1

The licensee proposed to revise TS Table 3.3.1-1, “Reactor Protective System Instrumentation,” by adding the footnotes (f) and (g) to the Allowable Value for Function 1.a, Nuclear Overpower High Setpoint. The proposed footnotes would state:

> (f) If the high accuracy indication (including the Leading Edge Flow Meter) is unavailable, reduce the overpower trip setpolnt as specified in Selected Licensee Commitment 16.7.18, “Leading Edge Flow Meter.”

(g) Following implementation of MUR on the respective unit, the value shall be 79.3% RTP.

NRC Staff Evaluation

The licensee evaluated the Reactor Protection System (RPS) trip function allowable values to ensure that the values after the implementation of the proposed MUR-PU were bounded by the analyses. The high flux trip for four RCP operation was reevaluated at a thermal power value bounded by the uprated power level. The evaluation demonstrated that the existing trip value of 105.5 percent RTP can be maintained without compromising required limits. As discussed in Section 3.1.5 of this safety evaluation, in areas (e.g., accident/transient analyses, components, systems) for which the existing AOR bound plant operation at the proposed uprated power level, the NRC staff determined that the proposed values are acceptable without a detailed review. Thus, the NRC staff finds the proposed footnotes acceptable.

As part of the proposed footnote (f), the licensee proposed to add SLC 16.7.18, "Leading Edge Flow Meter." The SLC would add requirements for plant operation when the LEFM is not functional. The SLC would allow operation at the uprated condition with the LEFM not functional for up to 72 hours with some restrictions. If the LEFM cannot be restored to functional within 72 hours, the use of LEFM data as input to the thermal power calorimetric calculations would be suspended and the thermal power level would be reduced to the pre-uprate value of 2568 MWt. The SLC would also direct various settings and trip setpoints be adjusted to correspond to the pre-uprate power value as required. The SLC would establish surveillances to verify that the LEFM is functional. As discussed throughout this safety evaluation, the NRC staff finds this change acceptable.

3.5.2.3 TS Figures 3.4.3-1 through 3.4.3-9

The licensee proposed to revise the applicability of the RCS heatup curves in TS Figures 3.4.3-1 through 3.4.3-9 to reflect updated EFPY limitations. The limits are based on revised reactor vessel material evaluations. The Unit 1 applicability would be changed from the first 54 EFPY to the first 44.6 EFPY; the Unit 2 applicability would be changed from the first 54 EFPY to the first 45.3 EFPY; and the Unit 3 applicability would be changed from the first 54 EFPY to the first 43.8 EFPY.

NRC Staff Evaluation

The applicability of the RCS heatup curves was proposed to be modified to incorporate more conservative EFPY values based on updated reactor vessel material evaluations. The NRC staff finds these changes to be acceptable as discussed in Section 3.2.1.2 of this safety evaluation and since the applicability is more conservative and reflects current material properties.

3.5.2.4 TS 3.4.4

The licensee proposed to revise TS LCO 3.4.4 for operation with three RCPs. The change would reduce the maximum THERMAL POWER for three RCP operation from 75 percent to 73.8 percent. The 73.8 percent RTP at MUR-PU conditions corresponds to the thermal power level at 75 percent RTP prior to the uprate, or 1926 MWt. The licensee would accomplish this change through the addition of a footnote to LCO 3.4.4.b.1, as follows:

* Following implementation of MUR on the respective unit, the value shall be 73.8% RTP.

NRC Staff Evaluation

For three RCP operation the power level would be reduced from 75 percent of the pre-MUR-PU power to 73.8 percent of the post-MUR-PU power. This would retain 1926 MWt as the three RCP operating thermal limit. The post-MUR-PU high flux trip setpoint for three RCP operation would be 79.3 percent of the uprated power level. This value was determined by adding 5.5 percent to the operating point of 73.8 percent. This change would result in a slight increase in power for the three RCP operation high flux trip setpoint. The licensee evaluated the new setpoint and determined that it is bounded by the analyses. Therefore, the NRC staff find this TS change acceptable.

3.5.3 Conclusion

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

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined 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, which was published in the *Federal Register* on June 2, 2020 (85 FR 33745), and there has been no public comment on such finding. 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.

6.0 CONCLUSION

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

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SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3 - ISSUANCE OF AMENDMENT NOS. 420, 422, AND 421 RE: MEASUREMENT UNCERTAINTY RECAPTURE POWER UPRATE (EPID L-2020-LLS-0000) DATED JANUARY 26, 2021

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